KG&E
THE ELECTRIC COMPANY
KANSAS GAS AND ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

July 20, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

KMLNRC 84-118
Re: Docket No. STN 50-482
Ref: Letter dated 6/28/84 from BJYoungblood, NRC,
to GLKoester, KG&E
Subj: Technical Specifications

Dear Mr. Denton:

The Referenced letter requested that page changes to the "Proof and Review" version of the Wolf Creek Technical Specifications be submitted for NRC approval.

Transmitted herewith are the Wolf Creek Technical Specifications "marked up" to indicate changes. These "marked up" pages include all changes issued to Callaway in NUREG-1058 that are applicable to Wolf Creek, as well as any plant specific changes and justifications required for Wolf Creek. A listing and categorization of all changes made in the "marked up" copy of the Wolf Creek Technical Specifications has also been provided.

This information is hereby incorporated into the Wolf Creek Generating Station, Unit No. 1, Operating License Application.

Yours very truly,

Glenn L Koester

GLK:bb
Attach

xc: PO'Connor (2)
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OATH OF AFFIRMATION

STATE OF KANSAS)
) SS:
COUNTY OF SEDGWICK)

I, Glenn L. Koester, of lawful age, being duly sworn upon oath, do depose, state and affirm that I am Vice President - Nuclear of Kansas Gas and Electric Company, Wichita, Kansas, that I have signed the foregoing letter of transmittal, know the contents thereof, and that all statements contained therein are true.

KANSAS GAS AND ELECTRIC COMPANY

ATTEST:

By Glenn L Koester
Glenn L. Koester
Vice President - Nuclear

EDProthro
E.D. Prothro, Assistant Secretary

STATE OF KANSAS)
) SS:
COUNTY OF SEDGWICK)

BE IT REMEMBERED that on this 20th day of July, 1984, before me, Evelyn L. Fry, a Notary, personally appeared Glenn L. Koester, Vice President - Nuclear of Kansas Gas and Electric Company, Wichita, Kansas, who is personally known to me and who executed the foregoing instrument, and he duly acknowledged the execution of the same for and on behalf of and as the act and deed of said corporation.

IN WITNESS WHEREOF, I have hereunto set my hand and affixed my seal the date and year above written.

EVELYN L. FRY

Evelyn L. Fry
Evelyn L. Fry, Notary

My Commission expires on August 15, 1984.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
VIII	Table 3.4-1	Typo
IX	3/4.5.5	Consistent with Callaway Technical Specifications (T.S.)
IX	3/4.5.6	Consistent with Callaway T.S.
IX	3/4.6.4	Consistent with Callaway T.S.
XII	3/4.8.4.2	Consistent with Callaway T.S.
XII	Table 3.8-2	Consistent with Callaway T.S.
XV	3/4.3.4	Consistent with Callaway T.S.
XVI	3/4.5.5	Consistent with Callaway T.S.
XVI	3/4.5.6	Consistent with Callaway T.S.
1-2	1.8	Consistent with Callaway T.S.
1.2	1.11	Typo
1.4	1.20	Typo
1-6	1.38	Consistent with Callaway T.S.
2-1	2.1.1	Consistent with Callaway T.S.
2-2	Figure 2.1-1	Consistent with Callaway T.S.
2-5	Table 2.2-1 (items 1 through 11)	Wolf Creek specific values provided.
2-6	Table 2.2-1 (items 12 through 15)	Wolf Creek specific values provided.
2-7	Table 2.2-1	Consistent with Callaway T.S.
2-7	Table 2.2-1 (items 18b through 18f)	Wolf Creek specific values provided.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
2-8	Table 2.2-1, Table Notations (K_2)	Wolf Creek specific value provided.
2-9	Table 2.2-1, Table Notations	Consistent with Callaway T.S.
2-9	Table 2.2-1, Table Notations (Note 2)	Wolf Creek specific value provided.
2-10	Table 2.2-1, Table Notations (K_4)	Wolf Creek specific value provided.
2-11	Table 2.2-1, Table Notations (Note 4)	Wolf Creek specific value provided.
B 2-2	2.1.2	Consistent with Callaway T.S.
B 2-4	Power Range Neutron Flux	Typo
B 2-5	Intermediate & Source Range	Consistent with Callaway T.S.
B 2-5	Overpower Δ T	Consistent with Callaway T.S.
B 2-6	Pressurizer Pressure	Consistent with Callaway T.S. and Typo
B 2-6	Pressurizer Water Level	Consistent with Callaway T.S.
B 2-8	Reactor Trip System Interlocks	Consistent with Callaway T.S.
3/4 1-1	3.1.1.1 (* note)	Consistent with Callaway T.S.
3/4 1-4	3.1.1.3 (# note)	Consistent with Callaway T.S.
3/4 1-6	3.1.1.4 (* note)	Consistent with Callaway T.S.
3/4 1-7	3.1.2.1.a & 3.1.2.1.b	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 1-8	3.1.2.2 (* note)	Consistent with Callaway T.S.
3/4 1-10	3.1.2.4 (* note)	Consistent with Callaway T.S.
3/4 1-10	3.1.2.4 (Action)	Consistent with Callaway T.S.
3/4 1-11	3.1.2.5.a.1) & 3.1.2.5.b.1)	Consistent with Callaway T.S.
3/4 1-11	4.1.2.5.a.3)	Consistent with Callaway T.S.
3/4 1-12	3.1.2.6, 3.1.2.6.a & 3.1.2.6.b	Consistent with Callaway T.S.
3/4 1-12	Action a,b,&c	Consistent with Callaway T.S.
3/4 1-13	4.1.2.6	Consistent with Callaway T.S.
3/4 1-14	3.1.3.1 (* note)	Consistent with Callaway T.S.
3/4 1-15	Action c	Consistent with Callaway T.S.
3/4 1-15	4.1.3.1.2	Typo
3/4 1-17	3.1.3.2	Consistent with Callaway T.S.
3/4 1-17	3.1.3.2 Action a & b	Consistent with Callaway T.S.
3/4 1-17	4.1.3.2	Consistent with Callaway T.S.
3/4 1-18	3.1.3.3 (# note)	Consistent with Callaway T.S.
3/4 1-20	3.1.3.5 (* note)	Consistent with Callaway T.S.
3/4 1-21	3.1.3.6	Consistent with Callaway T.S.
3/4 2-1	3.2.1	Consistent with Callaway T.S.
3/4 2-1	3.2.1 (Action 6)	Consistent with Callaway T.S.
3/4 2-1	3.2.1 (* note)	Consistent with Callaway T.S.
3/4 2-8	3.2.3.c	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 2-9	Figure 3.2-3	Make Wolf Creek specific.
3/4 2-10	4.2.3.5	Consistent with Callaway T.S. Includes Wolf Creek specific instrument.
3/4 2-11	3.2.4	Consistent with Callaway T.S.
3/4 2-15	Table 3.2-1	Consistent with Callaway T.S.
3/4 3-2	Table 3.3-1 (6b & 6c)	Consistent with Callaway T.S.
3/4 3-3	Table 3.3-1 (12a & 12b)	Consistent with Callaway T.S.
3/4 3-4	Table 3.3-1 (19)	Consistent with Callaway T.S.
3/4 3-5	Table 3.3-1, Table Notations	Consistent with Callaway T.S.
3/4 3-6	Table 3.3-1 (Action 5)	Consistent with Callaway T.S.
3/4 3-6	Table 3.3-5 (Action 5 - deletion of Boron Dilution Technical Specification)	Justification provided by KMLNRC 84-034, dated 3/16/84
3/4 3-7	Table 3.3-2 (1)	Consistent with Callaway T.S.
3/4 3-8	Table 3.3-2 (Heading & Item 12)	Consistent with Callaway T.S.
3/4 3-9	Table 4.3-1 (item 6 - deletion of note (12))	Justification provided by KMLNRC 84-034, dated 3/16/84
3/4 3-12	Table 4.3-1, Table Notations	Consistent with Callaway T.S.
3/4 3-12	Table 4.3-1, Table Notations (note (9) and note (12))	Justification provided by KMLNRC 84-034, dated 3/16/84

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 3-13	3.3.2 (Action c)	Consistent with Callaway T.S.
3/4 3-14	Table 3.3-3 (1, 1b, 1e, & 2b)	Consistent with Callaway T.S.
3/4 3-15	Table 3.3-3 (3.b.2), 3.c.3), & 3c4))	Consistent with Callaway T.S.
3/4 3-16	Table 3.3-3 (4.a.1), 4b, 4d, 4e, 5a, 5b, 5c)	Consistent with Callaway T.S.
3/4 3-17	Table 3.3-3 (Heading, 6a, 6b, 6c, 6d, 6e, & 6f)	Consistent with Callaway T.S.
3/4 3-18	Table 3.3-3 (Heading, 6g, 6h, & 7a)	Consistent with Callaway T.S.
3/4 3-19	Table 3.3-3 (9a, 9b, c, 9d, 10 & 11)	Consistent with Callaway T.S.
3/4 3-20	Table 3.3-3, Table Notations, (## note & ### note)	Consistent with Callaway T.S.
3/4 3-21	Table 3.3-3 (Action 24, 25, 26, & 27)	Consistent with Callaway T.S.
3/4 3-22	Table 3.3-4 (1, 1b, 1d, & 1e)	Consistent with Callaway T.S.
3/4 3-22	Table 3.3-4 (1.c, 1.d, & 1.e)	Wolf Creek specific values provided.
3/4 3-23	Table 3.3-4 (2b, 3a.2), & 3.b.2))	Consistent with Callaway T.S.
3/4 3-23	Table 3.3-4 (2.c & 3.b.3))	Wolf Creek specific values provided.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 3-24	Table 3.3-4 (Heading, 3.c.2), 3.c.3), 3.c.4), 4b, 4d, and 4e)	Consistent with Callaway T.S.
3/4 3-24	Table 3.3-4 (4.c, 4.d, & 4.e)	Wolf Creek specific values provided.
3/4 3-25	Table 3.3-4 (Heading, 5a, 5.b, 5.c, 6.b, 6.c, 6.d, 6.d.1 & 6.d.2)	Consistent with Callaway T.S.
3/4 3-25	Table 3.3-4 (5.b, 6.d.1), & 6.d.2))	Wolf Creek specific values provided.
3/4 3-26	Table 3.3-4 (7a)	Consistent with Callaway T.S.
3/4 3-26	Table 3.3-4 (7b)	Wolf Creek specific value provided.
3/4 3-27	Table 3.3-4 (Heading, 9b, 9c, 9d, 10 & 11)	Consistent with Callaway T.S.
3/4 3-27	Table 3.3-4 (11.a)	Wolf Creek specific value provided.
3/4 3-29	Table 3.3-5 (1m, 1o, 2.a, 2.a.2), 2.a.3), 2.a.4), 2.a.5), 2.a.6), 2.a.7), 2.a.8), 2.a.9))	Consistent with Callaway T.S.
3/4 3-30	Table 3.3-5 (3.a, 3.a.2), 3.a.3), 3.a.4), 3.a.5), 3.a.6), 3.a.7), 3.a.8), & 3.a.9), 4.a, 4.a.1), 4.a.2), 4.a.3), 4.a.4), 4.a.5), 4.a.6), 4.a.7), 4.a.8), & 4.a.9))	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 3-31	Table 3.3-5 (5.a, 7, & 8b)	Consistent with Callaway T.S.
3/4 3-32	Table 3.3-5 (14a, 14b, & 15b)	Consistent with Callaway T.S.
3/4 3-33	Table 3.3-5 ((2), (3), (4), (5), (6), (7), & (8))	Consistent with Callaway T.S.
3/4 3-34	Table 4.3-2 (1, 1b, & 2b)	Consistent with Callaway T.S.
3/4 3-35	Table 4.3-2 (Heading, 3.a.2), 3.b.2), 3.c.2), 3.c.3), & 3.c.4))	Consistent with Callaway T.S.
3/4 3-36	Table 4.3-2 (4.b, 4.e, 5.a, 5.c, 6.b, 6.c, & 6d)	Consistent with Callaway T.S.
3/4 3-37	Table 4.3-2 (Heading, & 7.a)	Consistent with Callaway T.S.
3/4 3-38	Table 4.3-2 (Heading, 9b, 9c, 9d, 10, 11, & footnotes (2) & (3))	Consistent with Callaway T.S.
3/4 3-40	Table 3.3.6 (1a, 1b, 1c, 1d, 2a, 2b, & 3)	Consistent with Callaway T.S.
3/4 3-41	Table 3.3-6 (* note, ** note, # note, ## note, ### note, Action 27, 28, & 30)	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 3-42	Table 4.3-3 (1a, 1b, 1c, 1d, 2a, 2b, &3)	Consistent with Callaway T.S.
3/4 3-46	Table 3.3-7	Consistent with Callaway T.S. and includes Wolf Creek setpoints.
3/4 3-46	Table 4.3-4 (2c, 2d, 2e, 2f, 3, * note, & ** note)	Consistent with Callaway T.S.
3/4 3-50	3.3.3.5 (LCO, Actions a, b, & c)	Consistent with Callaway T.S.
3/4 3-50	3.3.3.5 (* note)	Justification provided.
3/4 3-50	4.3.3.5.1, 4.3.3.5.2, & 4.3.3.5.3	Consistent with Callaway T.S.
3/4 3-51	Table 3.3-9 (5, 9, & 12)	Consistent with Callaway T.S.
3/4 3-52	Table 4.3-6 (12)	Consistent with Callaway T.S.
3/4 3-53	3.3.3.6 (Actions a, b, c, & d)	Consistent with Callaway T.S.
3/4 3-54	Table 3.3-10 (1, 12, 13, 14, 15, 16, 17, 18, 19, 20, & *** note)	Consistent with Callaway T.S.
3/4 3-55	Table 4.3-7 (1, 12, 13, 14, 15, 16, 17, 18, 19, 20, & *** note)	Consistent with Callaway T.S.
3/4 3-58	Table 3.3-11 (1110 & 1331)	Consistent with Callaway T.S.
3/4 3-59	Table 3.3-11 (Heading, 1509, & 1508)	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 3-60	Table 3.3-11 (Heading, 3409 & 3601)	Consistent with Callaway T.S.
3/4 3-61	Table 3.3-11 (ESW Pumphouse, & ESW Cooling Tower)	Consistent with Wolf Creek Design
3/4 3-61	Table 3.3-11 (ESF Transformers & note (1))	Consistent with Callaway T.S.
3/4 3-62	4.3.3.9.b	Consistent with Callaway T.S.
3/4 3-64	Table 3.3-12 (1a, 1b, 1c, & 1d)	Consistent with Callaway T.S.
3/4 3-64	Table 3.3-12 (2d)	Consistent with Wolf Creek design
3/4 3-65	Table 3.3-12 (Actions 32 & 33)	Consistent with Callaway T.S.
3/4 3-66	Table 4.3-8 (1a, 1b, 1c, 1d, 2a, 2b & 2c)	Consistent with Callaway T.S.
3/4 3-66	Table 4.3-8 (2d)	Consistent with Wolf Creek design
3/4 3-67	Table 4.3-8 (Notes (1), (1)a, & (1)b)	Consistent with Callaway T.S.
3/4 3-67	Table 4.3-8 (Note (2))	Consistent with Wolf Creek specific program
3/4 3-69	Table 3.3-13 (1a, 2a, 2b, 2c, 3a, 3b, 3c, 3d, & 3e)	Consistent with Callaway T.S.
3/4 3-69	Table 3.3-13 (2d)	Consistent with Wolf Creek design

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 3-70	Table 3.3-13 (4a, 4b, 4c, & 4d)	Consistent with Callaway T.S.
3/4 3-71	Table 3.3-13 (Action 39)	Consistent with Wolf Creek design
3/4 3-71	Table 3.3-13 (Actions 42 & 45)	Consistent with Callaway T.S.
3/4 3-72	Table 4.3.9 (1d, 2a, 2b, 2c, 3a, 3b, 3c, 3d, &3e)	Consistent with Callaway T.S.
3/4 3-72	Table 4.3-9 (2d)	Consistent with Wolf Creek design
3/4 3-73	Table 4.3-9 (4a, 4b, 4c, 4d, & 4e)	Consistent with Callaway T.S.
3/4 3-74	Table 4.3-9 (Notes (1), (1)a, (1)b, (6), & (7)	Consistent with Callaway T.S.
3/4 3-74	Table 4.3-9 (Note (3))	Consistent with Wolf Creek specific program
3/4 3-75	3.3.4 (* note)	Consistent with Callaway T.S.
3/4 3-75	4.3.4.2.a.3) & 4)	Consistent with Wolf Creek design
3/4 3-75	4.3.4.2 (a2), a3), b, c, d, e))	Wolf Creek specific justification provided
3/4 4-1	3.4.1.1 (Action & * note)	Consistent with Callaway T.S.
3/4 4-2	3.4.1.2 & 3.4.1.2 (Actions b & c, & ** note)	Consistent with Callaway T.S.
3/4 4-2	4.4.1.2.3	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 4-3	3.4.1.3 (* note)	Typo
3/4 4-4	4.4.1.3.2	Consistent with Callaway T.S.
3/4 4-5	3.4.1.4.1 (** note)	Typo
3/4 4-10	3.4.4 (Actions a, b, c, d, & e, & * note)	Consistent with Callaway T.S.
3/4 4-10	4.4.4.2	Consistent with Callaway T.S.
3/4 4-11	3.4.5 (Action)	Consistent with Callaway T.S.
3/4 4-13	4.4.5.3.c.3) & 4)	Consistent with Callaway T.S.
3/4 4-14	4.4.5.4.a.6) & 7)	Consistent with Callaway T.S.
3/4 4-15	4.4.5.5.c	Consistent with Callaway T.S.
3/4 4-18	4.4.6.1.6	Consistent with Callaway T.S.
3/4 4-19	3.4.6.2.f (* note)	Consistent with Callaway T.S.
3/4 4-20	4.4.6.2.1c, 4.4.5.2.2.b, & 4.4.6.2.2.d	Consistent with Callaway T.S.
3/4 4-21	Table 3.4-1	Consistent with Callaway T.S.
3/4 4-22	3.4.7 (Action a)	Consistent with Callaway T.S.
3.4 4-25	3.4.8 (Actions b and d)	Consistent with Callaway T.S.
3/4 4-26	3.4.8 (Action for Modes 1-5, Actions 2 & 5)	Consistent with Callaway T.S.
3/4 4-28	Table 4.4-4 (2 & 3)	Consistent with Callaway T.S.
3/4 4-28	Table 4.4-4 (4.a))	Typo

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 4-29	4.4.9.1.2	Consistent with Callaway T.S.
3/4 4-30	Figure 3.4-2	Consistent with Callaway T.S.
3/4 4-31	Figure 3.4-3	Consistent with Callaway T.S. with Wolf Creek specific notch in curve.
3/4 4-32	Table 4.4-5	Consistent with Wolf Creek analysis
3/4 4-34	3.4.9.3	Consistent with Callaway T.S. and justification provided
3/4 4-34	3.4.9.3 (a, b, & c and Actions a, b, & c)	Consistent with Callaway T.S. and justification provided.
3/4 4-35	4.4.9.3.2 & 4.4.9.3.3	Consistent with Callaway T.S.
3/4 4-36	Figure 3.4-4	Consistent with Callaway T.S.
3/4 5-1	3.5.1.a and Action b	Consistent with Callaway T.S.
3/4 5-2	4.5.1.1 (b,c,&d)	Consistent with Callaway T.S.
3/4 5-3	3.5.2 (* note)	Consistent with Callaway T.S.
3/4 5-5	4.5.2.e.1)	Consistent with Callaway T.S.
3/4 5-6	4.5.2.h.3) & 4.5.2.i	Consistent with Callaway T.S.
3/4 5-7	3/4.5.3 (Heading)	Consistent with Callaway T.S.
3/4 5-7	3.5.3.d and 3.5.3 (Action a & c)	Consistent with Callaway T.S.
3/4 5-8	4.5.3.2	Consistent with Callaway T.S.
3/4 5-9	3.5.4 (Action & * note)	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 5-10	3/4.5.5	Consistent with Callaway T.S.
3/4 5-11	3/4.5.6 (Heading)	Consistent with Callaway T.S.
3/4 5-11	3.5.6 (c & d)	Consistent with Callaway T.S.
3/4 6-1	4.6.1.1.a	Consistent with Callaway T.S.
3/4 6-2	3.6.1.2.a.2)	Consistent with 10CFR, Appendix J.
3/4 6-2	3.6.1.2.b	Consistent with Callaway T.S.
3/4 6-2	4.6.1.2.a	Typo
3/4 6-3	4.6.1.2(c.1), c.3), f & g)	Consistent with Callaway T.S.
3/4 6-3	4.6.1.2.d	Wolf Creek specific justification provided
3/4 6-5	4.6.1.3.a	Consistent with Callaway T.S.
3/4 6-6	3.6.1.4	Consistent with Callaway T.S.
3/4 6-8 thru 3/4 6-10	3.6.1.6 thru 4.6.1.6.2 & 4.6.1.6.3	Revise to be consistent with Callaway T.S.
3/4 6-11	3.6.1.7 (a & b and Actions a, b, & c)	Consistent with Callaway T.S.
3/4 6-12	4.6.1.7 (1, 2, 3, 4, & * note)	Consistent with Callaway T.S.
3/4 6-13	3.6.2.1 and Action	Consistent with Callaway T.S.
3/4 6-13	4.6.2.1.c (1) & 2))	Consistent with Table 3.3-3
3/4 6-14	4.6.2.1.(b, c.1) & e.2))	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 6-14	4.6.2.2.c	Consistent with Callaway T.S. and Table 3.3-3
3/4 6-15	3.6.2.3 (Action c)	Consistent with Callaway T.S.
3/4 6-15	4.6.2.3. (a.2) & b)	Consistent with Callaway T.S.
3/4 6-16	3.6.3 (Action, Actions b & c)	Consistent with Callaway T.S.
3/4 6-16	4.6.3.1	Consistent with Callaway T.S.
3/4 6-17	4.6.3.2	Consistent with Callaway T.S.
3/4 6-18	Table 3.6-1 (** note)	Consistent with Callaway T.S.
3/4 6-18	Table 3.6-1 (P-25)	Typo
3/4 6-19	Table 3.6-1 (** note)	Consistent with Callaway T.S.
3/4 6-20	Table 3.6-1 (P-101 & P30)	Typo
3/4 6-21	Table 3.6-1 (** note, first P-69, & P-58)	Consistent with Callaway T.S.
3/4 6-22	Table 3.6-1 (** note, *** note, P-16, P-13, P-45, P-65 (both), & P-67)	Consistent with Callaway T.S.
3/4 6-22	Table 3.6-1 (P-45)	Typo
3/4 6-23	Table 3.6-1 (*** note & 5)	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 6-23	Table 3.6-1 (P-40)	Typo
3/4 6-24	Table 3.6.1 (** note, P-15 & P-14)	Consistent with Callaway T.S.
3/4 6-25	Table 3.6-1 (P-15)	Typo
3/4 6-26	Table 3.6-1 (P-14 & P-89)	Typo
3/4 6-28	Table 3.6-1 (P-88)	Typo
3/4 6-29	Table 3.6-1 (P-98 & 9)	Consistent with Callaway T.S.
3/4 6-30	3.6.4.1 (Actions a & b)	Consistent with Callaway T.S.
3/4 6-30	4.6.4.1	Consistent with Callaway T.S.
3/4 6-31	3.6.4.2 & Action	Consistent with Callaway T.S.
3/4 6-31	4.6.4.2 (1.b.2) & 2)	Consistent with Callaway T.S.
3/4 7-5	4.7.1.2.1.b.3)	Consistent with Callaway T.S.
3/4 7-6	3.7.1.3 & Action b	Consistent with Callaway T.S.
3/4 7-11	4.7.3 (b.1) & b.2))	Consistent with Callaway T.S.
3/4 7-13	3.7.5.b	Consistent with Wolf Creek analysis
3/4 7-15	4.7.6 (c.1), c.2), d, & e.2))	Consistent with Callaway T.S.

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Page	Section	Reason for Change
3/4 7-16	4.7.6.e.5)	Consistent with Wolf Creek design & Typo
3/4 7-16	4.7.6.(f & g)	Consistent with Callaway T.S.
3/4 7-17	3.7.7 (Action)	Consistent with Callaway T.S.
3/4 7-17	4.7.7 (b.1) & b.2))	Consistent with Callaway T.S.
3/4 7-18	4.7.7 (c, d2), e, & f)	Consistent with Callaway T.S.
3/4 7-19	3.7.8	Consistent with Callaway T.S.
3/4 7-19	4.7.8	Consistent with Callaway T.S.
3/4 7-19	4.7.8 (* note)	Typo
3/4 7-20	4.7.8.c	Consistent with Callaway T.S.
3/4 7-20	4.7.8.e.2)	Typo
3/4 7-21	4.7.8.(e.2) & e.3))	Consistent with Callaway T.S. & Typo
3/4 7-21	4.7.8.e	Typo
3/4 7-22	4.7.8 (f.2) & g)	Consistent with Callaway T.S.
3/4 7-27	3.7.10.1.a	Wolf Creek specific justification provided
3/4 7-27	3.7.10.1 (Action a)	Wolf Creek specific justification provided
3/4 7-28	4.7.10.1.1.f.1)	Consistent with Callaway T.S.
3/4 7-28	4.7.10.1.1.f.2)	Wolf Creek specific justification provided
3/4 7-28	4.7.10.1.2.a	Wolf Creek specific justification provided

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 7-30	3.7.10.2 (a, c, & * note)	Consistent with Callaway T.S.
3/4 7-31	4.7.10.2.d	Consistent with Callaway T.S.
3/4 7-32	3.7.10.3 (* note)	Consistent with Callaway T.S.
3/4 7-32	4.7.10.3 (a, b, c.1) & c.2))	Consistent with Callaway T.S.
3/4 7-34	Table 3.7-5 (Turbine Bldg.)	Consistent with Callaway T.S.
3/4 7-36	3.7.11 & 3.7.11 (Action a)	Consistent with Callaway T.S.
3/4 7-36	4.7.11.1, 4.7.11.1.b, & 4.7.11.1.c	Consistent with Callaway T.S.
3/4 7-36	4.7.11.1.c	Typo
3/4 7-36	4.7.11.2.b	Consistent with Callaway T.S.
3/4 7-37	3.7.12	Consistent with Callaway T.S.
3/4 8-1	3.8.1.1 (b.1) & Actions a & b)	Consistent with Callaway T.S.
3/4 8-2	3.8.1.1 (Action e)	Consistent with Callaway T.S.
3/4 8-2	4.8.1.1.1 (a & b)	Consistent with Callaway T.S.
3/4 8-3	4.8.1.1.2 (a.4), d, e, & * note)	Consistent with Callaway T.S.
3/4 8-4	4.8.1.1.2 (f. , f.2), f.4)b), f.5))	Consistent with Callaway T.S.

TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
3/4 8-5	4.8.1.1.2 (f.5), f.6), f.7), & * note)	Consistent with Callaway T.S.
3/4 8-6	4.8.1.1.2 (g, h.1))	Consistent with Callaway T.S.
3/4 8-6	4.8.1.1.3	Consistent with Callaway T.S.
3/4 8-7	Table 4.8-1 (* note)	Consistent with Callaway T.S.
3/4 8-8	3.8.1.2 (Action)	Consistent with Callaway T.S.
3/4 8-9	3.8.2.1 (Actions a & b)	Consistent with Callaway T.S.
3/4 8-9	4.8.2.1.a.2	Consistent with Callaway T.S.
3/4 8-10	4.8.2.1. (b.2), c.3), c.4), d, & e)	Consistent with Callaway T.S.
3/4 8-11	Table 4.8-2 (Float voltage, Specific gravity)	Consistent with Callaway T.S.
3/4 8-12	3.8.2.2 (Actions a & b)	Consistent with Callaway T.S.
3/4 8-13	3.8.3.1 (a through j, Action c)	Consistent with Callaway T.S.
3/4 8-15	3.8.3.2	Consistent with Callaway T.S.
3/4 8-15	3.8.3.2 (a & b, and Action)	Consistent with Callaway T.S.
3/4 8-16	4.8.4.1.a.1	Consistent with Callaway T.S.
3/4 8-17	4.8.4.1.a.2	Consistent with Callaway T.S.
3/4 8-18 thru 3/4 8-24	Table 3.8-1	Consistent with Callaway T.S.

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Page	Section	Reason for Change
3/4 8-25	3.8.4.2	Consistent with Callaway T.S.
3/4 8-26	Table 3.8-2	Consistent with Callaway T.S.
3/4 9-1	3/4.9 (Heading)	Consistent with Callaway T.S.
3/4 9-1	4.9.1.3	Justification provided by KMLNRC 84-034 dated 3/16/84
3/4 9-6	3.9.6 (a.2)a), a.2)b), & a.3)	Consistent with Callaway T.S.
3/4 9-6	3.9.6.b.2)	Justification provided
3/4 9-7	4.9.6.1 & 4.9.6.2	Consistent with Callaway T.S.
3/4 9-10	3.9.8.2 (* note)	Consistent with Callaway T.S.
3/4 9-14	3.9.11 & 4.9.11	Consistent with Callaway T.S.
3/4 9-14	3.9.11 (Applicability) & 4.9.11	Justification provided
3/4 9-15	4.9.12	Consistent with Callaway T.S.
3/4 9-17	3.9.13	Consistent with Callaway T.S.
3/4 9-17	3.9.13	Justification provided
3/4 9-17	4.9.13.b.1)	Consistent with Callaway T.S.
3/4 9-18	4.9.13.b.2)	Typo
3/4 10-1	3.10.1	Consistent with Callaway T.S.
3/4 10-2	3.10.2 (Action)	Consistent with Callaway T.S.
3/4 10-4	3.10.4 (a, b, and Actions a & b)	Consistent with Callaway T.S.
3/4 10-4	4.10.4.3	Consistent with Callaway T.S.
3/4 11-2	Table 4.11-1 (1.b & 2)	Consistent with Callaway T.S.

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Page	Section	Reason for Change
3/4 11-3	Table 4.11-1, Table Notations (Note (2))	Per discussions with NRC reviewer (Bob Fell)
3/4 11-4	Table 4.11-1, Table Notations (Notes (4) & (6))	Consistent with Callaway T.S.
3/4 11-6	4.11.1.3.1	Consistent with Callaway T.S.
3/4 11-7	4.11.1.4	Consistent with Callaway T.S.
3/4 11-9	Table 4.11-2 (Items 1, 2 & 3)	Consistent with Callaway T.S.
3/4 11-9	Table 4.11-2 (Items 3b & 4)	Wolf Creek specific justification provided
3/4 11-9	Table 4.11-2 (Item 5)	Consistent with Callaway T.S.
3/4 11-10	Table 4.11-2 Table Notations (Note (1))	Consistent with Callaway T.S.
3/4 11-11	Table 4.11-2, Table Notations (Notes (4), (5), (7))	Consistent with Callaway T.S.
3/4 11-12	4.11.2.2	Typo
3/4 11-14	3.11.2.4	Typo
3/4 11-14	3.11.2.4 (Action a.2)	Typo
3/4 11-15	3.11.2.5	Consistent with Callaway T.S.
3/4 11-15	3.11.2.5 (Action a)	Consistent with Callaway T.S.
3/4 11-16	3.11.2.6	Consistent with Callaway T.S.
3/4 11-16	4.11.2.6	Wolf Creek specific justification provided

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Page	Section	Reason for Change
3/4 11-18	3.11.4 (Action a)	Typo
3/4 12-1	3.12.1 (Action c)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-2	3.12.1 (Action c cont'd)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-3	Table 3.12-1 (Item 1)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-4	Table 3.12-1 (Items 2 & 3)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-5	Table 3.12-1 (Items 4a & 4b)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-6	Table 3.12-1 (Item 4c)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-7	Table 3.12-1 Table Notations (Notes (1) & (2))	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-8	Table 3.12-1 Table Notations (Notes (6) & (7))	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-9	Table 3.12-2	Consistent with Callaway T.S.
3/4 12-10	Table 4.12-1	Consistent with Callaway T.S.
3/4 12-13	3.12.2 (* note)	Per discussions with NRC reviewer (Mike Wangler)
3/4 12-14	3.12.3	Per discussions with NRC reviewer (Mike Wangler)
B 3/4 0-1	3/4.0	Consistent with Callaway T.S.
B 3/4 0-2	3/4.0	Consistent with Callaway T.S.
B 3/4 0-3	3/4.0	Consistent with Callaway T.S.

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Page	Section	Reason for Change
B 3/4 1-2	3/4.1.2	Consistent with Callaway T.S.
B 3/4 1-3	3/4.1.2	Consistent with Callaway T.S.
B 3/4 2-4	3/4.2.2 & 3/4.2.3	Consistent with Callaway T.S.
B 3/4 2-5	3/4.2.2 & 3/4.2.3	Consistent with Callaway T.S.
B 3/4 2-5	3/4.2.4	Consistent with Callaway T.S.
B 3/4 2-6	3/4.2.5	Consistent with Callaway T.S.
B 3/4 3-2	3/4.3.1 & 3/4.3.2	Consistent with Callaway T.S.
B 3/4 3-3	3/4.3.3.1	Consistent with Callaway T.S.
B 3/4 3-3	3/4.3.3.2	Consistent with Callaway T.S.
B 3/4 3-4	3/4.3.3.5	Consistent with Callaway T.S.
B 3/4 3-4	3/4.3.3.6	Consistent with Callaway T.S.
B 3/4 3-5	3/4.3.3.8	Consistent with Callaway T.S.
B 3/4 3-5	3/4.3.3.9	Consistent with Callaway T.S.
B 3/4 3-5	3/4.3.3.10	Consistent with Callaway T.S.
B 3/4 3-6	3/4.3.4	Consistent with Callaway T.S.
B 3/4 4-1	3/4.4.1	Consistent with Callaway T.S.
B 3/4 4-2	3/4.4.2	Consistent with Callaway T.S.
B 3/4 4-2	3/4.4.3	Consistent with Callaway T.S.
B 3/4 4-3	3/4.4.5	Consistent with Callaway T.S.
B 3/4 4-4	3/4.4.5	Consistent with Callaway T.S.
B 3/4 4-4	3/4.4.6.2	Consistent with Callaway T.S.
B 3/4 4-7	3/4.4.9	Consistent with Callaway T.S.
B 3/4 4-8	3/4.4.9	Consistent with Callaway T.S.

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Page	Section	Reason for Change
B 3/4 4-9	3/4.4.9	Consistent with Callaway T.S.
B 3/4 4-14	3/4.4.9	Consistent with Callaway T.S.
B 3/4 5-2	3/4.5.2, 3/4.5.3, & 3/4.5.4	Consistent with Callaway T.S.
B 3/4 5-2	3/4.5.5 & 3/4.5.6	Consistent with Callaway T.S.
B 3/4 6-1	3/4.6.1.2	Consistent with 10CFR50, Appendix J
B 3/4 6-2	3/4.6.1.4	Consistent with Callaway T.S.
B 3/4 6-2	3/4.6.1.6	Consistent with Callaway T.S.
B 3/4 6-3	3/4.6.1.7	Consistent with Callaway T.S.
B 3/4 6-4	3/4.6.4	Consistent with Callaway T.S.
B 3/4 7-2	3/4.7.1.2	Consistent with Callaway T.S.
B 3/4 7-4	3/4.7.5	Wolf Creek specific justification provided
B 3/4 7-4	3/4.7.6	Consistent with Callaway T.S.
B 3/4 7-4	3/4.7.7	Consistent with Callaway T.S.
B 3/4 7-5	3/4.7.8	Consistent with Callaway T.S.
B 3/4 7-6	3/4.7.8	Wolf Creek specific justification provided
B 3/4 7-7	3/4.7.10	Consistent with Callaway T.S.
B 3/4 8-1	3/4.8.1, 3/4.8.2, & 3/4.8.3)	Consistent with Callaway T.S.
B 3/4 8-2	3/4.8.1, 3/4.8.2, & 3/4.8.3	Consistent with Callaway T.S.
B 3/4 8-3	3/4.8.4	Consistent with Callaway T.S.
B 3/4 9-1	3/4.9.1	Justification provided by KMLNRC 84-034 dated 3/16/84

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Page	Section	Reason for Change
B 3/4 11-1	3/4.11.1.1	Consistent with Callaway T.S.
B 3/4 11-1	3/4.11.1.2	Grammatical Correction
B 3/4 11-2	3/4.11.1.2	Consistent with Callaway T.S.
B 3/4 11-2	3/4.11.1.3	Consistent with Callaway T.S.
B 3/4 11-3	3/4.11.1.3	Consistent with Callaway T.S.
B 3/4 11-3	3/4.11.2.1	Consistent with Callaway T.S.
B 3/4 11-4	3/4.11.2.1	Consistent with Callaway T.S.
B 3/4 11-4	3/4.11.2.2	Consistent with Callaway T.S. and per discussions with NRC reviewer (Bob Fell)
B 3/4 11-5	3/4.11.2.3	Consistent with Callaway T.S. and per discussions with NRC reviewer (Bob Fell)
B 3/4 11-6	3/4.11.2.4	Consistent with Callaway T.S.
B 3/4 11-6	3/4.11.2.5	Consistent with Callaway T.S.
B 3/4 11-6	3/4.11.2.6	Consistent with Callaway T.S.
B 3/4 12-1	3/4.12.1	Per discussions with NRC reviewer (Mike Wangler)
B 3/4 12-1	3/4.12.2	Per discussions with NRC reviewer (Mike Wangler)
B 3/4 12-2	3/4.12.2	Typo
5-1	5.1.3	Consistent with Callaway T.S.
5-3	Figure 5.1-2	Wolf Creek specific justification provided
5-4	Figure 5.1-3	Wolf Creek specific justification provided
5-5	Figure 5.1-4	Wolf Creek specific justification provided

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Page	Section	Reason for Change
5-6	5.3.1	Consistent with Callaway T.S.
6-1	6.2.2.e	Wolf Creek specific justification provided
6-4	Figure 6.2-? (** note)	Wolf Creek specific justification
6-6	6.2.3.1	Consistent with Callaway T.S. and KG&E organization
6-6	6.2.3.4	Consistent with KG&E organization
6-9	6.5.1.8	Consistent with Callaway T.S. and KG&E organization
6-10	6.5.2.2	Consistent with KG&E organization
6-13	6.8.1	Consistent with Callaway T.S.
6-14	6.8.1 (* note)	Consistent with KG&E organization and policies
6-14	6.8.1.g	Consistent with Callaway T.S.
6-14	6.8.2.c	Consistent with KG&E organization and policies
6-15	6.8.3.c	Consistent with KG&E organization and policies
6-15	6.8.4.a	Consistent with Callaway T.S.
6-18	6.9.1.b	Consistent with Callaway T.S.
6-19	6.9.1.7 & 6.9.1.7 (* note)	Per discussion with NRC reviewer (Mike Wangler)
6-24	6.15.1.a	Consistent with KG&E organization

Note -' The organizations depicted in the Wolf Creek "Proof and Review" copy of Technical Specificaions in Figure 6.2-1 (pg. 6-3) and 6.22 (pg. 6-4) are as they existed for Wolf Creek Operations in 12/83. However, the most recent organizations are shown in the Wolf Creek FSAR site addendum. In order to reduce the number of changes needed to be made to the Technical Specification figures between now and fuel load, these figures will be updated just prior to fuel load.

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TECHNICAL SPECIFICATIONS

FOR

WOLF CREEK - UNIT 1

DOCKET NO. 50-482

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SECTION PAGE

1.0 DEFINITIONS

SECTION PAGE

DEFINITIONS (Continued)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SECTION PAGE

INSTRUMENTATION (Continued)

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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REACTOR COOLANT SYSTEM (Continued)

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds with the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

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

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CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.

b. All equipment hatches are closed and sealed,

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

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

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

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow ~~supplied to~~ from the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

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ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

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

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.15 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.22 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

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

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QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

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

SITE BOUNDARY

1.29 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.30 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.31 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

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SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

1.34 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.35 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.36 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

WASTE GAS HOLDUP SYSTEM

1.40 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off-gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

TABLE 1.1

FREQUENCY NOTATION

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

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

---

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

SECTION 2.0

SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three loop operation, respectively.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within ~~6~~ 1 hour~~s~~, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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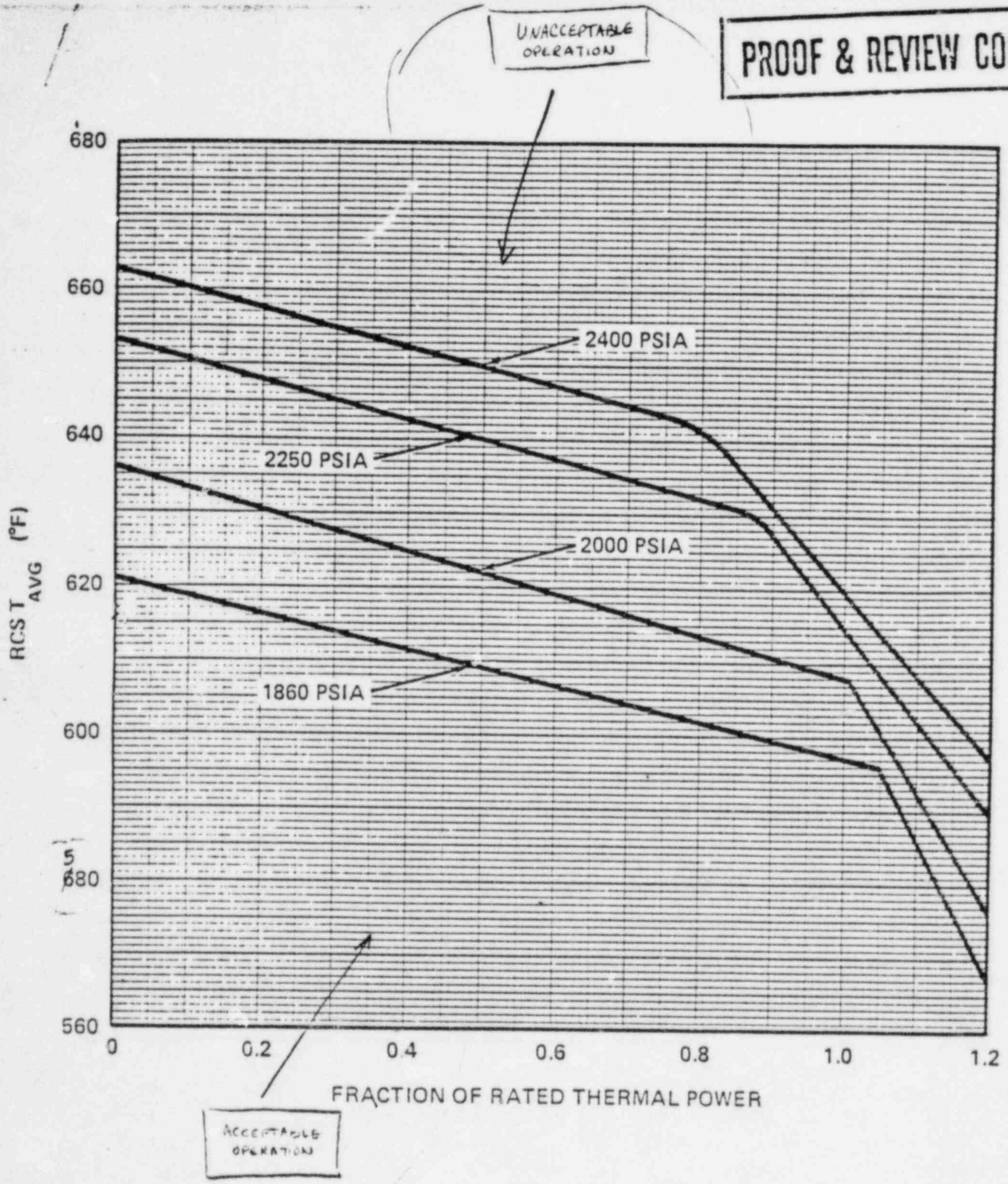

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

Figure 2.1-2 left blank pending NRC approval of three-loop operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlocks Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowances) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤~~111.2~~ 112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤~~27.2~~ 28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	~~2.0~~ 2.4	0.5	0	≤~~8~~ 4% of RTP* with a time constant ≥2 seconds	≤~~6.8~~ 6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	~~2.0~~ 2.4	0.5	0	≤~~8~~ 4% of RTP* with a time constant ≥2 seconds	≤~~6.8~~ 6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤~~31~~ 35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤~~1.4~~ 1.6 x 10^5 cps
7. Overtemperature ΔT	~~6.7~~ 6.9	~~2.79~~ 2.83	~~0.8~~ 2.26	See Note 1	See Note 2
8. Overpower ΔT	~~4.3~~ 5.5	~~1.3~~ 1.43	~~0.2~~ 1.35	See Note 3	See Note 4
9. Pressurizer Pressure-Low	~~6.8~~ 3.7	0.71	~~1.5~~ 2.49	≥~~1900~~ 1875 psig	≥~~1886~~ 1866 psig
10. Pressurizer Pressure-High	~~3.1~~ 7.5	0.71	~~1.5~~ 2.49	≤2385 psig	≤~~2396~~ 2400 psig
11. Pressurizer Water Level-High	~~5.0~~ 8.0	2.18	~~1.5~~ 1.96	≤92% of instrument span	≤~~93.8~~ 93.4% of instrument span

---

*RTP = RATED THERMAL POWER

**Loop design flow = 95,700 gpm

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	2.5	1.77 ~~1.0~~	0.60 ~~1.5~~	≥90% of loop design flow**	≥89.2% of loop design flow**
13. Steam Generator Water Level Low-Low	23.5 ~~30.0~~	21.18 ~~27.18~~	2.51 ~~1.5~~	23.5 ≥~~32.3~~ of narrow range instrument span	22.3 ≥~~30.4~~ of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	7.5 ~~5.0~~	1.30	0.0	10578 volts A.C. ≥~~70% of bus voltage~~	10355 volts A.C.
15. Underfrequency - Reactor Coolant Pumps	3.3 ~~1.2~~	0.0	0.0	57.2 ≥~~57.5~~ Hz	57.1 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥ 584.62 psig	≥ 534.75 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥ 1% open	≥ 1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥1 x 10^{-10} amps	≥6 x 10^{-11} amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	~~≥~~10% of RTP*	≥ 6.7% to ≤ 13.3% ~~≥12.2%~~ of RTP*
2) P-13 input	N.A.	N.A.	N.A.	≤10% of RTP* turbine impulse pressure equivalent	≤~~12.2~~ 12.4% of RTP* turbine impulse pressure equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤48% of RTP*	≤~~50.2~~ 51.3% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤ ~~≥~~50% of RTP*	≤ 53.3 ~~≥52.2%~~ of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A	~~≥~~10% of RTP*	≥ 6.7 to ≤ 13.3% ~~≥7.8%~~ of RTP*
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A	≤10% of RTP* turbine impulse pressure equivalent	≤~~12.2~~ 12.4% of RTP* turbine impulse pressure equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

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*RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} [T (\frac{1}{1 + \tau_6 S}) - T'] + K_3(P - P') - f_1(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT,

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s,

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT,

τ_3 = Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s,

ΔT_o = Indicated ΔT at RATED THERMAL POWER,

K_1 = 1.10,

K_2 = 0.0137 ~~0.0138~~/°F,

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation,

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 28 s, τ_5 = 4 s,

T = Average temperature, °F,

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg},

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T' ≤ 588.5°F (Nominal T_{avg} at RATED THERMAL POWER),

K_3 = $0.\overset{0}{\wedge}00671$,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -35% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35%, the ΔT Trip Setpoint shall be automatically reduced by 1.26% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 1.05% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than (~~3.8~~)% of ΔI span.
3.6

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2(\Delta I)\right\}$$

Where:

ΔT = Measured ΔT by RTD manifold instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_4 = ~~1.09~~ 1.08;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constant utilized in the rate-lag compensator for T_{avg}, $\tau_7 = 10$ s;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6 = 0.00128/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;

T = Average temperature, °F;

T'' = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.5°F);

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than (~~3.0~~)% of ΔI span.
3.5

BASES
FOR
SECTION 2.0
SAFETY LIMITS
AND
LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the W-3 correlation (R-GRID). The W-3 DNB correlation (R-GRID) has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures 2.1-1 and 2.1-2 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.2\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

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2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides (RCS) contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping and valves are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at greater than or equal to ~~3107 psig,~~ 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

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2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE OCCURRENCES.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore, providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a high and low range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the limit value.

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Intermediate and Source Range, ~~Nuclear~~ Neutron Flux

lower case

The Intermediate and Source Range, ~~Nuclear~~ Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection tc the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT Reactor trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam ~~Break~~ Releases."

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Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

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Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.3 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range ~~Reactor~~ trip (i.e., prevents premature block of Source Range trip), provides a backup block for Source Range Neutron Flux doubling, and allows deenergization of the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the single loop Low Flow trip.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range ~~Reactor~~ trip and the Low Setpoint Power Range ~~Reactor~~ trip; and automatically blocks the Source Range ~~Reactor~~ trip and de-energizes the Source Range high voltage power. On decreasing power, the Intermediate Range ~~Reactor~~ trip and the Low Setpoint Power Range ~~Reactor~~ trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

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SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,
b. At least HOT SHUTDOWN within the following 6 hours, and
c. At least COLD SHUTDOWN within the subsequent 24 hours.

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

This Specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

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SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, but

b. The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME BOILER AND PRESSURE VESSEL CODE AND APPLICABLE ADDENDA TERMINOLOGY FOR INSERVICE INSPECTION AND TESTING ACTIVITIES	REQUIRED FREQUENCIES FOR PERFORMING INSERVICE INSPECTION AND TESTING ACTIVITIES
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities;

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements; and

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} >200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% Δk/k for four loop operation.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.3% Δk/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

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

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

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*See Special Test Exception Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e. above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% $\Delta k/k$:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

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

    1) Reactor Coolant System boron concentration,

    2) Control rod position,

    3) Reactor Coolant System average temperature,

    4) Fuel burnup based on gross thermal energy generation,

    5) Xenon concentration, and

    6) Samarium concentration.

MODERATOR TEMPERATURE COEFFICIENT

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LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 $\Delta k/k/^\circ F$ for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition, ~~or~~ and

b. Less negative than -4.1×10^{-4} $\Delta k/k/^\circ F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2#* .
Specification 3.1.1.3b. - MODES 1, 2, and 3#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^\circ F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

---

*With K_{eff} greater than or equal to 1.

#See Special Test Exception Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading, and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} $\Delta k/k/^{\circ}F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} $\Delta k/k/^{\circ}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

MINIMUM TEMPERATURE FOR CRITICALITY

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LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

#With K_{eff} greater than or equal to 1.

*See Special Test Exception Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

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FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System if the Boric Acid Storage System ~~in Specification 3.1.2.5a.~~ is OPERABLE~~, or~~ as given in Specification 3.1.2.5a. for MODES 5 and 6 or as given in Specification 3.1.2.6a. for MODE 4 ; or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank ~~in Specification 3.1.2.5b~~. is OPERABLE. as given in Specification 3.1.2.5b. for MODES 5 and 6 or as given in Specification 3.1.2.6b. for MODE 4.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the Reactor Coolant System.

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* See attached note

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours prior to the temperature of one or more of the RCS cold legs exceeding 375°F.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One centrifugal charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With no centrifugal charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required centrifugal charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

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*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

CHARGING PUMPS - OPERATING

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LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two centrifugal charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one centrifugal charging pump OPERABLE, restore at least two centrifugal charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in ~~COLD~~ HOT SHUTDOWN within the next ~~30~~ 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two centrifugal charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Specification 4.0.5.

* add note on next page

upper case

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into Mode 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that centrifugal charging pump is restored to operable status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°.

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BORATED WATER SOURCE - SHUTDOWN

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LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

- a. A Boric Acid Storage System with:
  - 1) A minimum contained borated water volume of ~~2713~~ 2968 gallons,
  - 2) Between 7000 and 7700 ppm of boron, and
  - 3) A minimum solution temperature of 65°F.
- b. The refueling water storage tank (RWST) with:
  - 1) A minimum contained borated water volume of ~~53,500~~ 55,416 gallons,
  - 2) A minimum boron concentration of 2000 ppm, and
  - 3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
  - 1) Verifying the boron concentration of the water,
  - 2) Verifying the contained borated water volume, and
  - 3) Verifying the boric acid storage ~~tank~~ system solution temperature when it is the source of borated water.
- b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

BORATED WATER SOURCES - OPERATING

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LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2: for MODES 1, 2 and 3 and one of the following borated water sources shall be OPERABLE as required by Specification 3.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

  1) A minimum contained borated water volume of ~~16,142~~ 17,658 gallons,

  2) Between 7000 and 7700 ppm of boron, and

  3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

  1) A minimum contained borated water volume of ~~113,476~~ 394,000 gallons

  2) Between 2000 and 2100 ppm of boron,

  3) A minimum solution temperature of 37°F, and

  4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable in MODE 1, 2, or 3 and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable in MODE 1, 2, or 3, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

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4.1.2.6 Each required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the boron concentration in the water,

   2) Verifying the contained borated water volume of the water source, and

   3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full-length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

---

*See Special Test Exceptions 3.10.2 and 3.10.3.

Specifications

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

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LIMITING CONDITION FOR OPERATION

3.1.3.2 The ~~Shutdown and Control~~ Rod Digital Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

POSITION INDICATION SYSTEM-SHUTDOWN

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LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*#, and 5*#.

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicator agrees with the demand position indicator within 12 steps when exercised over the full range of rod travel at least once per 18 months.

---

*With the Reactor Trip System breakers in the closed position.

#See Special Test Exception 3.10.5.

Specification

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

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LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

CONTROL ROD INSERTION LIMITS

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LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and 3.1-2.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

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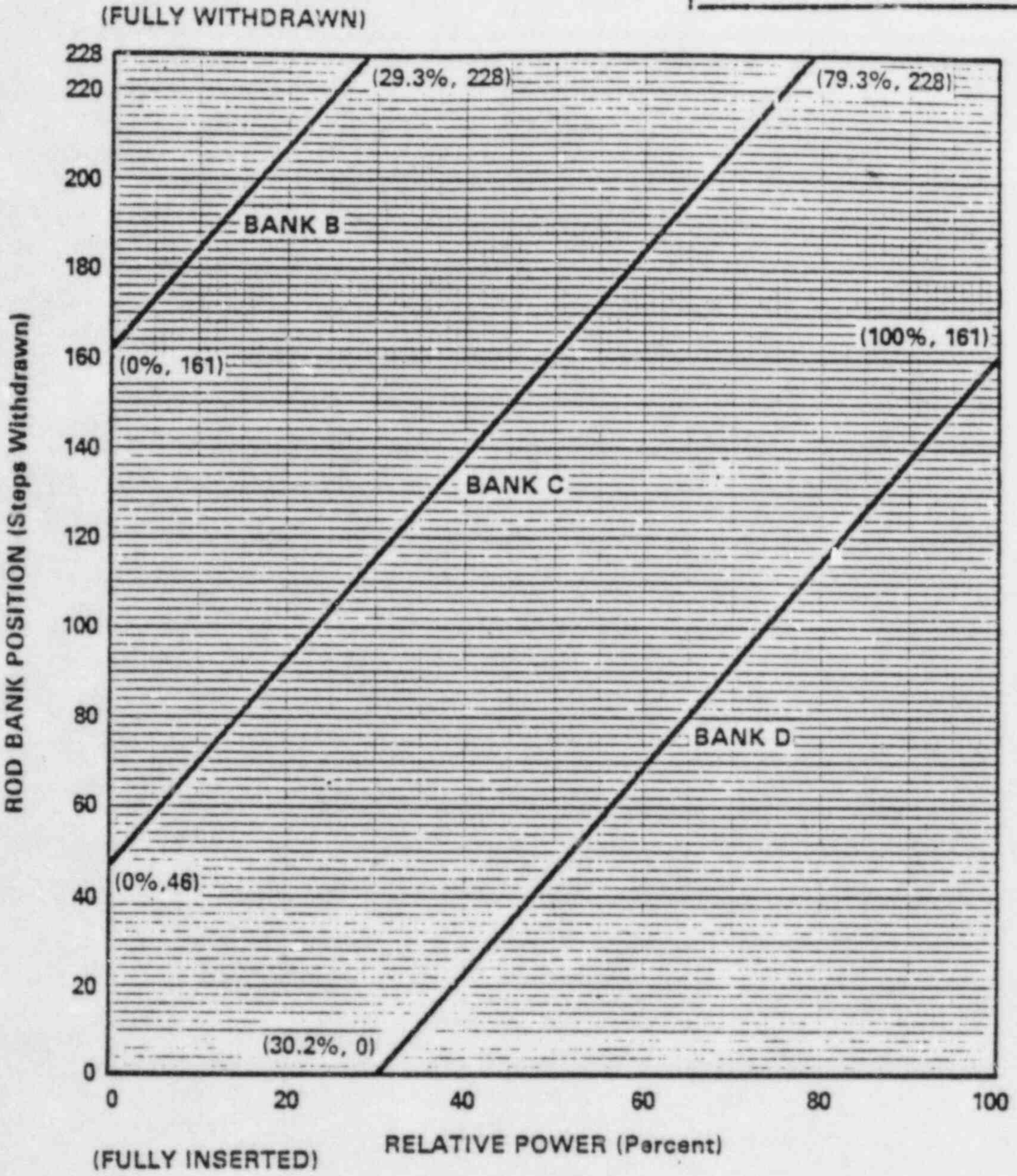

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS
THERMAL POWER-FOUR LOOP OPERATION

Figure 3.1-2 left blank pending NRC approval of three-loop operation

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU, and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

    1. Restore the indicated AFD to within the above required target band limits, or

    2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:
    1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and
    2. The Power Range Neutron Flux# - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

Specification

---

*See Special Test Exception 3.10.2.

#Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

ACTION (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulated penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

    1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

    2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

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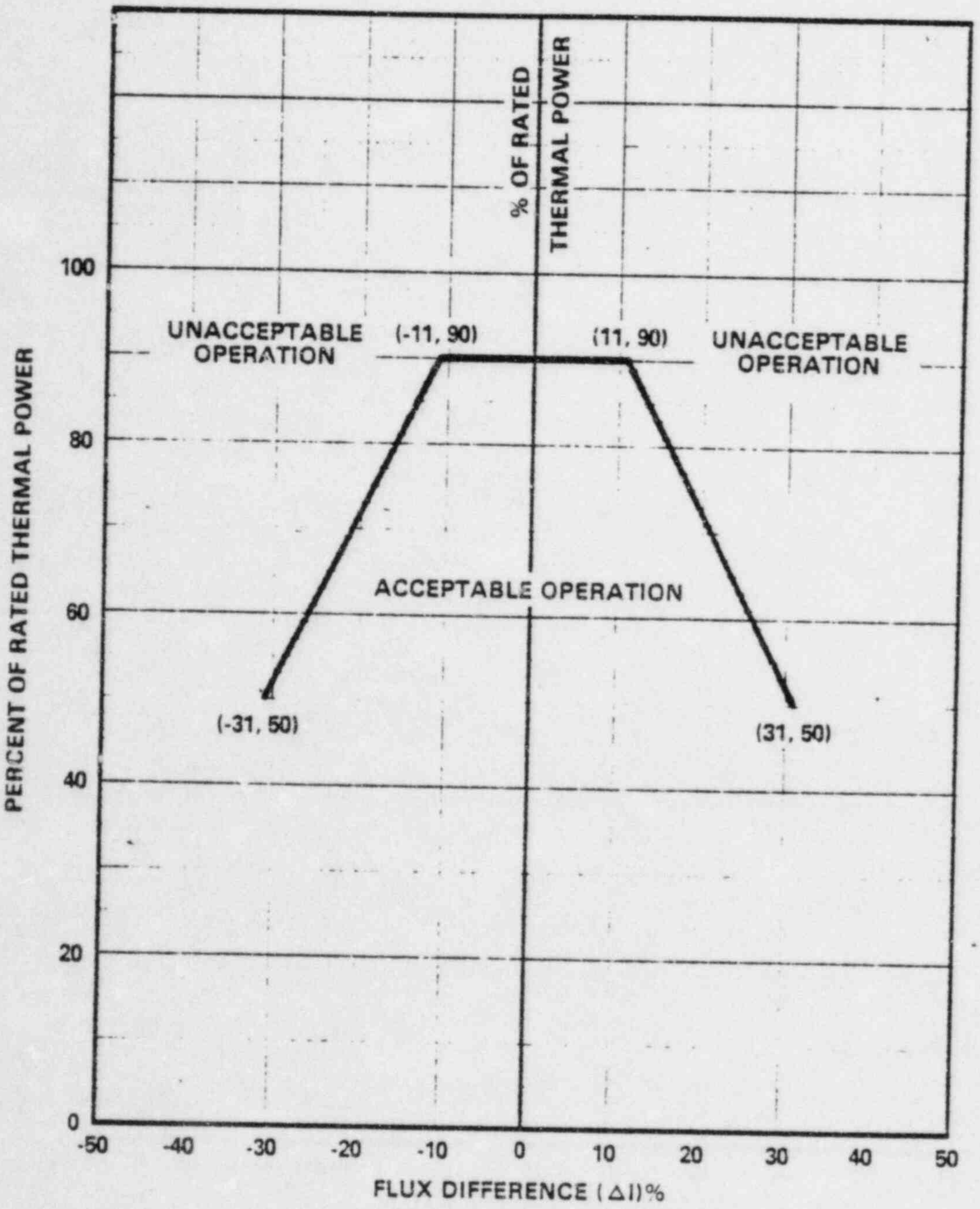

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A
FUNCTION OF RATED THERMAL POWER

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3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5 \text{, and}$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5.$$

Where:

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}} \text{, and}$$

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

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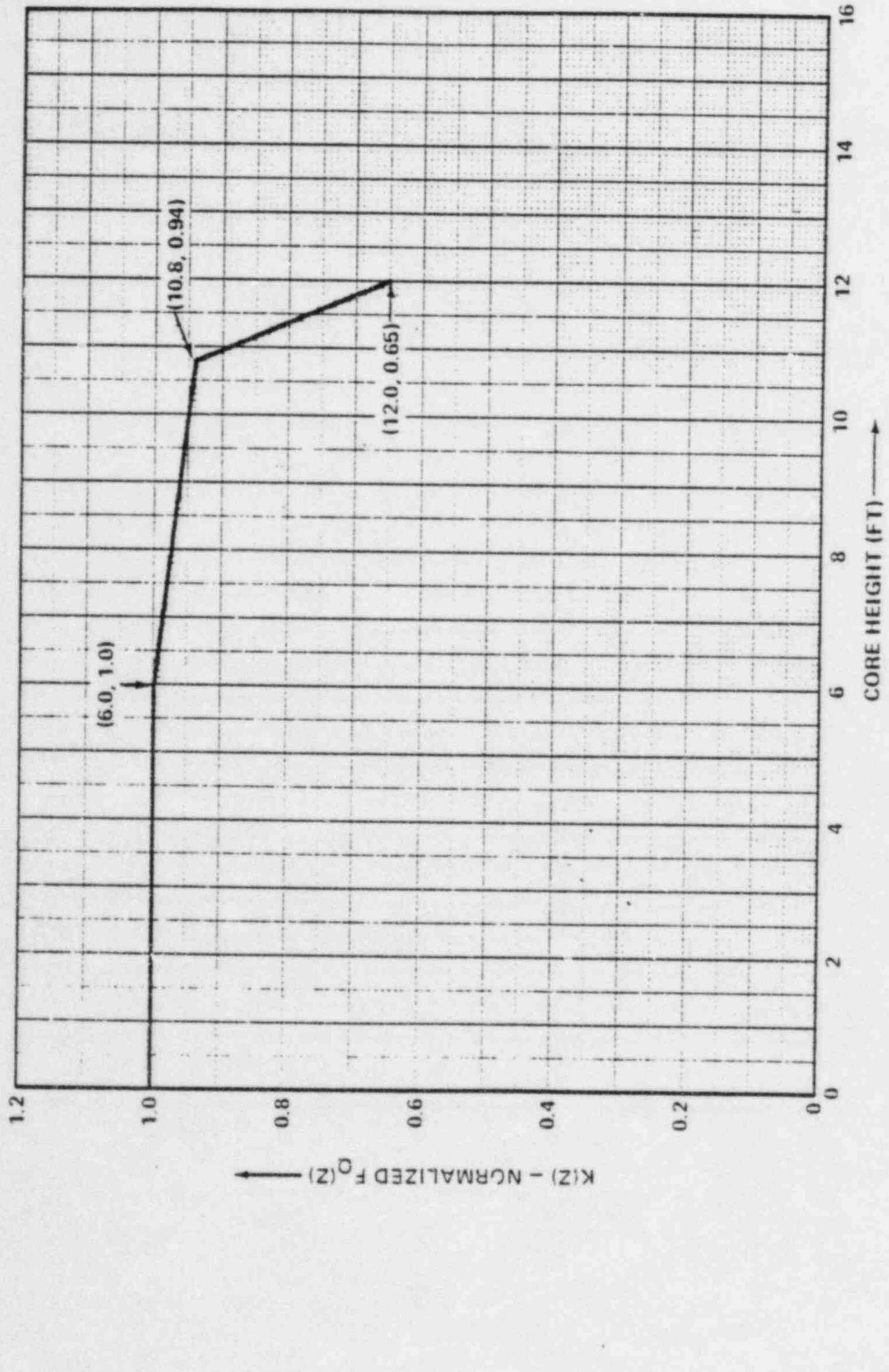

FIGURE 3.2-2

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER;

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties;

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specifications 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP} [1+0.2(1-P)]$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^{L} either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes ~~penalties for~~ measurement uncertainties of 2.1 ~~2.0%~~ for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

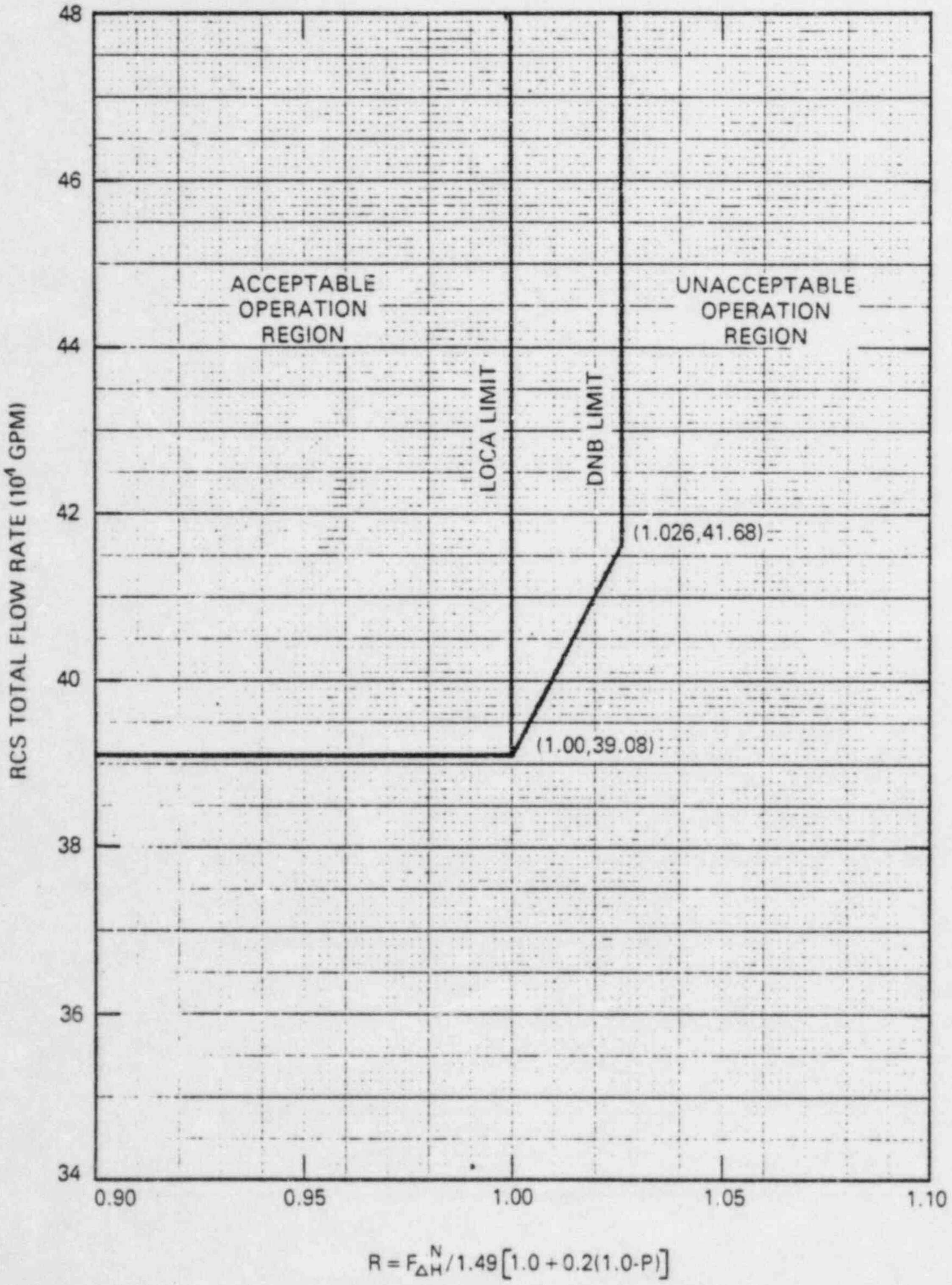
ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

   1. Restore the combination of RCS total flow rate and R to within the above limits, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.



MEASUREMENT UNCERTAINTIES OF 2.1% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F_{\Delta H}^{N}$ ARE INCLUDED IN THIS FIGURE

FIGURE 3.2-3

RCS TOTAL FLOW RATE VERSUS R

FOUR LOOPS IN OPERATION

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ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

    1. A nominal 50% of RATED THERMAL POWER,

    2. A nominal 75% of RATED THERMAL POWER, and

    3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. within 7 days prior to performing the precision heat balance the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

3/4.2.4 QUADRANT POWER TILT RATIO

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LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

  1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

     a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

     b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

  2. Within 2 hours either:

     a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

     b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

  3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours, and

  4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

---

*See Special Test Exception 3.10.2.

Specification

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

   2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes;

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

   4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

   1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

      a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

      b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

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ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, and

b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation	Three Loops in Operation
Reactor Coolant System T_{avg}	≤ ~~595~~ 592.5°F	**
Pressurizer Pressure	≥ 2220 ~~psia~~ psig*	**

---

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of three-loop operation.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	~~10~~ 5
~~c. Shutdown~~	~~2~~	~~0~~	~~1~~	~~3, 4, 5~~	~~5~~
7. Overtemperature ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	**	**	**	**	**
8. Overpower ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	**	**	**	**	**
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level-High	3	2	2	1	7#
12. Reactor Coolant Flow - Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop two operating loops	2/loop in ~~each~~ two operating loop	2/loop each operating loop	1	7#
13. Steam Generator Water Level-Low-Low	4/stm. gen. in any operating stm. gen.	2/stm. gen. ~~each~~ in any operating stm. gen.	3/stm. gen. each operating stm. gen	1, 2	6#
14. Undervoltage-Reactor Coolant Pumps	4-2/bus	2-1/bus	3	1	6#
15. Underfrequency-Reactor Coolant Pumps	4-2/bus	2-1/bus	3	1	6#
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	7#
b. Turbine Stop Valve Closure	4	4	1	1	11#
17. Safety Injection Input from ESF	2	1	2	1, 2	9

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
f. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
19. Reactor Trip Breakers	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*⑩ →	
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	9
	2	1	2	3*, 4*, 5*	10

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

Only if

*~~With~~ the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System capable of rod withdrawal.

**Values left blank pending NRC approval of three loop operation.

#The provisions of Specification 3.0.4 are not applicable.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1; and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, veri~~fy compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1~~ or 3.1.1.2, ~~as applicable, within 1 hour and at least once per 12~~ hours ~~thereafter~~. restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers, ~~suspend all operations involving positive reactivity changes and verify Valves BGV178 and BGV601 are closed and secured in position within the next hour.~~

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	N.A.
2.	Power Range, Neutron Flux	≤ 0.5 second*
3.	Power Range, Neutron Flux, High Positive Rate	N.A.
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5.	Intermediate Range, Neutron Flux	N.A.
6.	Source Range, Neutron Flux	N.A.
7.	Overtemperature ΔT	≤ 6.0 seconds*
8.	Overpower ΔT	≤ 6.0 seconds*
9.	Pressurizer Pressure-Low	≤ 2.0 seconds
10.	Pressurizer Pressure-High	≤ 2.0 seconds
11.	Pressurizer Water Level-High	N.A

---

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Reactor Coolant Flow-Low	
a. Single Loop (Above P-8)	≤ 1.0 second
b. Two Loops (Above P-7 and below P-8)	≤ 1.0 second
13. Steam Generator Water Level-Low-Low	≤ 2.0 seconds
14. Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds
15. Underfrequency-Reactor Coolant Pumps	≤ 0.6 second
16. Turbine Trip	
a. Low Fluid Oil Pressure	N.A.
b. Turbine Stop Valve Closure	N.A.
17. Safety Injection Input from ESF	N.A.
18. Reactor Trip System Interlocks	N.A.
19. Reactor Trip Breakers	N.A.
20. Automatic Trip and Interlock Logic	N.A.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT		CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2.	Power Range, Neutron Flux						
	a. High Setpoint	S	D(2, 4) M(3, 4) Q(4, 6) R(4, 5)	M	N.A.	N.A.	1, 2
	b. Low Setpoint	S	R(4)	M	N.A.	N.A.	1###, 2
3.	Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
4.	Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	M	N.A.	N.A.	1, 2
5.	Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1),M	N.A.	N.A.	1###, 2
6.	Source Range, Neutron Flux	S	R(4, 5, ~~12~~)	S/U(1),M(9)	N.A.	N.A.	2##, 3, 4, 5
7.	Overtemperature ΔT	S	R(13)	M	N.A.	N.A.	1, 2
8.	Overpower ΔT	S	R	M	N.A.	N.A.	1, 2
9.	Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	1
10.	Pressurizer Pressure-High	S	R	M	N.A.	N.A.	1, 2
11.	Pressurizer Water Level--High	S	R	M	N.A.	N.A.	1
12.	Reactor Coolant Flow--Low	S	R	M	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	TRIP ANALOG CHANNEL OPERATIONAL TEST	ACTUATING DEVICE OPERATIONAL TEST	MODES FOR WHICH ACTUATION LOGIC TEST	SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	M	N.A.	1
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	M	N.A.	N.A.	2##
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	M(8)	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	M(8)	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	M(8)	N.A.	N.A.	1

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
e. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(8)	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

Only if happen to be

*~~With~~ the Reactor Trip System breakers closed and the control rod drive system is capable of rod withdrawal.

##Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1)If not performed in previous 7 days.

(2)Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3)Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4)Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5)Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6)Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7)Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8)With power greater than or equal to the interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.

(9)Monthly surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. ~~Monthly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.~~

(10)Setpoint verification is not required.

(11)At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12)~~At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves will close and the centrifugal charging pumps suction valves from the RWST will open within 30 seconds.~~

(13)CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

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3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable take the ACTION shown in Table 3.3-3

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection, (Phase "A" Isolation, Reactor Trip, Feedwater Isolation, Turbine Trip, Auxiliary Feedwater - Motor Driven Pump, Emergency Component Cooling Water, ~~Start~~ Diesel Generator~~s~~ Operation, Containment Cooling, and Essential Service Water Operation)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure - Low	4	2	3	1, 2, 3#	19*
e. Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3##	15*
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair operated simultaneously	2 pair	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-3	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	MINIMUM TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
b. Phase "B" Isolation					
1) Manual Initiation	2 pair	1 pair operated simultaneously	2 pair	1, 2, 3, 4	18
2) Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
3) Containment Pressure-High-3	4	2	3	1, 2, 3	16
c. Containment Purge Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	17
4 ~~3~~) Phase "A" Isolation	See Item 3.a. for all Phase "A" Isolation initiating functions and requirements.				
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	2	1	2	1, 2, 3, 4	17

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	1/steam line	1/steam line	1/operating steam line	1, 2, 3	23
2) System	2	1	2	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3	21
c. Containment Pressure-High-2	3	2	2	1, 2, 3	15*
d. Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#~~#~~	15*
e. Steam Line Pressure-Negative Rate-High	3/steam line	2/steam line any steam line	2/steam line	3##~~#, 4~~	15*
5. Turbine Trip & Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relay (SSPS)	2	1	2	1, 2	~~21~~ 27
b. Steam Generator Water Level-High-High	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	19*
c. Safety Injection	See Item 1 above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	MINIMUM TOTAL NO. CHANNELS OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS APPLICABLE OPERABLE	MODES	ACTION
6. Auxiliary Feedwater					
a. Manual Initiation	3(1/pump) ~~2~~	1/pump ~~1~~	1/pump ~~2~~	1, 2, 3	24 ~~22~~
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3	21
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	2	1	2	1, 2, 3	21
d ~~c~~. Stm. Gen. Water Level-Low-Low					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	19*
e ~~d~~. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
f ~~e~~. Loss-of-Offsite Power - Start Turbine-Driven Pump	2	1	2	1,2,3	22

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	MINIMUM TOTAL NO. CHANNELS OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS APPLICABLE OPERABLE	MODES	ACTION
5. Auxiliary Feedwater (Continued)					
g ~~f.~~ Trip of All Main Feedwater Pumps - Start Motor-Driven Pumps	4-(2/pump)**	2-(1/pump in same separation)	3-~~(2/pump) in same separation)~~	1, 2	19
h ~~g.~~ Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to ESW)	3	2	2	1, 2, 3	15*
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
b. RWST Level - Low-Low Coincident With Safety Injection	4	2	3	1, 2, 3, 4	16
	See Item 1. above for Safety Injection initiating functions and requirements.				
8. Loss of Power					
a. 4 kV Bus Undervoltage -Loss of Voltage	4/Bus	2/Bus	3/Bus	1, 2, 3, 4	19*
b. 4 kV Bus Undervoltage -Grid Degraded Voltage	4/Bus	2/Bus	3/Bus	1, 2, 3, 4	19*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Control Room Isolation					
a. Manual Initiation	2	1	2	All	~~18~~ 26
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	~~All~~ 1,2,3,4	~~14~~ 26
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	2	1	2	All	~~14~~ 26
d ~~e.~~ Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation initiating functions and requirements.				
11 ~~10.~~ Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Reactor Trip, P-4	4-2/Train	2/Train	2/Train	1, 2, 3	22
10. Solid-State Load Sequencer	2 - 1/Train	1/Train	2 - 1/Train	1,2,3,4	25

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TABLE 3.3-3 (Continued)

TABLE NOTATIONS

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

~~##Trip function may be bypassed in this MODE above the P-11 (Pressurizer Pressure Interlock) Setpoint.~~

###Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

*The provisions of Specification 3.0.4 are not applicable.

**One in Separation Group 1 and one in Separation Group 4.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

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TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperate channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the action required by Specification 3.7.1.5.

ACTION 24 - See attached ACTION 24
ACTION 25 - See attached ACTION 25
ACTION 26 - See attached ACTION 26
ACTION 27 - See attach Action 27

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, declare the affected auxiliary feedwater pump inoperable and take the ACTION required by Specification 3.7.1.2.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, declare the affected diesel generator and off-site power source inoperable and take the ACTION required by Specification 3.8.1.1.

ACTION 26 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or initiate and maintain operation of the Control Room Emergency Ventilation System.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection Phase "A" Isolation (Reactor Trip, ~~Feedwater Isolation~~ Turbine Trip, ~~Component Cooling Water~~ Auxiliary Feedwater - Motor-Driven Pump, ~~Start~~ Emergency Diesel Generator Operation, Containment Cooling, and Essential Service Water Operation)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure High-1	~~2.5~~ 3.6	0.71	~~1.5~~ 1.48	≤ 3.5 psig	≤ ~~4.0~~ 4.5 psig
d. Pressurizer Pressure - Low ~~13.1~~ 16.2	10.71	~~1.5~~ 2.49	≥ ~~1849~~ 1830 psig	≥ ~~1839~~ 1815 psig	
e. Steam Line Pressure - Low ~~14.2~~ 19.6	~~10.71~~ 14.81	~~1.5~~ 1.93	≥ ~~585~~ 615 psig	≥ ~~564~~ 571 psig*	

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-3	4.3 ~~5.0~~	0.71	1.98 ~~1.5~~	≤ 27.0 psig	≤ 28.3 ~~28.0~~ psig
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure-High-3	4.3 ~~5.0~~	0.71	1.98 ~~1.5~~	≤ 27.0 psig	≤ 28.3 ~~28.0~~ psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation (Continued)					
c. Containment Purge Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
4 ~~3~~) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-2	4.3 ~~5.0~~	0.71	1.98 ~~1.5~~	≤ 17.0 psig	≤ ~~18.0~~ 18.3 psig
d. Steam Line Pressure-Low 14.[illegible] 14.6 →	14.81 ~~10.71~~ →		1.43 ~~1.5~~ →	≥ ~~585~~ 615 psig →	≥ ~~564~~ 571 psig* →
e. Steam Line Pressure-Negative Rate - High	3.0 ~~8.0~~	0.5	0	≤ (100 psi/s)	≤ (~~111.6~~ 125 psi/s)**

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	TOTAL	SENSOR	TRIP	ALLOWABLE	
FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	ERROR (S)	SETPOINT	VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High	5.0	2.18	2.51 ~~1.5~~ narrow range instrument span	< 78% of instrument span	< 79.7 ~~79.8~~% of narrow range
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Automatic Actuation Logic and Actuation Relay (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d ~~c.~~ Steam Generator Water Level-Low-Low					
1) Start Motor-Driven Pumps	23.5 ~~30.0~~	21.18 ~~27.18~~	2.51 ~~1.5~~ narrow range instrument span	> 23.5 ~~32.2~~% of instrument span	> 22.3 ~~30.4~~% of narrow range
2) Start Turbine-Driven Pumps	23.5 ~~30.0~~	21.18 ~~27.18~~	2.51 ~~1.5~~ narrow range instrument span	> 23.5 ~~32.2~~% of instrument span	> 22.3 ~~30.4~~% of narrow range

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
d. Safety Injection - Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power-Start Turbine-Driven Pump	N.A.	N.A.	N.A.	N.A.	N.A.
f. Trip of All Main Feedwater Pumps - Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to ESW)	N.A.	N.A.	N.A.	≥ 21.56 psia ~~(LATER)~~	≥ 20.58 psia ~~(LATER)~~
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level-Low-Low Coincident with Safety Injection	3.4 ~~N.A.~~	1.21 ~~N.A.~~	1.86 ~~N.A.~~	≤ 36.0% of ~~18%~~ instrument span	≤ 35.1% of ~~15%~~ instrument span
	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. 4 kV Undervoltage -Loss of Voltage	N.A.	N.A.	N.A.	≥ 83V (120V Bus) w/1s delay ~~(LATER)~~	≥ 74.7V (120V Bus) w/1+0.2,-0.5s delay ~~(LATER)~~
b. 4 kV Undervoltage -Grid Degraded Voltage	N.A.	N.A.	N.A.	≥ 107.1V (120V Bus) w/119s delay ~~(LATER)~~	≥ 104.5V (120 Bus) w/119 ± 11.6s delay ~~(LATER)~~
9. Control Room Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d ~~c.~~ Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
11 ~~10.~~ Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1970 psig	≤ ~~1981~~ 1979 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
10. Solid-State Load Sequencer	N.A.	N.A.	N.A.	N.A	N.A.

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Purge Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Service Water	N.A.
j Containment Cooling	N.A.
k. Control Room Isolation	N.A.
l. Reactor Trip	N.A.
m. ~~Start~~ Emergency Diesel Generator	N.A.
n. Component Cooling Water	N.A.
o. Turbine Trip	N.A.
2. Containment Pressure-High-1	
a. Safety Injection (ECCS)	$\leq 29^{(1)}/12^{(\overset{4}{\cancel{5}})}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 7^{(\cancel{2})}$
3) Phase "A" Isolation	$\leq 1.5^{(\overset{5}{\cancel{7}})}$
~~4) Containment Purge Isolation~~	~~≤ 4.5~~
4 ~~5~~) Auxiliary Feedwater	≤ 60
5 ~~6~~) Essential Service Water	$\leq 60^{(1)}$
6 ~~7~~) Containment Cooling	$\leq 60^{(1)}$
7 ~~8~~) Component Cooling Water	N.A.
8 ~~9~~) ~~Start~~ Emergency Diesel Generator	$\leq 14^{(\overset{6}{\cancel{8}})}$
9) Turbine Trip	N.A.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 29^{(1)}/12^{(\overset{4}{\cancel{8}})}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	$\leq 7^{(\cancel{2}3)}$
3) Phase "A" Isolation	$\leq 2^{(\overset{5}{\cancel{7}})}$
~~4) Containment Purge Isolation~~	~~≤ 4.5~~
4~~5~~) Auxiliary Feedwater	≤ 60
5~~6~~) Essential Service Water	$\leq 60^{(1)}$
6~~7~~) Containment Cooling	$\leq 60^{(1)}$
7~~8~~) Component Cooling Water	N.A.
8~~9~~) ~~Start~~ Emergency Diesel Generator	$\leq 14^{(\overset{6}{\cancel{2}})}$
9 Turbine Trip	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq \overset{24}{\cancel{22}}^{(\overset{3}{\cancel{4}})}/12^{(\overset{4}{\cancel{2}})}$
1) Reactor Trip (from SI)	≤ 2
2) Feedwater Isolation	$\leq 7^{(\cancel{2})}$
3) Phase "A" Isolation	$\leq 2^{(\overset{5}{\cancel{7}})}$
~~4) Containment Purge Isolation~~	~~≤ 5~~
4~~5~~) Auxiliary Feedwater	≤ 60
5~~6~~) Essential Service Water	$\leq 60^{(1)}$
6~~7~~) Containment Cooling ~~Fans~~	$\leq 60^{(1)}$
7~~8~~) Component Cooling Water	N.A.
8~~9~~) ~~Start~~ Emergency Diesel Generators	$\leq 14^{(\overset{6}{\cancel{2}})}$
9. Turbine Trip	N.A.
b. Steam Line Isolation	≤ 7

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Containment Pressure--High-3	
a. Containment Spray	≤ ~~60~~ 32(1)/~~50~~ 20(2) ~~/30(8)~~
b. Phase "B" Isolation	≤ 31.5
6. Containment Pressure--High-2	
Steam Line Isolation	≤ 7
7. Steam Line Pressure-Negative Rate-High	
Steam Line Isolation	≤ 7 ~~N.A.~~
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 7(3)
9. Steam Generator Water Level - Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine-Driven Auxiliary Feedwater Pumps	≤ 60
10. Loss-of-Offsite Power	
Start Turbine-Driven Auxiliary Feedwater Pumps	N.A.
11. Trip of All Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	N.A

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Auxiliary Feedwater Pump Suction Pressure-Low	
Transfer to Essential Service Water	N.A.
13. RWST Level-Low-Low Coincident with Safety Injection	
Automatic Switchover to Containment Sump	≤ 60
14. Loss of Power	
a. 4 kV Bus Undervoltage-Loss of Voltage	≤ ~~13~~ 14
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	≤ 14~~4~~
15. Phase "A" Isolation	
a. Control Room Isolation	N.A.
b. Containment Purge Isolation	≤ 2(5)

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TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting ~~and sequence loading~~ delay not included. Offsite power available.

~~(3) Air operated valves.~~

(3) ~~(4)~~ Diesel generator starting and sequence loading delay included. RHR pumps not included.

(4) ~~(5)~~ Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included.

~~(6) Sequence delays not included.~~

(5) ~~(7)~~ Does not include valve closure time.

(6) ~~(8)~~ Includes time for diesel to reach full speed.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, ~~&~~Component Turbine Trip, Cooling Water, ~~& Start~~ Diesel Auxiliary Feedwater-Motor-Driven Pump, Emergency Operation Generators, Containment Cooling, and Essential Service Water) Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Purge Isolation								
1) Manual Initiation	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(1)(2)	N.A.	N.A.	1, 2, 3, 4
4 ~~3~~) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S S	R R	M M	N.A. N.A.	N.A. N.A.	N.A. N.A.	N.A. N.A.	1, 2, 3 3, ④
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2
b. Steam Generator Water Level-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
d ~~c.~~ Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A	N.A.	M(1)(2)	N.A	N.A	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
d. Safety Injection	See Item 1 above for all Safety Injection Surveillance Requirements							
e. Loss-Offsite Power	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2
g. Auxiliary Feedwater Pump Suction Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
b. RWST Level - Low-Low Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Loss of Power								
a. 4 kV Undervoltage - Loss of Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4 kV Undervoltage - Grid Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Control Room Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	~~All~~ 1, 2, 3, 4
~~c.~~ d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(1)(2)	N.A.	N.A.	All
10. Solid State Load Sequencer	N.A.	N.A.	N.A.	N.A.	M(1)(2)	N.A.	N.A.	1, 2, 3, 4
~~10.~~ 11. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

---

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Continuity check may be excluded from the ACTUATION LOGIC TEST

(3) Except Relays K602, K620, K622, K624, K630, K740 and K741 which shall be tested, at least once per 18 months during refueling and during each COLD SHUTDOWN exceeding 24 hours unless they have been tested within the previous 90 days.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Gaseous Radioactivity-High (GT-RE-31 & 32)	1	2	All	### ~~≤ 2 mR/h~~	26
~~b. Containment Purge Exhaust Radioactivity-High~~	~~1~~	~~2~~	~~All~~	~~#~~	~~26~~
c. Gaseous Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
d. Particulate Radioactivity RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
2. Fuel Building					
a. Fuel Building Exhaust- ~~Spent Fuel Pool~~ Gaseous Radioactivity-High (GG-RE-27 & 28)	1	2	**	## ~~≤ 2 mR/h~~	30 ~~27~~
b. Criticality-High Radiation Level (SD-RE-37 & 38)	1	2	*	≤ 15 mR/h	28
3. Control Room					
Air Intake - Gaseous Radioactivity-High (GK-RE-04 & 05)	1	2	All	# ~~≤ 2 mR/h~~	27

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TABLE 3.3-6 (Continued)

TABLE NOTATIONS

*With fuel in the fuel storage areas ~~of~~ or fuel building.

**With irradiated fuel in the fuel storage areas ~~of~~ or fuel building.

#~~Must satisfy Specification 3.11.2.1 requirements~~. Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 2 mR/hr in the Control Room

See attached ## note

See attached ### note

ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge valves are maintained closed.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel ~~pool~~ area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel building.

ACTION 29 - Must satisfy the ACTION requirements for Specification 3.4.6.1.

ACTION 30 — See attached Action 30

##Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 4 mR/h in the fuel building.

###Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 9 mR/h in the containment building. The Setpoint value may be increased up to the equivalent limits of Specification 3.11.2.1 in accordance with the methodology and parameters in the ODCM during containment purge or vent provided the Setpoint value does not exceed twice the maximum concentration activity in the containment determined by the sample analysis performed prior to each release in accordance with Table 4.11-2.

ACTION 30 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Fuel Building Ventilation System and initiate operation of the Emergency Exhaust System to maintain the fuel building at a negative pressure.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Gaseous Radioactivity-High (GT-RE-31 & 32)	S	R	M	All
~~b. Containment Purge Exhaust Radioactivity-High~~	~~S~~	~~R~~	~~M~~	~~All~~
c. Gaseous Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	S	R	M	1, 2, 3, 4
d. Particulate Radioactivity - RCS Leakage Detection (GT-RE-31 & 32)	S	R	M	1, 2, 3, 4
2. Fuel Building				
a. Fuel Building Exhaust- ~~Spent Fuel Pool~~ Gaseous Radioactivity-High (GG-RE-27 & 28)	S	R	M	**
b. Criticality-High Radiation Level (SD-RE-37 & 38)	S	R	M	*
3. Control Room				
Air Intake - Gaseous Radioactivity-High (GK-RE-04 & 05)	S	R	M	All

---

*With fuel in the fuel storage areas or fuel building.

**With irradiated fuel in the fuel storage areas or fuel building.

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MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}

ACTION:

a. With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$, and F_{xy}.

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SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Peak Recording Accelerographs		
a. Radwaste Base Slab	± 1.0g	1
b. Control Room	± 1.0g	1
c. ESW Pump Facility	± 1.0g	1
d. Ctmt Structure	± 2.0g	1
e. Auxiliary Bldg. SI Pump Suctions	± 1.0g	1
f. SGB Piping	± 2.0g	1
g. SGB Support	± 1.0g	1
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)		
a. Ctmt. Base Slab	± 1.0g	1
b. Ctmt. Oper. Floor	± 1.0g	1
c. Reactor Support	± 1.0g	1
d. Aux. Bldg. Base Slab	± 1.0g	1
e. Aux. Bldg. Control Room Air Filter	± 1.0g	1
f. Free Field	± 0.5g	1
3. Triaxial Response-Spectrum Recorder (Passive)		
Ctmt. Base Slab	± 1.0g	1

4. Triaxial Seismic Switches		ACCELERATION LEVEL		
	North	East	Vertical	
a. OBE Ctmt. Base Slab	0.06 g	0.06g	0.06g	1
b. SSE Ctmt. Base Slab	0.15 g	0.15g	0.16 g	1
c. OBE Ctmt. Oper. Fl.	0.07g	~~(0.06)g~~ 0.07g	0.07 g	1
d. SSE Ctmt. Oper. Fl.	0.16g	~~(0.16)g~~ 0.17g	0.16g	1
e. System Trigger	0.01g	(0.01)g	0.01g	1

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Peak Recording Accelerographs			
a. Radwaste Base Slab	N.A.	R	N.A.
b. Control Room	N.A.	R	N.A.
c. ESW Pump Facility	N.A.	R	N.A.
d. Ctmt Structure	N.A.	R	N.A.
e. Auxiliary Bldg. SI Pump Suction	N.A.	R	N.A.
f. SGB Piping	N.A.	R	N.A.
g. SGB Support	N.A.	R	N.A.
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)			
a. Ctmt. Base Slab	M	R	SA
b. Ctmt. Oper. Floor	M	R	SA
c. Reactor Support	M	R	SA **
d. Aux. Bldg. Base Slab	M	R	SA **
e. Aux. Bldg. Control Room Filters	M	R	SA **
f. Free Field	M	R	SA **
3. Triaxial Response-Spectrum Recorder (Passive)			
Ctmt. Base Slab	N.A.	R	~~SA~~ NA. *
4. Triaxial Seismic Switches			
a. OBE Ctmt. Base Slab	M	R	SA
b. SSE Ctmt. Base Slab	M	R	SA
c. OBE Ctmt. Oper. Fl.	M	R	SA
d. SSE Ctmt. Oper. Fl.	M	R	SA
e. System Trigger	M	R	SA

---

* Checking at the Main Control Board Annunciators for contact closure output in the Control Room shall be performed at least once per 184 days.

** The bistable trip setpoint need not be determined during the performance of an ANALOG CHANNEL OPERATIONAL TEST.

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METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed	Nominal Elev. 10m	1
	Nominal Elev. 60m	1
2. Wind Direction	Nominal Elev. 10m	1
	Nominal Elev. 60m	1
3. Air Temperature - ΔT	Nominal Elev. 10m-60m	1

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 10m	D	SA
b. Nominal Elev. 60m	D	SA
2. Wind Direction		
a. Nominal Elev. 10m	D	SA
b. Nominal Elev. 60m	D	SA
3. Air Temperature - ΔT		
a. Nominal Elev. 10-60m	D	SA

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REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels given in Table 3.3-9 and the auxiliary shutdown panel (ASP) controls shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY:* MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days; ~~or be in~~ otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the ~~next 12~~ following 6 hours.

c ~~b~~. The provisions of Specification 3.0.4 are not applicable.

b. With the ASP controls inoperable, restore the inoperable ASP controls to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies given in Table 4.3-6.

4.3.3.5.2 The ASP controls shall be demonstrated OPERABLE at least once per 18 months, by operating each actuated component from the ASP.

4.3.3.5.3 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven auxiliary feedwater pump or the atmospheric dump valves.

---

* The Source Range Neutron Flux Monitor is not required to be Operable in Mode 1 and Mode 2 above the P6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Specification 3.3.3.5

Justification -

The asterisked note has been added to the applicability requirements of this specification to be consistent with the applicability requirements of Table 3.3-1 (item 6).

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. RCS Pressure-Wide Range	ASP*	2	1
2. Reactor Coolant Temperature-Cold Leg	ASP*	4	1
3. Source Range Neutron Flux	ASP*	2	1
4. Reactor Trip Breaker Indication	RTS**	1/trip breaker	1/trip breaker
5. Reactor Coolant Temperature - ~~Average~~ Hot Leg	ASP*	2	1
6. Reactor Coolant Pump Breakers	***	1/pump	1/pump
7. Pressurizer Pressure	ASP*	1	1
8. Pressurizer Level	ASP*	2	1
9. Steam Generator Pressure	ASP*	2/~~loop~~ steam generator	1/~~loop~~ steam generator
10. Steam Generator Level	ASP*	2/steam generator	1/steam generator
11. Auxiliary Feedwater Flow Rate	ASP*	4	1
12. Auxiliary Feedwater Suction Pressure	ASP*	3	1

---

*Auxiliary Shutdown Panel

**Reactor Trip Switchgear

***13.8 kV Switchgear

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. RCS Pressure - Wide Range	M	R
2. Reactor Coolant Temperature - Cold Leg	M	R
3. Source Range, Neutron Flux	M	R
4. Reactor Trip Breaker Indication	M	N.A.
5. Reactor Coolant Temperature - Hot Leg	M	R
6. Reactor Coolant Pump Breakers	N.A.	N.A.
7. Pressurizer Pressure	M	R
8. Pressurizer Level	M	R
9. Steam Generator Pressure	M	R
10. Steam Generator Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Auxiliary Feedwater Pump Suction Pressure	M	R

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days; otherwise, ~~or~~ be in at least HOT STANDBY ~~SHUTDOWN~~ within the next 6 ~~12~~ hours and in HOT SHUTDOWN within the following 6 hours.

b. With the number of OPERABLE accident containment monitoring instrumentation channels except the ~~reactor coolant~~ radiation level monitor and the unit vent - high range noble gas monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours; otherwise, ~~or~~ be in at least HOT STANDBY ~~SHUTDOWN~~ within the next 6 ~~12~~ hours and in HOT SHUTDOWN within the following 6 hours

d ~~c~~. The provisions of Specification 3.0.4 are not applicable.

c. See attached action c.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

ACTION:

c. With the number of OPERABLE channels for the containment radiation level monitor or the unit vent - high range noble gas monitor less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore the inoperable channel to OPERABLE status within 7 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days that provides actions taken, cause of the inoperability and plans and schedule for restoring the channels to OPERABLE status.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure - a) Normal ~~Dual~~ Range b) Extended Range	2 2	1 1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Containment Hydrogen Concentration Level	2	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
~~12. Reactor Coolant System Subcooling Margin Monitor~~#	~~2~~	~~1~~
12 ~~13~~. PORV Position Indicator*	1/Valve	1/Valve
13 ~~14~~. PORV Block Valve Position Indicator**	1/Valve	1/Valve
14 ~~15~~. Safety Valve Position Indicator	1/Valve	1/Valve
15 ~~16~~. Containment Water Level	2	1
16 ~~17~~. Containment Radiation Level (High Range)	~~2~~ N.A.	1
17 ~~18~~. Thermocouple/Core Cooling Detection System	~~2~~ 4/core quadrant	~~1~~ 2/core quadrant
~~18 19. Reactor Coolant Radiation Level~~ #	~~1/stm. line~~	~~1/stm. line~~***
18 ~~20~~. Unit Vent - High Range~~s~~ Noble Gas Monitor	~~1/stack~~ N.A.	1~~/stack~~

---

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

***If inoperable, Action a. ~~in Specification 3.3.3.6 shall be applicable~~.

these ~~instruments need not be~~ required OPERABLE ~~until prior to~~ STARTUP following the first ~~refueling outage~~

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Containment Pressure - ~~Dual Range~~	M	R
2.	Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3.	Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4.	Reactor Coolant Pressure - Wide Range	M	R
5.	Pressurizer Water Level	M	R
6.	Steam Line Pressure	M	R
7.	Steam Generator Water Level - Narrow Range	M	R
8.	Steam Generator Water Level - Wide Range	M	R
9.	Refueling Water Storage Tank Water Level	M	R
10.	Containment Hydrogen Concentration Level	M	R
11.	Auxiliary Feedwater Flow Rate	M	R
~~12.~~	~~Reactor Coolant System Subcooling Margin Monitor #~~	~~M~~	~~R~~
12 ~~13~~.	PORV Position Indicator*	M	N.A.
13 ~~14~~.	PORV Block Valve Position Indicator**	M	N.A.
14 ~~15~~.	Safety Valve Position Indicator	M	N.A.
15 ~~16~~.	Containment Water Level	M	R
16 ~~17~~.	Containment Radiation Level (High Range)	M	R***
17 ~~18~~.	Thermocouple/Core Cooling Detection System	M	R
~~19.~~	~~Reactor Coolant Radiation Level #~~	~~M~~	~~R~~
18 ~~20~~.	Unit Vent - High Range Noble Gas Monitor	M	R

---

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

***CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

~~# These instruments need not be required OPERABLE until prior to STARTUP following the first refueling outage.~~

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

a. With one Chlorine Detection System inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable or with any Function B fire detection instruments shown in Table 3.3-11, inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1101-Aux. Bldg. 1974' Gen. Flr. #1	100			0/11
1102-Chiller & Surge Tks. Area	100			0/4
1102-Chiller & Surge Tks. Area	101			2/0
1107-Cent. Charg. Pmp. Rm. B	101			2/0
1108-Safety Inj. Pmp. Rm. B	101			2/0
1109-Res. Ht. Remov. Pmp. Rm. B	101			1/0
1110-Ctmt. Spray Pmp. Rm. ~~A~~B	101			1/0
1111-Res. Ht. Remov. Pmp. Rm. A	101			1/0
1112-Ctmt. Spray Pmp. Rm. A	101			1/0
1113-Safety Inj. Pmp. Rm. A	101			2/0
1114-Cent. Charg. Pmp. Rm. A	101			2/0
1115-Pos. Disp. Charg. Pmp. Rm.	101			2/0
1116, 1117-Boric Acid Tk. Rms.	101			2/0
1116, 1117-Boric Acid Tk. Rms.	101		2/0	
1120-Aux. Bldg. 1974' Gen. Flr. #2	101			4/0
1122-Aux. Bldg. 1974' Gen. Flr. #3	100			0/3
1122-Aux. Bldg. 1974' Gen. Flr. #3	101			5/0
1126-Boron Inj. Tk. & Pmp. Rm.	101			1/0
1127-Stair A-Z	109			1/0
1128-	117			2/0
1130-Aux. Bldg. 1974' N. Corr.	100			0/2
1206-W. Pipe Chase Below AFWP Area	117			2/0
-Aux. Bldg. Elec. Chase S. 1988'	117			1/0
1301-Aux. Bldg. 2000' Corridor #1	103			0/10
1301-Aux. Bldg. 2000' Corridor #1	117			2/0
1311-Aux. Bldg. Sampling Rm.	117			2/0
1312-Boron Meter/RC Activity Mon. Rm.	103			0/1
1314-Aux. Bldg. 2000' Corridor #3	103			0/3
1314-Aux. Bldg. 2000' Corridor #3	117			2/0
1315-Ctmt. Spray Add. Tk. Area	103			0/2
1316-Vlv. Rm. by Seal Wtr. Ht. Exch.	103			0/1
1320-Aux. Bldg. 2000' Corridor #4	103			0/3
1321-Aux. Bldg. 2000' S. Exit Vest.	103			0/1
1322-Pipe Pene. Rm. B	117			5/0
1323-Pipe Pene. Rm. A	117			6/0
1325-Aux. FW Pmp. Rm. B	117			2/0
1326-Aux. FW Pmp. Rm. A	117			2/0
1331-Aux. FW Pmp. Rm. C	111	~~2/0~~	2/0	
1331-Aux. FW Pmp. Rm. C	117			1/0
1335-Aux. Bldg. Elec. Chase N. 2000'	117			1/0
1336-Aux. Bldg. Elec. Chase S. 2000'	117			1/0
1401-Comp. Cool. Pmp. & Ht. Exch. B	118			5/0
1402-Aux. Bldg. 2026' Corridor #1	104			0/6
1403-MG Set Rm.	105			0/9(1)
1403-MG Set Rm.	112			0/9(1)

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1405-Chemical Stg. Area	118			6/0
1406-Comp. Cool. Pmp. & Ht. Exch. A	104			0/1
1406-Comp. Cool. Pmp. & Ht. Exch. A	118			2/0
1408-Aux. Bldg. 2026' Corridor #2	104			0/9
1408-Aux. Bldg. 2026' Corridor #2	118			5/0
1409-Elec. Pene. Rm. B	106			0/4(1)
1409-Elec. Pene. Rm. B	113			0/4(1)
1410-Elec. Pene. Rm. A	107			0/8(1)
1410-Elec. Pene. Rm. A	114			0/8(1)
1413-Aux. Shutdown Pnl. Rm.	118			4/0
1501-Ctrl. Rm. A/C & Filt. Units B	110			10/0
1504-Ctmt. Purge Exh. & Mech. Equip. B	108			18/0
1506-Cmt. Purge Sup. AHU Rm. A	109			18/0
1507-Personnel Hatch Area	108			3/0
1512-Ctrl. Rm. A/C & Filt. Units A	110			10/0
1513-Ctrl. Bldg. Vent Sup. A/C Unit Rm.	109			3/0
Aux. Bldg. Duct 2047'6"	119			1/0
Containment**	201	1/0(2)		
Containment**	202	2/0(2)		
Containment**	203	1/0(2)		
Containment**	204	1/0(2)		
Containment**	206	3/0(2)		
Containment**	215	1/0(2)		
Containment**	216	1/0(2)		
Containment**	217	1/0(2)		
Containment**	218	1/0(2)		
Containment**	219			4/0
Containment**	220	1/0(2)		
3101-Ctrl. Bldg. 1974' Pipe Space	300			11/0
3105-Ctrl. Bldg. Elec. Chase S. 1974'	300			1/0
3106-Ctrl. Bldg. Elec. Chase N. 1974'	300			1/0
-Area Above Access Control	301			12/0
3229-Ctrl. Bldg. Elec. Chase S. 1984'	300			1/0
3230-Ctrl. Bldg. Elec. Chase N. 1984'	300			1/0
3301-ESF Swgr. Rm. #1	314			0/7(1)
3301-ESF Swgr. Rm. #1	315			0/7(1)
3302-ESF Swgr. Rm. #2	316			0/5(1)
3302-ESF Swgr. Rm. #2	317			0/5(1)
3305-Ctrl. Bldg. Elec. Chase S. 2000'	301			1/0
3306-Ctrl. Bldg. Elec. Chase N. 2000'	301			1/0
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	304			0/1(1)
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	305			0/1(1)
3404-Switchboard Rm. #4	321			0/2(1)
3404-Switchboard Rm. #4	322			0/2(1)
3405-Battery Rm. #4	303			2/0
3407-Battery Rm. #1	303			2/0
3408-Switchboard Rm. #1	325			0/2(1)

1509 - Main Steam Iso. Valve Rm. #2	115		1/0	
1508 - Main Steam Iso. Valve Rm. #1	115		1/0	

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
3408-Switchboard Rm. #1	326			0/2(1)
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	323			0/1(1)
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	327			0/2(1)
3410-Switchboard Rm. #2	324			0/2(1)
3410-Switchboard Rm. #2	328			0/2(1)
3411-Battery Rm. #2	303			2/0
3413-Battery Rm. #3	303			1/0
3414-Switchboard Rm. #3	318			0/2(1)
3414-Switchboard Rm. #3	320			0/2(1)
3415-Acc. Ctrl. & Elec. Equip. A/C Units #1	303			4/0
3416-Acc. Ctrl. & Elec. Equip. A/C Units #2	303			4/0
3418-Ctrl. Bldg. Elec. Chase S. 2016'	303			1/0
3419-Ctrl. Bldg. Elec. Chase N. 2016'	303			1/0
-Ctrl. Bldg. Elec. Chase N. 2016'	303			1/0
-Ctrl. Bldg. Elec. Chase S. 2016'	303			1/0
3501-Lower Cable Spreading Rm.	306			0/13
3504-Ctrl. Bldg. Elec. Chase N. 2032'	303			1/0
3505-Ctrl. Bldg. Elec. Chase S. 2032'	303			1/0
-Ctrl. Bldg. Elec. Chase N. 2032'	303			1/0
-Ctrl. Bldg. Elec. Chase S. 2032'	303			1/0
3601-Control Room	308			4/0
3601-Control Room	309			0/7(1)
3601-Control Room	319			0/7(1)
3601-Control Room >	329 >			20/0 >
3602-Pantry	308			1/0
3603-Shift Supv. Office	308			1/0
3605-Equipment Cabinet Area	308			13/0
3606-Emerg. Equip. Storage Rm.	308			1/0
3608-Janitor's Closet	308			1/0
3609-SAS Rm.	308			1/0
3617-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			1/0
3618-Ctrl. Bldg. Elec. Chase N. 2047'6"	308			1/0
-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			1/0
3801-Upper Cable Spreading Rm.	307			0/18
3804-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			1/0
-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			1/0
5201-W. Diesel Gen. Rm.	501		4/0	
5201-W. Diesel Gen. Rm.	502	0/8		
5203-E. Diesel Gen. Rm.	500		4/0	
5203-E. Diesel Gen. Rm.	503	0/8		
6102-Fuel Bldg. Railroad Bay	600	0/8		
6104-Fuel Pool Cool. HX Rm. B	601			6/0
6105-Fuel Pool Cool. HX Rm. A	601			6/0
6202-Elec. Equipment Rm.	601			3/0

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
6203-Air Handling Equip. Rm.	601			3/0
6301-Fuel Bldg. 2047'6" Gen. Flr.	602		2/0	
6303-Fuel Bldg. Exh. Filt. Absorb. Rm. A	601			2/0
6304-Fuel Bldg. Exh. Filt. Absorb. Rm. B	601			2/0
-~~North~~ ESW Pumphouse Train B	002			3/0
-~~South~~ ESW Pumphouse Train A	001			3/0
-~~ESW Cooling Tower~~	~~001~~			~~1/0~~
-~~ESW Cooling Tower~~	~~002~~			~~1/0~~
-ESF Transformer XNB01	016	0/6		
-ESF Transformer XNB02	017	0/6		

TABLE NOTATIONS

*(x/y): x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

**The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

(1) Zone is associated with a Halon-protected space. Each space has two separate detection circuits (zones). One zone, in its entirety, needs to remain operable.

upper case

(2) Line-type heat detector.

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INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.9 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACITON:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the Loose-Part Detection System shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of Setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

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RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7, why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor HB- (RE-18)	1	31
b. Steam Generator Blowdown Discharge Monitor BM- (RE-52)	1	32
c. Turbine Building Drain Monitor LE- (WRE-59)	1	32
d. Secondary Liquid Waste System Monitor HF- (RE-45)	1	33
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Discharge Line		
1) Waste Monitor Tank A Discharge Line	1	34
2) Waste Monitor Tank B Discharge Line	1	34
b. Steam Generator Blowdown Discharge Line	1	34
c. Secondary Liquid Waste System Discharge Line	1	34
~~d. Cooling Tower Blowdown Line~~	~~1~~	~~34~~

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TABLE 3.3-12 (Continued)

ACTION STATEMENTS

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release.

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are analyzed for principal gamma emitters and I-131 ~~radioactivity~~ for up to 30 days at a lower limit of detection ~~of no more than 10 microCurie/ml.~~ as specified in Table 4.11-1.

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

ACTION 33 - ~~With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml.~~

See attached Action 33

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

ACTION 33 - With the number of channels OPERABLE Less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Discharge Monitor (HB-RE-18)	D	P	R(2)	Q(1)
b. Steam Generator Blowdown Discharge Monitor (BM-RE-52)	D	M	R(2)	Q(1)
c. Turbine Building Drain Monitor (LE-RE-59)	D	M	R(2)	Q(1)
d. Secondary Liquid Waste System Monitor (HF-RE-45)	D	~~M~~ P	R(2)	Q(1)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Discharge Line	D(3)	N.A.	R	~~Q~~ N.A.
b. Steam Generator Blowdown Discharge Line	D(3)	N.A.	R	~~Q~~ N.A.
c. Secondary Liquid Waste System Discharge Line	D(3)	N.A.	R	~~Q~~ N.A.
~~d. Cooling Tower Blowdown Line~~	~~D(3)~~	~~N.A.~~	~~R~~	~~Q~~

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

as appropriate

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

(isolation and alarm)

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

(alarm only)

b. Circuit failure, or

SET

c. Instrument indicates a downscale failure ~~(alarm only)~~, or

d. Instrument controls not set in operate mode (alarm only).

(gas or liquid and solid)

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy, ~~and~~ measurement range, For subsequent CHANNEL CALIBRATION, ~~sources that have been related to the initial calibration shall be used.~~

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

and establish monitor response to a solid calibration source

NBS traceable standard (gas, liquid, or solid) may be used; or a sample of the process stream (gas, liquid, or solid) provided the activity of the stream is determined using plant equipment calibrated using NBS reference standards or standards obtained from suppliers that participate in measurement assurance activities with NBS.

Table 4.3-8, Table Notations

Justification -

The proposed wording currently in the Callaway Technical Specifications implies that the same geometry as found in the monitors will be used for calibrations. It is not always possible to duplicate geometry on count room equipment. KG&E's calibration program may also differ from other plants; therefore, a time frame is not justified. Note (2) has been modified to resolve these concerns. This modification allows the use of a process stream for calibration. A portion of the process stream can be analyzed for radioactivity using the currently calibrated system. This system is calibrated using NBS reference standards or standards obtained from suppliers participating in measurement asurance activities with the NBS.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION, or explain in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7, why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1/recombiner	**	~~42~~, 44
b. Oxygen Monitor	2/recombiner	**	42
2. Unit Vent System			
a. Noble Gas Activity Monitor - Providing Alarm (GT-RE-21)	1	*	40
b. Iodine ~~Activity Monitor (RE-21)~~ Sampler	1	*	43
c. Particulate ~~Activity Monitor (RE-21)~~ Sampler	1	*	43
d. Flow Rate ~~Monitor~~	1	*	~~39~~ 45
e. Sampler Flow Rate Monitor	1	*	39
3. Containment Purge System			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (GT-RE-22, GT-RE-33) ~~RE-31, RE-32)~~	1	*	41
b. Iodine Sampler	1	*	43
c. Particulate Sampler	1	*	43
d. Flow Rate	N.A.	*	45
e. Sampler Flow Rate Monitor	1	*	39

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Radwaste Building Vent System			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (GH-RE-10)	1	*	38, 40
b. Iodine ~~Activity Monitor (RE-10)~~ Sampler	1	*	43
c. Particulate ~~Activity Monitor (RE-10)~~ Sampler	1	*	43
d. Flow Rate ~~Monitor~~	~~1~~ N.A.	*	~~39~~ 45
e. Sampler Flow Rate Monitor	1	*	39

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 42 - ~~With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operation and at least once per 24 hours during other operations.~~

See attached Action 42

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sample equipment as required in Table 4.11-2.

ACTION 44 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, suspended oxygen supply to the recombiner.

ACTION 45 - Flow rate for this system shall be based on fan status and operating curves or actual measurements.

ACTION 42 - With the Outlet Oxygen Monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both oxygen channels or both ~~of~~ the inlet oxygen and inlet hydrogen monitors inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System					
a. Inlet Hydrogen Monitor	D	N.A.	Q(4)	M	**
b. Outlet Hydrogen Monitor	D	N.A.	Q(4)	M	**
c. Inlet Oxygen Monitor	D	N.A.	Q(5)	M	**
d. Outlet Oxygen Monitor	D	N.A.	Q(~~5~~6)	M	**
2. Unit Vent System					
a. Noble Gas Activity Monitor-Providing Alarm (GT-RE-21)	D	M	R(3)	Q(2)	*
b. Iodine ~~Activity Monitor (RE-21)~~ Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate ~~Activity Monitor (RE-21)~~ Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate ~~Monitor~~	~~D~~ N.A	N.A.	R(7)	~~Q~~ N.A.	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
3. Containment Purge System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (GT-RE-22, GT-RE-33) ~~RE-31, RE-32)~~	D	P	P(3)	Q(~~2~~1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate	N.A.	N.A.	R(7)	N.A.	*
e. Sampler Flow Rate Monitor	D	N.A.	R	~~Q~~ N.A.	*

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Radwaste Building Vent System					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RE-10)	D, P	M, P	R(3)	Q(1)	*
b. Iodine Sampler GH- ~~Activity Monitor (RE-10~~)	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler ~~Activity Monitor (RE-10)~~	W	N.A.	N.A.	N.A.	*
d. Flow Rate ~~Monitor~~	~~H~~ N.A.	N.A.	R(7)	~~Q~~ N.A.	*
e. Sampler Flow Rate Monitor	D	N.A.	R	~~Q~~ N.A.	*

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TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur ~~if~~ as appropriate any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or

b. Circuit failure (alarm only), or

c. Instrument indicates a downscale failure (alarm only) or

d. Istrument controls not set in operate mode (alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be ~~performed~~ using one or more of the reference (gas or liquid and solid) standards certified ~~by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement~~ assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy ~~and~~ measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, ~~sources that~~

See insert >

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

(6) See attached note (6)

(7) See attached note (7)

Insert

NBS traceable standard (gas, liquid, or solid) may be used; or a sample of the process stream (gas, liquid, or solid), provided the activity of the stream is determined using plant equipment calibrated using NBS reference standards or standards obtained from suppliers that participate in measurement assurance activities with NBS.

Table 4.3-9, Table Notations

Justification -

The proposed wording currently in the Callaway Technical Specifications implies that the same geometry as found in the monitors will be used for calibrations. It is not always possible to duplicate geometry on count room equipment. KG&E's calibration program may also differ from other plants; therefore, a time frame is not justified. Note (3) has been modified to resolve these concerns.

TABLE NOTATIONS

(6) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

   a. 10 ppm by volume oxygen, balance nitrogen, and

   b. 80 ppm by volume oxygen, balance nitrogen.

(7) If flow rate is determined by exhaust fan status and fan performance curves, the following surveillance operations shall be performed at least once per 18 months:

   a. The specific vent flows by direct measurement, or

   b. The differential pressure across the exhaust fan and vent flow established by the fan's "flow-ΔP" curve, or

   c. The fan motor horsepower measured and vent flow established by the fan's "flow-horsepower" curve.

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3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

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

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lines or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:

1) Four high pressure turbine stop valves,

~~2) Four high pressure turbine governor valves~~,

2 ~~3~~) ~~Four~~ Six low pressure turbine reheat stop valves, and

3 ~~4~~) ~~Four~~ Six low pressure turbine reheat intercept valves.

See insert

c ~~b~~. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position;

d ~~c~~. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems; and

e ~~d~~. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

---

* Not applicable in MODE 2 or 3 with all main steam line isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.

insert

4.3.4.2b At least once per 31 days by cycling each of the four high pressure Main Turbine Governor Valves through at least one complete cycle from the running position.

Specification 4.3.4.2b

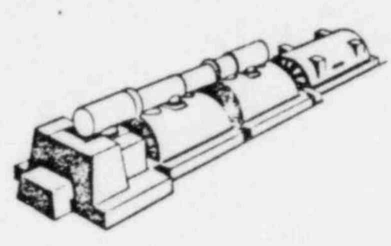
Justification -

This revision to the surveillance requirements is based on the vendor recommendations. These recommendations are provided in the attached General Electric Technical Information Letter No. 969, dated 5/22/84.

GENERAL ELECTRIC

LARGE STEAM TURBINE-GENERATOR DIVISION

GENERAL ELECTRIC COMPANY SCHENECTADY, NEW YORK



Technical Information Letter No. 969

MAY 22, 1984

Periodic Turbine Steam Valve Test - Nuclear Units

GE recommendations for periodic nuclear turbine steam valve tests as contained in the Turbine Instruction Book call for daily test of the main stop, intermediate stop, and intercept valves and weekly test of the control valves.

These recommendations are similar to the test frequencies that have been in practice since the late 40's and early 50's on fossil-fueled turbines.

The operating experience accumulated on in-service nuclear units during the past 24 years has shown considerably lower valve failure rates than those values upon which the recommendations were based. These reduced failure rates are due to many design improvements to the nuclear turbine valves and controls that have been incorporated through Technical Information Letters (TIL's) and Engineering Change Notices (ECN's).

In the "Memo Report - Hypothetical Turbine Missiles - Probability of Occurrence," dated March 14, 1973, the probability of runaway failure and wheel burst of a GE nuclear turbine was given, based on the nuclear experience up to that time. Included in the probability calculations were the recommended valve test intervals; i.e., daily for main stop and intermediate stop & intercept valves, weekly for control valves. The Nuclear Wheel Information Letter No. 2, dated November 8, 1982, gave comparative values for the increased overspeed probabilities due to increasing test intervals.

Based on past in-service experience with nuclear turbine steam valves, turbine steam inlet valve reliability and testing intervals are no longer the major contributing factors in determining hypothetical turbine missiles. The overall probability of a hypothetical missile is therefore increased only a negligible amount by increasing the test interval of the valves. Increasing test intervals will correspondingly decrease the probability of a system upset during such testing and should therefore increase the nuclear plant availability. Of course any problems detected during any testing should be brought to the attention of your local A&ES service engineer. The service engineer may call on the LST-G Dept. if further assistance is necessary.

Effective with this Technical Information Letter, the recommended valve test intervals for nuclear turbines are:

Main Stop Valves	Weekly
Intermediate Stop Valves	Weekly
Intercept Valves	Weekly
Control Valves	Monthly

Recommended test intervals for other control components remain unchanged.

Utilities should revise their Instruction Books to the new test intervals based on this TIL. Revised Instruction Book Articles will not be sent from General Electric Co.

Because of the higher temperature and resulting increased oxidation build-up on the stems and bushings of fossil-fueled turbines, the valve test interval recommendations remain unchanged; i.e., daily test of the main stop, combined intercept and reheat stop valves, and weekly test of the control valves.

If your Technical Specification or other documentation upon which your NRC operating license is based contains any obligation or commitment to test on a specific schedule, it is suggested that you take appropriate steps to modify that document if you wish to change your test intervals to these new recommendations.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within ~~1~~ 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

---

*See Special Test Exception Specification 3.10.4.

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least ~~two~~ three of the reactor coolant loops listed below shall be OPERABLE and at least ~~one~~ two of these reactor coolant loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. See attached ACTION b

c ~~b~~. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side wide range water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.2.3 At least ~~one~~ two reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

---

*All Reactor Coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** See Special Test Exception Specification 3.10.4.

ACTION:

b. With only one reactor coolant loop in operation, restore at least two loops to operation within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump,**

e. RHR Loop A, and

f. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant and/or RHR loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no reactor coolant or RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour ~~1 hour~~ provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pump, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side wide range water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or RHR loop shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10% of the wide range.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

##A reactor coolant pump shall not be started unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE# and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

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3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

---

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

---

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with at least two groups of backup pressurizer heaters each having a capacity of at least 150 kW and a water level of less than or equal to 92% (1657 cubic feet).

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of backup pressurizer heaters inoperable, restore at least two groups of backup heaters to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) ~~and remove power from the block valve(s)~~; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. See attached ACTION b.

c. See attached ACTION c.

d ~~b~~. With one or more block valve(s) inoperable, within 1 hour ~~either~~ (1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s) or close the PORV and remove power from its associated solenoid valve; ~~otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~ and (2) apply ACTION b. or c. above, as appropriate, for the isolated PORV(s).

e. ~~c~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION ~~a.~~ b. or c. in Specification 3.4.4.

---

* ~~Reactor coolant temperature above [illegible]°F~~
With all RCS cold leg temperatures above 363 °F.

ACTION:

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status, or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

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3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

    1) Reactor-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

    2) A seismic occurrence greater than the Operating Basis Earthquake, or

    3) A ~~Condition IV~~ loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

    4) A ~~Condition IV~~ main steam line or feedwater line break.

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4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to ~~53%~~ 48% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3, ~~and require prompt notification of the Commission~~ shall be reported pursuant to Specification 6.9.~~1~~2 prior to resumption of plant operation. ~~The written followup of~~ this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

a Special Report to the Commission

within 30 days and

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N. A.	N. A.	N. A.	N. A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N. A.	N. A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N. A.	N. A.
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N. A.	N. A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Particulate Radioactivity Monitoring System,

b. The Containment Normal Sump Level Measurement System, and

c. Either the Containment Air Cooler Condensate Flow Rate or the Containment Atmosphere Gaseous Radioactivity Monitoring System.

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

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for gaseous and particulate radioactivity or a gamma isotopic analysis of the containment atmosphere is performed using the Post Accident Sampling System at least once per 24 hours when the required Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Normal Sump Level Measurement System-performance of CHANNEL CALIBRATION at least once per 18 months, and

c. Containment Air Cooler Condensate Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 8 gpm per RC pump CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, reduce the leakage rate to within limits within 4 hours, or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours with an RCS pressure of less than 600 psig.

---

* See attached asterisked note

---

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

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SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous or particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the containment normal sump inventory and discharge at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE ~~to~~ from the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days, ~~with the modulating valve fully open.~~ The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the ~~plant~~ unit has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, except for valves BGV8702 A/B and EJV8701 A/B.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
BBV8948 A, B, C, D	SI/RHR/Accum. Injection Cold Leg
BBV8949 A, B, C, D	SI/RHR Hot Leg Injection
BBV001, 022, 040, 059	Bit ~~Disch To~~ Cold Leg Injection
H BBPV8702 A, B	RHR Normal Suction
EJV8841 A, B	RHR Hot Leg Recirc Ctmt Iso
EJV8701 A, B	RHR Normal Suction
EMV001, 002, 003, 004	SI Hot Leg Inj Ctmt Iso-~~EM01~~
EM8815	Bit Inj. Ctmt Isolation-~~EM02~~
EPV010, 020, 030, 040	Cold Leg SI ~~accm~~ Inj Ctmt Iso
EPV8818, A, B, C, D	Cold Leg RHR ~~Accum~~ Inj Ctmt Iso
EPV8956 A, B, C, D	Accum Inj Isolation

3/4.4.7 CHEMISTRY

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LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours~~, and~~.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
Chloride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
Fluoride	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

---

*Limit not applicable with T_{avg} less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

---

*Not required with T_{avg} less than or equal to 250°F

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3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. The provisions of Specification 3.0.4 are not applicable;

b. With the total cumulative operating time at a reactor coolant specific activity greater than 1 microCurie per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission within 30 days, pursuant to Specification 6.9.2~~,~~ indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable;

c. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

d. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram of gross radioactivity, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours~~; and~~.

---

*With T_{avg} greater than or equal to 500°F.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/Ē microCuries per gram of gross radioactivity, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

~~For this ACTION statement~~, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Results of a. the last isotopic analysis for radioiodine performed prior to exceeding the limit, b. analysis while limit was exceeded and c. one analysis after the radioiodine ~~activity~~ was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentrations,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the reactor ~~primary~~ coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

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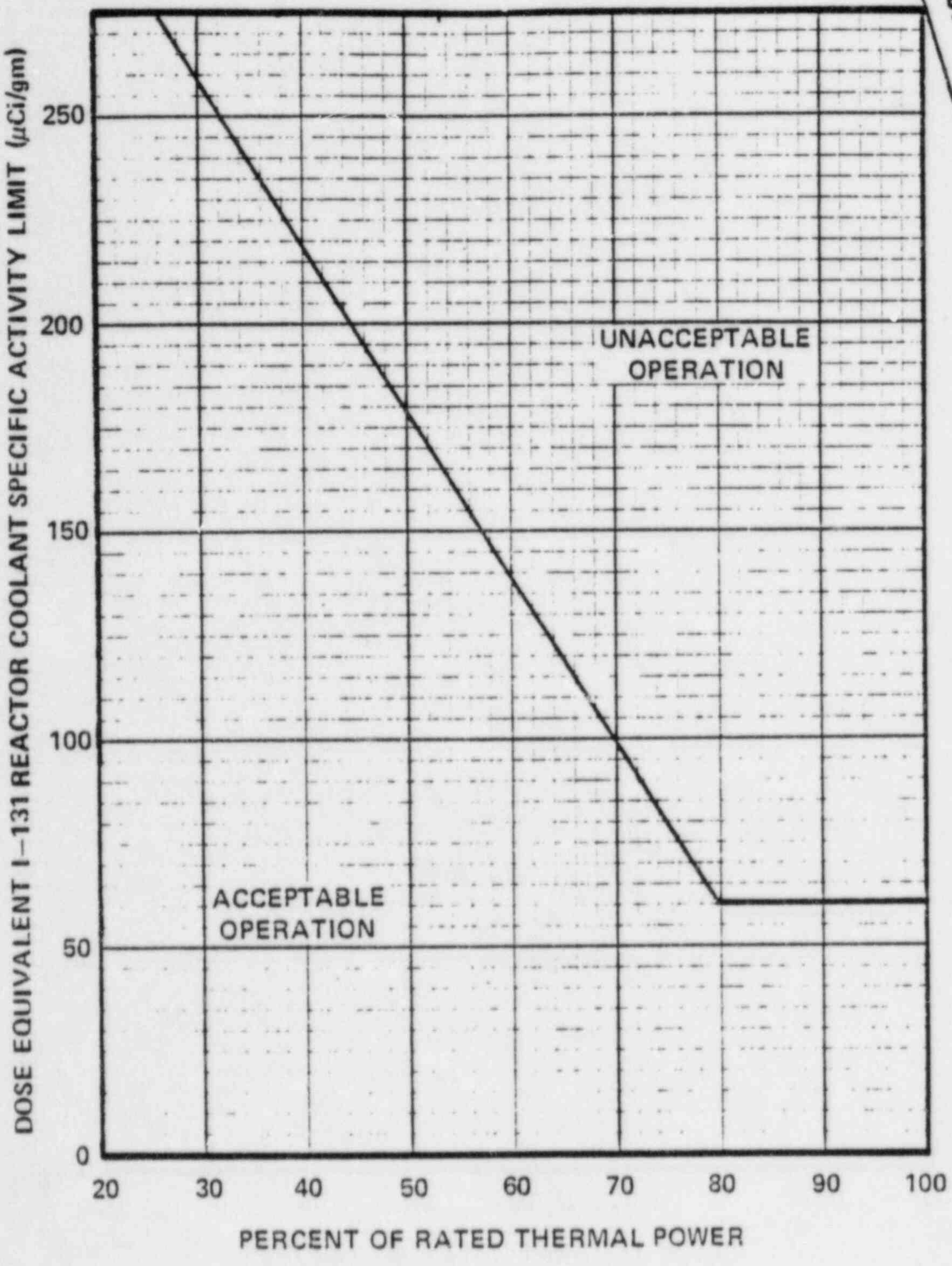

FIGURE 3.4.1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 ~~Once~~ per 14 days	1
3. Radiochemical for Ē Determination	1 ~~Once~~ per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131 or 100/Ē µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

#Until the specific activity of the Reactor Coolant System is restored within its limits

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, ~~and~~ 3.4-3, and 3.4-4.

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MATERIAL PROPERTY BASIS

COPPER CONTENT : ~~CONSERVATIVELY~~ ASSUMED TO BE 0.10 WT%

RT_{NDT} INITIAL : 40°F

RT_{NDT} AFTER 16 EFPY : 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR AND 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

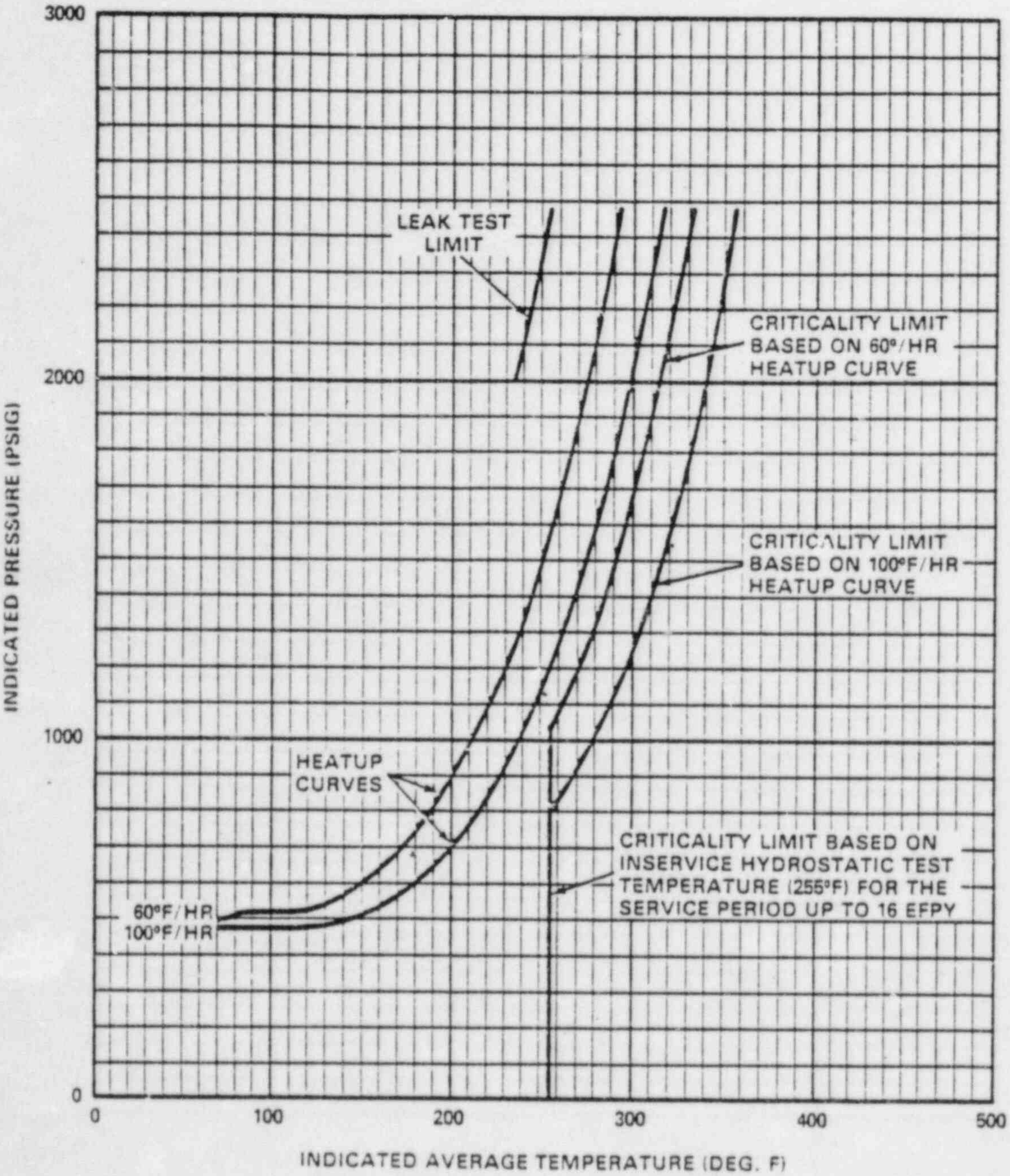

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
APPLICABLE UP TO 16 EFPY

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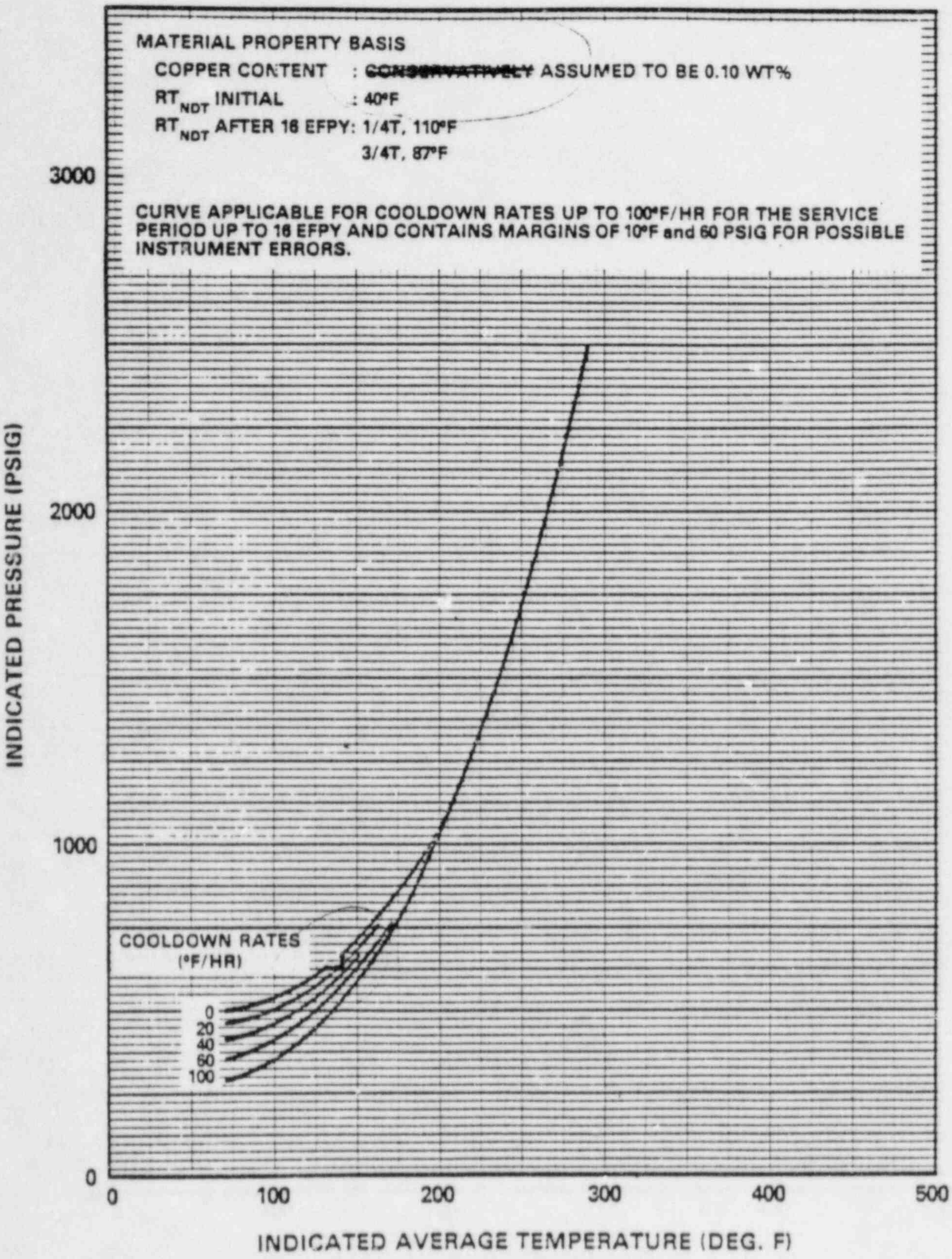

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
APPLICABLE UP TO 16 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5°	4.00	1st Refueling
Y	241°	3.69	~~8~~ 5
V	61°	3.69	9
X	238.5°	4.00	15
W	121.5°	4.00	Standby
Z	301.5°	4.00	Standby

Table 4.4-5

Justification -

Revised to be consistent with WCAP-10015 "Kansas Gas & Electric Company, Wolf Creek Generating Station, Unit No. 1, Reactor Vessel Radiation Surveillance Program."

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PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 583°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

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OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig (± 3%), or

b. ~~a.~~ Two power-operated relief valves (PORVs) with Setpoints which do not exceed the limit established in Figure 3.4-4, or

c. ~~b.~~ The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With only one PORV ~~inoperable~~ OPERABLE or with only one RHR suction relief valve OPERABLE, either restore ~~the inoperable PORV~~ the PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs or both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event ~~either~~ the PORVs or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

Specification 3.4.9.3a

Justification -

The existing tolerances were chosen based on the tolerances for the pressurizer code safety valves. They should have been established to be consistent with the requirements of ASME B & PV Code as long as the maximum pressure allowed is less than 530 psig (the lowest pressure setting of the PORV's). This specification has therefore been revised.

Specification 3.4.9.3 (Action a)

Justification -

It was the intent of this Specification to give credit for the availability of the RHR suction relief valves for cold overpressure protection (COP). In the Standard Technical Specifications, when relying on PORV's for COP, if one PORV becomes inoperable, you have 7 days to restore it or open a 2 square inch vent. This same action should be applicable if you are relying on the RHR suction relief valves for COP and you lose one, regardless of the status of the PORV's. The action statement has therefore been revised.

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

insert >

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

---

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

insert

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve 8708B:

    1) By verifying at least once per 31 days that RHR RCS Suction Isolation Valve (RRSIV) 8701B is open with power to the valve operator removed, and

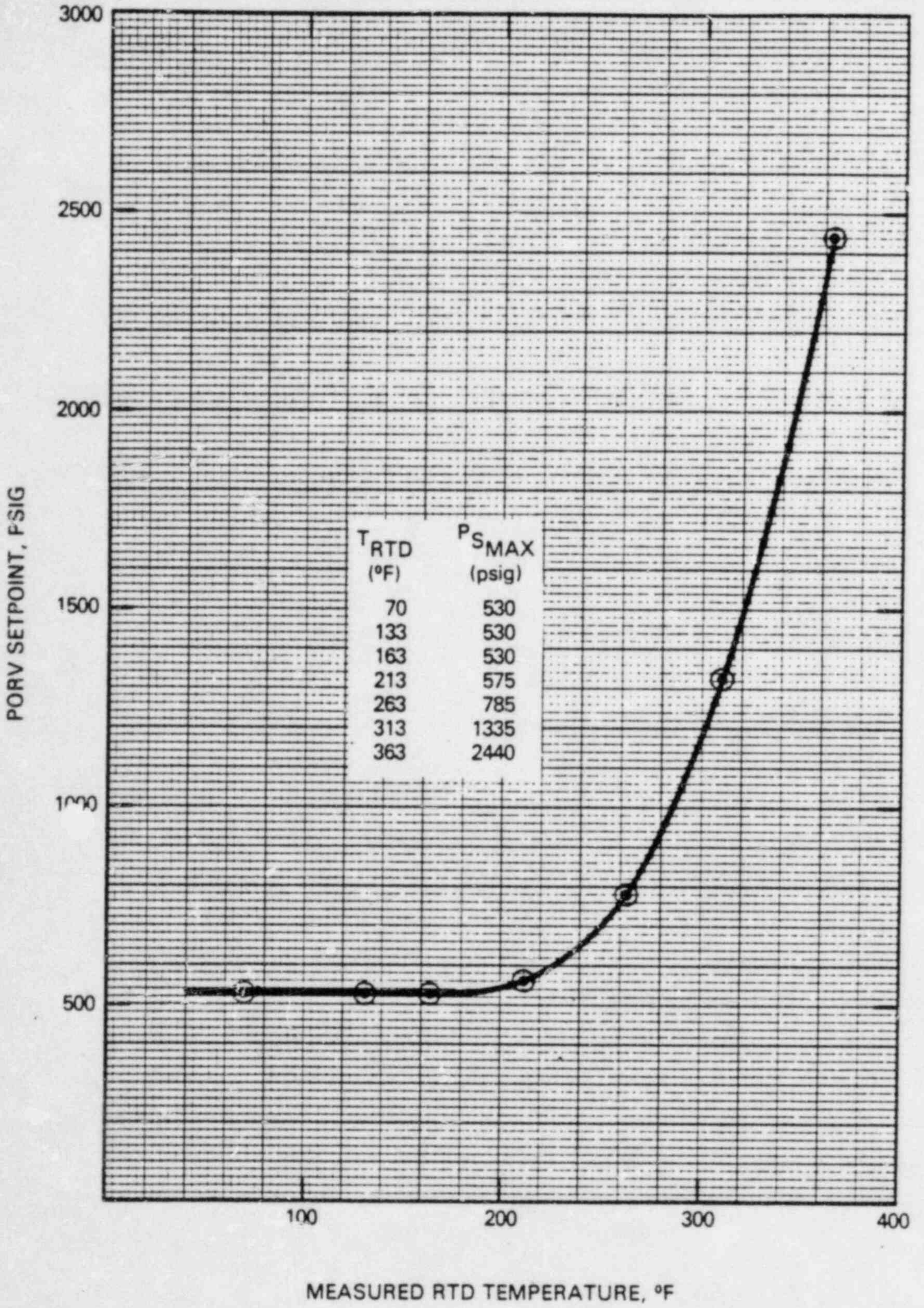
    2) By verifying at least once per 12 hours that RRSIV 8702B is open.

b. For RHR suction relief valve 8708A:

    1) By verifying at least once per 31 days that RRSIV 8702A is open with power to the valve operator removed, and

    2) By verifying at least once per 12 hours that RRSIV 8701A is open.

c. Testing pursuant to Specification 4.0.5.



T_{RTD} (°F)	$P_{S_{MAX}}$ (psig)
70	530
133	530
163	530
213	575
263	785
313	1335
363	2440

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open,* and power removed,

b. A contained borated water volume of between 6122 and 6594 gallons,

c. A boron concentration of between 1900 and 2100 ppm, and

d. A nitrogen cover-pressure of between 602 and 648 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1-hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within ~~1~~ 6-hours and in HOT SHUTDOWN within the following ~~12~~ 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

---

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution, and

c. At least once per 31 days when the RCS pressure is above ~~2000~~ 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open, ~~and~~

~~d. At least once per 18 months by verifying that each accumulator isolation valve opens automatically under each of the following conditions:~~

~~1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and~~

~~2) Upon receipt of a Safety Injection test signal.~~

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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* See attached asterisked note

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump and the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.2 provided the centrifugal charging pump and the Safety Injection pumps are restored to OPERABLE status within 4 hours ~~or~~ prior to the temperature of one or more of the RCS cold legs exceeding 375°F

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
BN-HV-8813	Safety Injection to RWST Isolation Vlv	Open
EM-HV-8802A(B)	SI Pump Discharge Hot Leg Iso Vlvs	Closed
EM-HV-8835	Safety Injection Cold Leg Iso Valve	Open
EJ-HV-8840	RHR/SI Hot Leg Recirc Iso Valve	Closed
EJ-HV-8809A	RHR to Accum Inj Loops 1 & 2 Iso Vlv	Open
EJ-HV-8809B	RHR to Accum Inj Loops 3 & 4 Iso Vlv	Open

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR System from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to 750 psig the interlocks will cause the valves to automatically close.

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2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal and/or on ~~an~~ ~~RHR~~ Automatic Switchover to Containment Sump from RWST level Low-Low Coincident with Safety Injection test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the required differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1)	Centrifugal charging pump	≥ 2400 psid,
2)	Safety Injection pump	≥ 1445 psid, and
3)	RHR pump	≥ 165 psid.

g. By verifying the correct position of each mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

HPSI System Valve Numbers		CVCS System Valve Numbers
EMV095	EMV109	BGV-198
EMV096	EMV110	BGV-199
EMV097	EMV089	BGV-200
EMV098	EMV090	BGV-201
EMV107	EMV091	BGV-202
EMV108	EMV092	

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

   1) For centrifugal charging pump lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 346 gpm, and

      b) The total pump flow rate is less than or equal to 550 gpm.

   2) For Safety Injection pump lines, with a single pump running:

      a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 462 gpm, and

      b) The total pump flow rate is less than or equal to 650 gpm.

   ~~3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 2848 gpm.~~

i. See attached item:.

i. By performing a flow test, during shutdown, following completion of modifications to the RHR subsystems that alter the subsystem flow characteristics and verifying that for RHR pump lines, with a single pump running:

1) The sum of the injection line flow rates is greater than or equal to 3800 gpm, and

2) The total pump flow rate is less than or equal to 5500 gpm.

3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} \overset{<}{\cancel{>}}$ 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the ~~refueling water storage tank~~ RWST upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the RWST ~~refueling water storage tank~~, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission within 90 days, pursuant to Specification 6.9.2, describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

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SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All centrifugal charging pumps and Safety Injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position ~~at least once per 31 days~~. within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more of the RCS cold legs decreasing below 325° and at least once per 31 days thereafter.

---

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

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3/4.5.4 ECCS SUBSYSTEMS - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.5.4 All Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

---

*An inoperable pump may be energized for testing ~~per Specification 4.0.5~~ or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 BORON INJECTION SYSTEM

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons, and

b. A boron concentration of between 2000 and 2100 ppm.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% Δk/k at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days, and

b. Verifying the boron concentration of the water in the tank at least once per 7 days.

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3/4.5.6 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.6 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 394,000 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum ~~water~~ solution temperature of 37°F, and

d. A maximum ~~water~~ solution temperature of 100°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.6 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1) Verifying the contained borated water volume in the tank, and

   2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification ~~[illegible]~~, ~~[illegible]~~ 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, 48 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

   1) Less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48 psig, or

   2) Less than or equal to L_t, ~~0.14% by weight of the containment air per 24 hours~~ at a reduced pressure of P_t, 24 psig.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, 48 psig.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than either P_a, 48 psig, or at P_t, 24 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

Specification 3.6.1 2.a 2)

Justification -

This specification has been revised to be consistent with 10CFR50, Appendix J.

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b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

   1) Confirms the accuracy of the ~~Type A~~ test by verifying that the ~~difference between~~ supplemental ~~and Type A~~ test ~~data is with~~in ~~0.25 L_a, or 0.25 L_t~~; result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or 0.25 L_t;

   2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

   3) Requires that the rate at which ~~quantity of~~ gas is injected into the containment or bled from the containment during the supplemental test ~~to be equivalent to at least 25% of the total measured leakage at P_a, 48 psig, or P_t, 24 psig.~~ is between 0.75 L_a and 1.25 L_a or between 0.75 L_t and 1.25 L_t.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48 psig, at intervals no greater than 24 months except for tests involving:

   1) Air locks and

   2) Purge supply and exhaust isolation valves with resilient material seals and

insert >

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.~~3~~2 or 4.6.1.7.4, as applicable; and

g. The provisions of Specification 4.0.2 are not applicable.

Insert

3) Valves, which are in systems that are designed to contain water subsequent to a leakage design basis loss-of-coolant accident shall be pressurized with that fluid to a pressure not less than 1.10 P_a , 52.8 psig.

Specification 4.6.1.2.d3)

Justification -

10CFR50, Appendix J, Section III.C.3 states leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate provided that a) such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the technical specifications or associated bases and b) the installed isolation valve sealwater system fluid inventory is sufficient to assure the sealing function for at least 30 days at a pressure of $1.10P_a$. The essential service water penetrations, P-28, 29, 71 and 73, qualify for this exclusion based on the fact that the water volume contained in the piping inside containment provides a passive water seal on the applicable valves. The water inventory inside containment is sufficient to assure the sealing function for at least 30 days. This specification has therefore been revised.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exits through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 48 psig.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by ~~pressurizing the volume between the door seals to at least 10 psig for at least 30 seconds and verifying the leakage does not exceed 0.01 L_a~~;

b. By conducting overall air lock leakage tests at not less than P_a, 48 psig, and verifying the overall air lock leakage rate is within its limit:

1) At least once per 6 months,# and

2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

verifying that the seal leakage is less than 0.005 L_a as
determined by precision flow measurements when measured for
at least 30 seconds with the volume between the seals
at a constant pressure of ≥ 10 psig;

#The provisions of Specification 4.0.2 are not applicable.

*This represents an exemption to Appendix J of 10 CFR Part 50.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between
+2 and -2 psig.
+1.5 -0.3

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. Containment Cooler Inlet located near NNE wall (El 2068'-8");

b. Containment Cooler Inlet located near West wall (El 2068'-8");

c. Containment Cooler Inlet located near NNW wall (El 2068'-8"); and

d. Containment Cooler Inlet located near East wall (El 2068'-8").

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CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With the structural integrity not conforming to the requirements of Specification 4.6.1.6.1, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity of the containment not conforming at a level consistent with the acceptance criteria of Specification 4.6.1.6.2, restore structural integrity or complete an engineering evaluation that assures structural integrity prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Tendons, End Anchorages and Adjacent Concrete Surfaces. The containment vessel tendons structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at a 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 11 tendons (4 inverted U and 7 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the ~~un~~observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 inverted U and 3 hoop). If more than one tendon has an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit, or with~~in~~ one tendon below 90% of the predicted lower limit, it shall be considered as evidence of possible abnormal degradation for the purposes of Specification 4.6.1.6.1g.

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b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (inverted U and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wired that:

1) The tendon wires are free of unacceptable (pitting of 1/64 inch or deeper and minimum of 1/32 inch in diameter) corrosion, cracks, and damage. The presence of unacceptable corrosion, cracks, or other damage shall be considered evidence of ~~possible~~ abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.;

2) There are no changes in the presence or physical appearance of the sheathing filler-grease. Abnormal changes in the presence or physical appearance of the sheathing filler grease shall be considered evidence of ~~possible~~ abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g; and

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) exists for at least three wire ~~or strand~~ samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire ~~or strand~~ samples to meet the minimum tensile strength test shall be considered as evidence of ~~possible~~ abnormal degradation of the containment vessel structure for the purposes of Specification 4.6.1.6.1g.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses adjuste [illegible] account for elastic losses exceed the average minimum design value given below:

Inverted U		139 ksi
Hoop:	Cylinder	147 ksi
	Dome	134 ksi

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e. Verifying the OPERABILITY of the sheathing filler grease by assuring:

1) If the installed quantity of grease exceeds that withdrawn by 5% or more, an investigation shall be conducted to assure that excessive leakage has not occurred in the tendon duct system,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

Failure to satisfy Specification 4.6.1.6.1e. 2) or 3) above for OPERABILITY of the sheathing filler grease shall be considered as evidence of ~~possible~~ abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.

f. Determining through inspection that no apparent degradation has occurred in the visual appearance of the end anchorage or the concrete surfaces adjacent to the end anchorages. If apparent degradation has occurred in the visual appearance of the end anchorage or the concrete surfaces adjacent to the end anchorages, it shall be considered as evidence of ~~possible~~ abnormal degradation of the containment structure for the purposes of Specification 4.6.1.6.1g.; and

g. If evidence of ~~possible~~ abnormal degradation of the containment structure is detected during the performance and/or evaluation of the results of the above tests, the following actions shall be completed:

1) Reported to the NRC within 10 days,

2) Perform an engineering evaluation demonstrating the continued ability of the containment structure to perform its design function. If continued containment integrity cannot be assured by engineering analysis within 90 days, ACTION a. required by Specification 3.6.1.6 shall be taken, and

3) Provide a determination of the cause of the apparent degradation and performance of any corrective actions necessary to ensure continued containment integrity.

4.6.1.6.2 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.6, restore the containment vessel to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Tendons. The containment vessel tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 11 tendons (4 inverted U and 7 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and acceptable. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 inverted U and 3 hoop);

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (inverted U and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire that:

   1) The tendon wires are free of corrosion, cracks, and damage,

   2) There are no changes in the presence or physical appearance of the sheathing filler-grease, and

   3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment vessel structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses adjusted to account for elastic losses exceed the average minimum design value given below:

Inverted U		139 ksi
Hoop:	Cylinder	147 ksi
	Dome	134 ksi

e. Verifying the OPERABILITY of the sheathing filler grease by assuring:

   1) No voids in excess of 5% of the net duct volume,

   2) Minimum grease coverage exists for the different parts of the anchorage system, and

   3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment vessel is at its maximum test pressure.

4.6.1.6.3 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

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CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 36-inch containment shutdown purge supply and exhaust isolation valve shall be closed and ~~sealed closed~~ blank flanged, and

b. The 18-inch containment mini-purge supply and exhaust isolation valve(s) may be open for up to ~~500~~ 2000 hours during a calendar year.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve open or not ~~sealed closed~~ blank flanged, close and/or ~~seal close~~ blank flange that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 18-inch containment mini-purge supply and/or exhaust isolation valve(s) open for more than ~~500~~ 2000 hours during a calendar year, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.~~8.3~~ 7.2 and/or 4.6.1.~~8~~ 7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS

4.6.1.7.1 ~~The~~ Each 36-inch containment shutdown purge supply and exhaust isolation valve(s)* shall be verified ~~sealed closed~~ blank flanged and closed at least once per 31 days.

4.6.1.7.2 See attached item 4.6.1.7.2

4.6.1.7.~~2~~3 The cumulative time that all 18-inch containment mini-purge supply and~~/or~~ exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.

~~4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS, the inboard and outboard valves with resilient material seals in sealed closed 36-inch containment shutdown purge supply and exhaust penetrations shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than 0.05 L_a when pressurized to P_a.~~

4.6.1.7.4 At least once per 3 months each 18-inch containment mini-purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than ~~0.01~~ 0.05 L_a when pressurized to P_a.

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* See attached asterisked note

4.6.1.7.2 Each 36-inch containment shutdown purge supply and exhaust isolation valve and its associated blank flange shall be leak tested at least once per 24 months and following each reinstallation of the blank flange when pressurized to P_a, 48 psig, and verifying that when the measured leakage rate for these valves and flanges, including stem leakage, is added to the leakage rates determined pursuant to Specification 4.6.1.2d for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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*Except valves and flanges which are located inside containment. These valves shall be verified to be closed with their blank flanges installed prior to entry into MODE 4 following each COLD SHUTDOWN.

CONTAINMENT SYSTEMS

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3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Containment Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

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

ACTION:

With one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Containment Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to ~~265~~ 250 psig when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment ~~Hi-3~~ Pressure - High - 3 (CSAS) test signal, and

   2) Verifying that each spray pump starts automatically on a Containment ~~Hi-3~~ Pressure - High - 3 (CSAS) test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 4340 and 4540 gallons of between 28 and 31% by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

  1) Verifying the contained solution volume in the tank, and

  2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment ~~Hi-3~~ Pressure - High - 3 (CSAS) test signal; and

d. At least once per 5 years by verifying:

  1) Each eductor flow rate is greater than or equal to 52 gpm using the RWST as the test source throttled to 17 psig at the eductor inlet, and

  2) The lines between the spray additive tank and the eductors are not blocked by verifying flow.

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CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent groups of containment cooling fans shall be OPERABLE with two fan systems to each group.

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

ACTION:

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes, and

   2) Verifying a cooling water flow rate of greater than or equal to 2200 gpm to each cooler group.

b. At least once per 18 months by verifying that on a Safety Injection test signal, the fans start in slow speed or, if operating, shift to slow speed and the cooling water flow rate increases to at least 4000 gpm to each cooler group.

3/4.6.3 CONTAINMENT ISOLATION VALVES

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LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

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

ACTION:

With one or more of the containment isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or ~~and the provisions of Specification 3.0.4 are not applicable, or~~

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, ~~and the provisions of Specification 3.0.4 are not applicable,~~ or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each containment isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Purge Isolation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (active)				
P-62	BB HV-8026**	PRT Nitrogen Iso Valve	C	10
P-62	BB HV-8027**	PRT Nitrogen Iso Valve	C	10
P-24	BG HV-8100	Seal Water Return CTMT Iso Valve	C	10
P-24	BG HV-8112	Seal Water Return CTMT Iso Valve	C	10
P-23	BG HV-8152	Letdown System CTMT Iso Valve	C	10
P-23	BG HV-8160	Letdown System CTMT Iso Valve	C	10
P-25	BL HV-80~~5~~47	Reactor Makeup Water CTMT Iso Valve	C	10
P-21	EJ HCV-8825**	RHR to SI Test Line Iso Valve	A	10
P-82	EJ HCV-8890A**	RHR A to SI Pumps Test Line Iso Valve	A	13
P-27	EJ HCV-8890B**	RHR B to SI Pumps Test Line Iso Valve	A	13
P-49	EM HV-8823**	SI/Accumulator Injection Test Line Iso Valve	A	10
P-48	EM HV-8824**	Safety Injection Pump B Test Line Iso Valve	A	10

** The provisions of Specification 3.0.4 are not applicable.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (active) - (Continued)				
P-88	EM HV-8843**	Boron Injection Upstream Test Line Iso	A	10
P-92	EM HV-8871**	SI Test Line to RWST Iso Valve	C	10
P-87	EM HV-8881**	Safety Injection Pump Test Line Iso Valve	A	10
P-92	EM HV-8964**	SI Test Line System Outside CTMT Iso	C	10
P-99	GS HV-3	Hydrogen Analyzer B Inlet Iso	A,C	5
P-99	GS HV-4	Hydrogen Analyzer B Inlet Iso	A,C	5
P-99	GS HV-5	Hydrogen Analyzer B Inlet Iso	A,C	5
P-56	GS HV-8	Hydrogen Analyzer B Disch Iso	A,C	5
P-56	GS HV-9	Hydrogen Analyzer B Disch Iso	A,C	5
P-101	GS HV-12	Hydrogen Analyzer A Inlet Iso	A,C	5
P-101	GS HV-13	Hydrogen Analyzer A Inlet Iso	A,C	5
P-101	GS HV-14	Hydrogen Analyzer A Inlet Iso	A,C	5
P-97	GS HV-17	Hydrogen Analyzer A Disch Iso	A,C	5
P-97	GS HV-18	Hydrogen Analyzer A Disch Iso	A,C	5

** The provisions of Specification 3.0.4 are not applicable.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (active) - (Continued)				
P-101	GS HV-31	Sample Line to CTMT Atmos Monitor	A,C	5
P-101	GS HV-32	Sample Line to CTMT Atmo~~n~~s Monitor	A,C	5
P-97	GS HV-33	Hydrogen Sample Return From PASS	A,C	5
P-97	GS HV-34	Hydrogen Sample Return From PASS	A,C	5
P-99	GS HV-36	Sample Line to CTMT Atmos Monitor	A,C	5
P-99	GS HV-37	Sample Line to CTMT Atmos Monitor	A,C	5
P-56	GS HV-38	Sample Return CTMT Atmos Monitor	A,C	5
P-56	GS HV-39	Sample Return CTMT Atmos Monitor	A,C	5
P-44	HB HV-7126	RCDT Vent Inside CTMT	C	10
P-26	HB HV-7136	RCDT Pumps Disch Hdr Outside CTMT Iso	C	10
P-44	HB HV-7150	RCDT Vent Outside CTMT	C	10
P-26	HB HV-7176	RCDT Pumps Disch Hrd Inside CTMT Iso	C	10
P-30	KA FV-29	Reactor Bldg Instr Air Suppl~~e~~y Outside CTMT Iso	C	5
P-32	LF FV-95	CTMT Normal Sumps to Floor Drain Tank Inside CTMT Iso	C	30

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation (active) - (Continued)				
P-32	LF FV-96	CTMT Normal Sumps to Floor Drain Tank Outside CTMT Iso	C	4
P-93	SJ HV-5**	PZR/RCS Liquid Sample Inner CTMT Iso	C	5
P-93	SJ HV-6**	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-69	SJ HV-12**	PZR Vapor Sample ~~Inner~~ Outer CTMT Iso	C	5
P-69	SJ HV-13**	PZR Vapor Sample Inner CTMT Iso	C	5
P-95	SJ HV-18**	Accumulator Sample Inner CTMT Iso	C	5
P-95	SJ HV-19**	Accumulator Sample Outer CTMT Iso	C	5
p-93	SJ HV-127**	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-64	SJ HV-128**	PZR/RCS Liquid Sample Inner CTMT Iso	A,C	5
P-64	SJ HV-129**	PZR/RCS Liquid Sample Outer CTMT Iso	A,C	5
P-64	SJ HV-130**	PZR/RCS Liquid Sample Outer CTMT Iso Valve	A,C	5
P-57	SJ HV-131**	PASS Discharge to RCDT	A,C	5
P-57	SJ HV-132**	PASS Discharge to RCDT	A,C	5
2. Phase "A" Isolation (passive)*				
P-58	EM HV-8888 **	Accumulator Tank Fill Line Iso Valve	C	~~N.A.~~ 5

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*May be opened on an intermittent basis under administrative control.

** The provisions of Specification 3.0.4 are not applicable.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
2. Phase "A" Isolation (passive)* - (Continued)				
P-16	EN HV-01**	CTMT Recirc Sump to CTMT Spray Pump A Iso	A	~~N.A.~~ 30
P-13	EN HV-07**	CTMT Recirc Sump to CTMT Spray Pump B Iso	A	~~N.A.~~ 30
P-45	EP HV-8880**	CTMT Nitrogen Supply Iso Valve	C	~~N.A.~~ 10
P-65	GS HV-20**	Hydrogen Purge Inner CTMT Iso	C	~~N.A.~~ 5
P-65	GS HV-21**	Hydrogen Purge Outer CTMT Iso	C	~~N.A.~~ 5
P-67	KC HV-253**	Fire Protection System Hdr Outer CTMT Iso	C	~~N.A.~~ 30
3. Phase "B" Isolation (active)				
P-74	EG HV-58	CCW to RCS Iso	C	30
P-75	EG HV-59	CCW Return From RCS Iso	C	30
P-75	EG HV-60	CCW Return From RCS Iso	C	30
P-76	EG HV-61	CCW Return From RCS Iso	C	30
P-76	EG HV-62	CCW Return From RCS Iso	C	30
4. Containment Purge Isolation (active)				
V-161	GT HZ-4***	CTMT Mini-Purge Supply Outside CTMT Iso	C	3
V-161	GT HZ-5***	CTMT Mini-Purge Supply Inside CTMT Iso	C	3

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*May be opened on an intermittent basis under administrative control.

**The provisions of Specification 3.0.4 are not applicable.

***The provisions of Specification 3.0.4 are not applicable provided the penetration is isolated by the passive devices.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
4. Containment Purge Isolation (active) - (Continued)				
V-160	GT HZ-11 ***	CTMT Mini-Purge Exh Inside CTMT Iso	C	3
V-160	GT HZ-12 ***	CTMT Mini-Purge Exh Outside CTMT Iso	C	3
5. Containment Purge Isolation (passive)				
V-161	GT HZ-6 ***	CTMT S/D Purge Supply Outside CTMT Iso	C	10 ~~N.A.~~
V-161	GT HZ-7 ***	CTMT S/D Purge Supply Inside CTMT Iso	C	10 ~~N.A.~~
V-160	GT HZ-8 ***	CTMT S/D Purge Exh Inside CTMT Iso	C	10 ~~N.A.~~
V-160	GT HZ-9 ***	CTMT S/D Purge Exh Outside CTMT Iso	C	10 ~~N.A.~~
6. Remote Manual				
P-41	BB HV-8351A	RCP A Seal Water Supply	C	N.A.
P-22	BB HV-8351B	RCP B Seal Water Supply	C	N.A.
P-39	BB HV-8351C	RCP C Seal Water Supply	C	N.A.
P-40	BB HV 8351D	RCP D Seal Water Supply	C	N.A.
P-79	BB PV-8702A	RCS Hot Leg 1 to RHR Pump A Suction	A	N.A.

*** The provisions of Specification 3.0.4 are not applicable provided the penetration is isolated by two passive devices

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
6. Remote Manual - (Continued)				
P-52	BB PV-8702B	RCS Hot Leg 4 to RHR Pump B Suction	A	N.A.
P-15	EJ HV-23**	PASS Sump Sample CTMT Iso	C	~~N.A.~~ 5
P-15	EJ HV-25**	PASS Sump Sample CTMT Iso	C	~~N.A.~~ 5
P-14	EJ HV-24**	PASS Sump Sample CTMT Iso	C	~~N.A.~~ 5
P-14	EJ HV-26**	PASS Sump Sample CTMT Iso	C	~~N.A.~~ 5
P-71	EF HV-31	ESW Supply To Containment Coolers	C	N.A.
P-28	EF HV-32	ESW Supply To Containment Coolers	C	N.A.
P-71	EF HV-33	ESW Supply To Containment Coolers	C	N.A.
P-28	EF HV-34	ESW Supply To Containment Coolers	C	N.A.
P-73	EF HV-45	ESW Return From Containment Coolers	C	N.A.
P-29	EF HV-46	ESW Return From Containment Coolers	C	N.A.
P-73	EF HV-47	ESW Return From Containment Coolers	C	N.A.

** The provisions of Specification 3.6.4 are not applicable.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
6. Remote Manual - (Continued)				
P-29	EF HV-48	ESW Return From Containment Coolers	C	N.A.
P-73	EF HV-49	ESW Return From Containment Coolers	C	N.A.
P-29	EF HV-50	ESW Return From Containment Coolers	C	N.A.
P-74	EG HV-127*	CCW Supply to RCP	C	N.A.
P-75	EG HV-130*	CCW Return from RCP	C	N.A.
P-75	EG HV-131*	CCW Return From RCP	C	N.A.
P-76	EG HV-132*	CCW Return From RCP Thermal Barriers	C	N.A.
P-76	EG HV-133*	CCW from RCP Thermal Barrier	C	N.A.
P-79	EJ HV-8701A	RCS Hot Leg 1 to RHR Pump A Suction	A	N.A.
P-52	EJ HV-8701B	RCS Hot Leg 4 to RHR Pump B Suction	A	N.A.
P-82	EJ HV-8809A	RHR Pump A Cold Leg Injection Iso Valve	A	N.A.
P-27	EJ HV-8809B	RHR Pump B Cold Leg Injection Iso Valve	A	N.A.
P-15	EJ HV-8811A	CTMT Recirc Sump to RHR Pump A Sucion t	A	N.A.

---

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under adminstrative control.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
6. Remote Manual - (Continued)				
P-14	EJ HV-8811B	CTMT Recirc Sump to RHR Pump B Suction	A	N.A.
P-21	EJ HV-8840	RHR Hot Leg Recirc Iso Valve	A	N.A.
P-87	EM HV-8802A*	SI Pump A Disch Hot Leg Iso Valve	A	N.A.
P-48	EM HV-8802B*	SI Pump B Disch Hot Leg Iso Valve	A	N.A.
P-49	EM HV-8835	SI Pumps Disch to Cold Leg Iso Valve	A	N.A.
P-89	EN HV-6	CTMT Spray Pump A Discharge Iso Valve	A	N.A.
P-66	EN HV-12	CTMT Spray Pump B Discharge Iso Valve	A	N.A.
7. Active for SIS				
P-80	BG HV-8105	CVCS Charging Line	C	~~N.A.~~ 10
P-88	EM HV-8801A	Boron Injection to RCS Cold Legs	A	N.A.
P-88	EM HV-8801B	Boron Injection to RCS Cold Legs	A	N.A.
8. Hand-Operated and Check Valves				
P-41	BB V-118	RCP A Seal Water Supply	C	N.A.
P-22	BB V-148	RCP B Seal Water Supply	C	N.A.
P-39	BB V-178	RCP C Seal Water Supply	C	N.A.
P-40	BB V-208	RCP D Seal Water Supply	C	N.A.

---

*These valves were assumed to be closed during the accident analysis and are normally closed but may be opened on an intermittent basis under administrative control.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-24	BG V-135	RCP Seal Water Return	C	N.A.
P-80	BG 8381	CVCS Charging Line	C	N.A.
P-25	BL 8046	Reactor Makeup Water Supply	C	N.A.
P-78	BM V-045	Steam Generator Drain Line Iso Valve	C	N.A.
P-78	BM V-046	Steam Generator Drain Line Iso Valve	C	N.A.
P-53	EC V-083	Refueling Pool Supply From Fuel Pool Cleanup	C	N.A.
P-53	EC V-084	Refueling Pool Supply From Fuel Pool Cleanup	C	N.A.
P-54	EC V-087	Refueling Pool Return to Fuel Pool Cooling	C	N.A.
P-54	EC V-088	Refueling Pool Return to Fuel Pool Cooling	C	N.A.
P-55	EC V-095	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	N.A.
P-55	EC V-096	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	N.A.
P-74	EG V-204	CCW Supply to RCP	C	N.A.
P-82	EJ 8818A	RHR Pump to Cold Leg 1 Injection	A	N.A.
P-82	EJ 8818B	RHR Pump to Cold Leg 2 Injection	A	N.A.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-27	EJ 8818C	RHR Pump to Cold Leg 3 Injection	A	N.A.
P-27	EJ 8818D	RHR Pump to Cold Leg 4 Injection	A	N.A.
P-21	EJ 8841A	RHR Pump Disch to RCS Hot Leg 2	A	N.A.
P-21	EJ 8841B	RHR Pump Disch to RCS Hot Leg 3	A	N.A.
P-87	EM V-001	SI Pump Hot Leg 1 Injection	A	N.A.
P-87	EM V-002	SI Pump Hot Leg 2 Injection	A	N.A.
P-48	EM V-003	SI Pump Hot Leg 3 Injection	A	N.A.
P-48	EM V-004	SI Pump Hot Leg 4 Injection	A	N.A.
P-58	EM V-006	Accumulator Fill Line From SI Pumps	C	N.A.
P-49	EM V-010	SI Pump Disch to Cold Leg 1	A	N.A.
P-49	EM V-020	SI Pump Disch to Cold Leg 2	A	N.A.
P-49	EM V-030	SI Pump Disch to Cold Leg 3	A	N.A.
P-49	EM V-040	SI Pump Disch to Cold Leg 4	A	N.A.
P-88	EM V-8815	BIT to RCS Cold Leg Injection	A	N.A.
P-89	EN V-013	CTMT Spray Pump A to CTMT Spray Nozzles	A	N.A.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A	N.A.
P-45	EP V-046	Accumulator Nitrogen Supply Line	C	N.A.
P-43	HD V-016	Auxiliary Steam to Decon System	C	N.A.
P-43	HD V-017	Auxiliary Steam to Decon System	C	N.A.
P-63	KA V-039	Rx Bldg Service Air Supply	C	N.A.
P-63	KA V-118	Rx Bldg Service Air Supply	C	N.A.
P-30	KA V-204	Rx Bldg Instrument Air Supply	C	N.A.
P-67	KC V-478	Fire Protection Supply to RX Bldg	C	N.A.
P-57	SJ V-111	Liquid Sample from PASS to RCDT	A,C	N.A.
P-98	KB V-001	Breathing Air Supply to RX Bldg	C	N.A.
P-98	KB V-002	Breathing Air Supply to RX Bldg	C	N.A.

insert > See attached

insert

9. Other Automatic Valves

P-1	AB-HV-11**	Mn. Stm. Isol.	A	5
P-2	AB-HV-14**	Mn. Stm. Isol.	A	5
P-3	AB-HV-17**	Mn. Stm. Isol.	A	5
P-4	AB-HV-20**	Mn. Stm. Isol.	A	5
P-5	AE-FV-42**	Mn. FW Isol.	A	5
P-6	AE-FV-39**	Mn. FW Isol.	A	5
P-7	AE-FV-40**	Mn. FW Isol.	A	5
P-8	AE-FV-41**	Mn. FW Isol.	A	5
P-9	BM-HV-4**	SG Blowdn. Isol.	A	10
P-10	BM-HV-1**	SG Blowdn. Isol.	A	10
P-11	BM-HV-2**	SG Blowdn. Isol.	A	10
P-12	BM-HV-3**	SG Blowdn. Isol.	A	10

**The provisions of Specification 3.0.4 are not applicable.

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3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one containment hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen analyzers inoperable, restore at least one analyzer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment hydrogen analyzer shall be demonstrated OPERABLE by the performance of ~~a CHANNEL CHECK at least once per 12 hours~~, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, ~~and at least once per 92 days~~ and at least once per 31 days, on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing ten volume percent hydrogen, balance nitrogen.

~~a. One volume percent hydrogen, balance nitrogen, and~~

~~b. Four volume percent hydrogen, balance nitrogen.~~

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HYDROGEN CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.2 A Hydrogen Control System shall be OPERABLE with two independent Hydrogen Recombiner Systems ~~or a Hydrogen Purge Subsystem and one independent Hydrogen Recombiner System.~~

APPLICABILITY: MODES 1 and 2

ACTION:

With one of the two independent Hydrogen Recombiner Systems ~~and the Hydrogen Purge Subsystem~~ inoperable, restore the inoperable Hydrogen Recombiner System ~~or the Hydrogen Purge Subsystem~~ to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2.1 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a Hydrogen Recombiner System functional test, that the heater air temperature increases to greater than or equal to 1150°F within 5 hours; and

b. At least once per 18 months by:

1) Performing a CHANNEL CALIBRATION of all hydrogen recombiner system instrumentation and control circuits,

2) Verifying through a visual examination that there is no evidence of abnormal conditions within the hydrogen recombiner system enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

~~4.6.4.2.2 The Hydrogen Purge Subsystem shall be demonstrated OPERABLE by cycling valves GS-HV20, GS-HV21, and KA-HV30 at least once per 31 days.~~

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With three reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (Percent of RATED THERMAL POWER)
1	87
2	65
3	44

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR *	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (Percent of RATED THERMAL POWER)
1	**
2	**
3	**

---

*At least two safety valves shall be OPERABLE on the non-operating steam generator.

**These values left blank pending NRC approval of three loop operation.

TABLE 3.7-3

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING (±1%)	ORIFICE SIZE
Loop 1	Loop 2	Loop 3	Loop 4		
V055	V065	V075	V045	1185 psig	16.0 sq. in.
V056	V066	V076	V046	1197 psig	16.0 sq. in.
V057	V067	V077	V047	1210 psig	16.0 sq. in.
V058	V068	V078	V048	1222 psig	16.0 sq. in.
V059	V069	V079	V049	1234 psig	16.0 sq. in.

---

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1535 psig on recirculation flow when tested pursuant to Specification 4.0.5;

   2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1625 psig at a flow of greater than or equal to 120 gpm when the secondary steam supply pressure is greater than 900 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the ESW supply to the auxiliary feedwater pumps actuates to its full open position upon receipt of an Auxiliary Feedwater Pump Suction Pressure-Low test signal,

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal, and

3) Verifying that each auxiliary feedwater motor-operated discharge valve limits the flow to each steam generator from the motor-driven pump to less than or equal to 320 gpm.

4.7.1.2.2 An auxiliary feedwater flow path shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to at least two steam generators from one auxiliary feedwater pump.

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained water volume of at least ~~212,700~~ 231,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the CST inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the Essential Service Water (ESW) System as a backup supply to the auxiliary feedwater pumps and restore the CST to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The ESW System shall be demonstrated OPERABLE at least once per 12 hours by verifying that the ESW System is in operation whenever the ESW System is the supply source for the auxiliary feedwater pumps.

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SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSLIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSLIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3:

With one MSLIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise. be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSLIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generator shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

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

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position. In addition, an ANALOG CHANNEL OPERATIONAL TEST of the surge tank level and flow instrumentation which provide automatic isolation of the non-nuclear safety-related portion of the system shall be performed at least once per 31 days;

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment or isolating the non-nuclear safety-related portion of the system actuates to its correct position on a Loss-of-Power or Safety Injection ~~test signal~~ and on a simulated High Flow and Low Surge Tank Level test signal, and

   2) Each OPERABLE Component Cooling Water System pump starts automatically on a Safety Injection ~~or~~ and Loss-of-Power test signal.

c. At least once per 18 months during shutdown, by performing a CHANNEL CALIBRATION of the surge tank level and flow instrumentation which provide automatic isolation of the non-nuclear safety-related portion of the system.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent essential service water (ESW) loops shall be OPERABLE.

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

ACTION:

With only one ESW loop OPERABLE, restore at least two ESW loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two ESW loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position. In addition, at least once per 31 days, an ANALOG CHANNEL OPERATIONAL TEST of the differential pressure instrumentation for automatic isolation of the ESW to the air compressors shall be performed;

b. At least once per 18 months during shutdown, by verifying that:

   1) Each automatic valve servicing safety-related equipment or isolating the non-nuclear safety-related portion of the system actuates to its correct position on a Loss-of-Power or Safety Injection test signal and on a simulated High Differential Pressure test signal, and

   2) Each Essential Service Water System pump starts automatically on a Safety Injection, Low Suction Pressure (AFW pumps), and Loss-of-Power test signal.

c. At least once per 18 months during shutdown, by performing a CHANNEL CALIBRATION of the differential pressure instrumentation for automatic isolation of ESW to the air compressors.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink (UHS) shall be OPERABLE with:

a. The crest of the UHS dam without the rip rap cover on top and corresponding water level at or above elevation 1069.5 Mean Sea Level, USGS datum, and

b. The ~~lake outlet (~~plant inlet~~)~~ water temperature of less than or equal to 90°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The UHS shall be determined OPERABLE:

a. At least once per 24 hours by verifying the above required water temperature and water level to be within their limits, and

b. At least once per 12 months by verifying that the crest of the UHS dam without the rip rap cover on top is above elevation 1069.5 Mean Sea Level, USGS datum.

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3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent Control Room Emergency Ventilation Systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Emergency Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one Control Room Emergency Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation System in the recirculation mode.

b. With both Control Room Emergency Ventilation Systems inoperable, or with the OPERABLE Control Room Emergency Ventilation System, required to be in the recirculation mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.6 Each Control Room Emergency Ventilation System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 84°F;

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers of both the Filtration and Pressurization Systems and verifying that the Pressurization System operates for at least 10 continuous hours with the heaters operating;

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SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the ~~cleanup~~ Control Room Emergency Ventilation system satisfies the in-place penetration and bypass leakage testing acceptance criteria; of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ± 10% for the Filtration System and 2000 cfm ± 10% for the Pressurization System with 500 cfm ± 10% going through the Pressurization System filter adsorber unit;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 2000 cfm ± 10% for the Filtration System and 2000 cfm ± 10% for the Pressurization System with 500 cfm ± 10% going through the Pressurization System filter adsorber unit during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

e. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.4 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10% for the Filtration System and 500 cfm ± 10% for the Pressurization System filter adsorber unit,

2) ~~Verifying that on a Control Room Ventilation Isolation~~ test signal ~~on High Smoke [illegible] Signal~~, ~~the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.~~

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/4 inch Water Gauge relative to the outside atmosphere during system operation,

4) Verifying that the Pressurization System filter adsorber unit heaters dissipate 15 ± 2 kW in the Pressurization System when tested in accordance with ANSI N510-1975, and

5) Verifying that on a High Chlorine ~~and Toxic Gas~~ test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks within 15 seconds.

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing ~~acceptance~~ criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 2000 cfm ± 10% for the Filtration System and 500 cfm ± 10% for the Pressurization System filter adsorber unit; and

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing ~~acceptance~~ criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 2000 cfm ± 10% for the Filtration System and 500 cfm ± 10% for the Pressurization System filter adsorber unit.

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3/4.7.7 EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Emergency Exhaust Systems shall be OPERABLE.

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

ACTION:

With one Emergency Exhaust System inoperable, restore the inoperable Emergency Exhaust system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each Emergency Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorters and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

   1) Verifying that ~~with~~ the Emergency Exhaust system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 9000 cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

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SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than 5.4 inches Water Gauge while operating the system at a flow rate of 9000 cfm ± 10%,

2) Verifying that the system maintains the Fuel Building at a negative pressure of greater than or equal to ¼ inch ~~in~~ Water Gauge relative to the outside atmosphere during system operation,

3) Verifying that the system starts on a Safety Injection test signal, and

4) Verifying that the heaters dissipate 37 ± 3 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing ~~acceptance~~ criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing ac~~ceptan~~ce criteria of less than 1% ~~remove greater than~~ in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ± 10%.

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3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All ~~hydraulic and mechanical~~ snubbers shall be OPERABLE. The only snubbers excluded from the requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program, in lieu of the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATON and shall include all hydraulic and mechanical snubbers. If all snubbers of each type on any system are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that system shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type on Any System per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

---

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

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c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; or (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8f. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.

d. Transient Event Inspections

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.8f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of ~~all~~ each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the

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e. Functional Tests (Continued)

total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.8f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's, testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on the Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that ~~group~~ type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that ~~type~~ group may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that ~~type~~ group have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2 where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning ~~the~~ testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating envi~~m~~ronments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

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f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed rate or release rate where required ~~for hydraulic snubbers~~ is present in both tension and compression, within the specified range; and

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Service Life Monitoring Program ~~Functional Test Failure Analysis~~

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement

SURVEILLANCE REQUIREMENTS (Continued)

snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for varous seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

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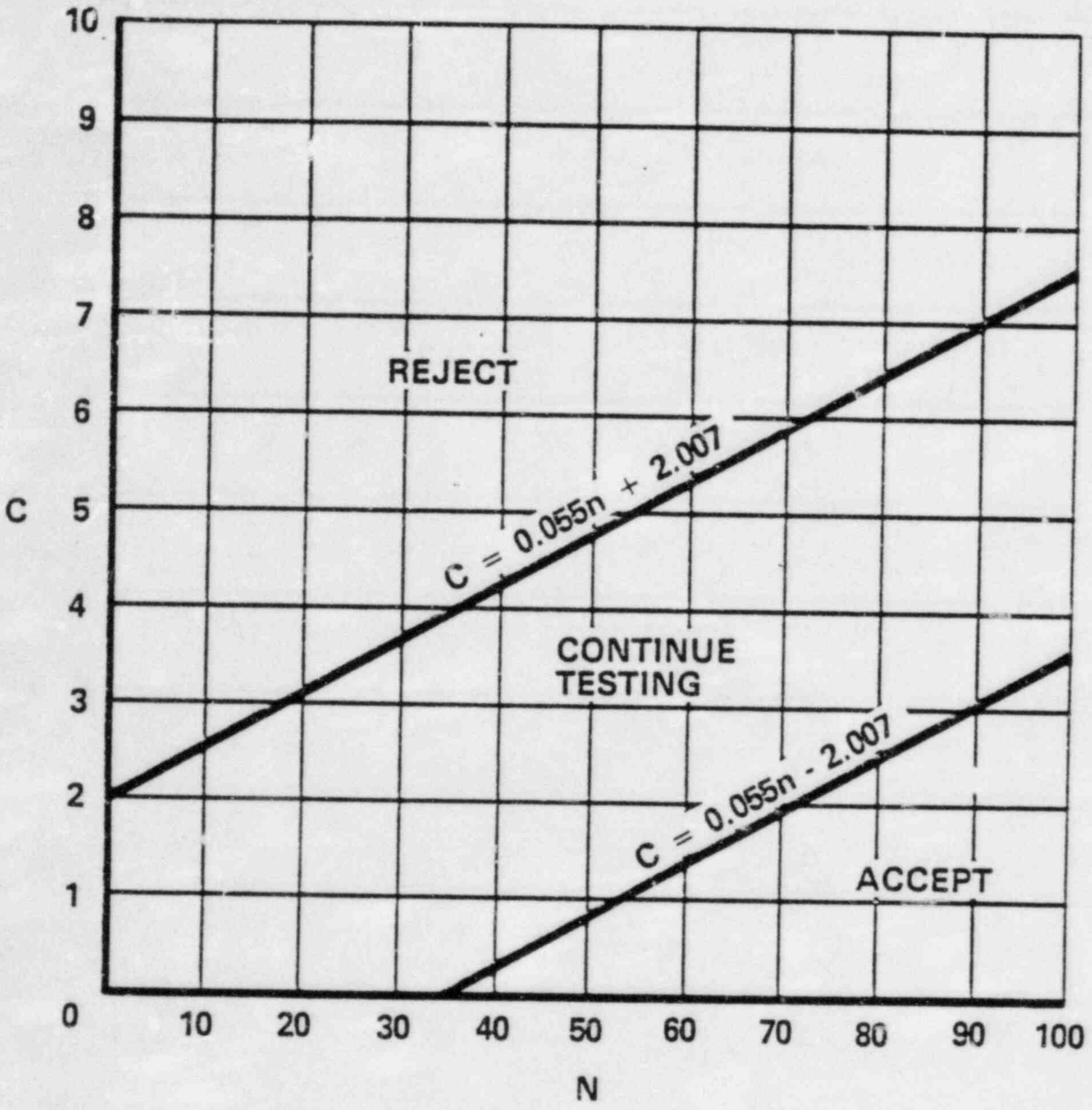

FIGURE 4.7-1

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

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3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma-emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

    1. Decontaminate and repair the sealed source, or

    2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

    1) With a half-life greater than 30 days (excluding Hydrogen 3), and

    2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

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3/4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10.1 The Fire Suppression Water System shall be OPERABLE with:

a. At least two fire suppression pumps, each with a capacity of ≥ ~~3000~~ 3000 gpm, with their discharge aligned to the fire suppression header, and

b. An OPERABLE flow path capable of taking suction from the Wolf Creek Generating Station cooling lake and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.10.2 and 3.7.10.4.

APPLICABILITY: At all times.

ACTION:

a. With one pump ~~and/or one water supply~~ inoperable, restore the inoperable equipment to OPERABLE status within 7 days. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that lake level exceeds 1075 feet,

b. At least once per 31 days on a STAGGERED TEST BASIS by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position, Specification 3.7.10.1.a

Justification -

The Technical Specification capacity of the fire suppression pumps has been revised to allow for pump degradation. The revised value is the rated capacity of the pump. This rated capacity still exceeds the design requirements for the system.

Specification 3.7.10.1 (Action a)

Justification -

The words "and/or one water supply" have been deleted from Action a. This was done because the Wolf Creek fire suppression water system has only one water supply (the Wolf Creek cooling lake).

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d. At least once per 6 months by performance of a yard loop and fire hydrant flush,

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

~~1) Verifying that each automatic valve in the flow path actuates to its correct position,~~

1 ~~2)~~ Verifying that each pump develops at least 3000 ~~3300~~ gpm at a system ~~head of 277 feet,~~ pressure of 125 psi,

2 ~~3)~~ Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3 ~~4)~~ Verifying that each fire suppression pump starts (sequentially) on decreasing pressure in the fire suppression header at a header pressure greater than or equal to 80 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank contains at least 200 ~~175~~ gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

Specification 4.7.10.1.1.f.1

Justification -

The Technical Specification capacity and pressure of the fire suppression pumps has been revised to allow for pump degradation. The revised values are the rated capacity and pressure of the pumps. The values still exceed the design requirements of the system.

Specification 4.7.10.1.2.a 1)

Justification -

The volume of fuel in the fuel storage tank has been revised to account for the 5% sump of unusable fuel in the tank.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.10.1.3 Each fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1) The electrolyte level of each battery is above the plates, and

   2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months, by verifying that:

   1) The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2) The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

SPRAY AND/OR SPRINKLER SYSTEMS

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LIMITING CONDITION FOR OPERATION

3.7.10.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Wet Pipe Sprinkler Systems

Building	Elevation	Area Protected
Auxiliary	2000/2026	North Electric Cable Chase
Auxiliary	1988/2000/2026	South Electric Cable Chase
Control	1974 - 2073	Vertical Electrical Chases
Control	1974	Pipe Space and Tank Room
Control	1992	Cable Area Above Access Control *

b. Pre-Action Sprinkler Systems

Building	Elevation	Area Protected
Auxiliary	1974	Cable Trays *
Auxiliary	2000	Cable Trays *
Auxiliary	2026	Cable Trays *
Control	2032	Lower Cable Spreading Room
Control	2073	Upper Cable Penetration Area
Reactor	2026	North Cable Penetration Area
Reactor	2026	South Cable Penetration Area
Diesel Gen. (E)	2000	East Diesel Generator Room
Diesel Gen. (W)	2000	West Diesel Generator Room

c. Water Sprays Systems

Building	Elevation	Area Protected
Auxiliary	2000	Auxiliary Feedwater Pump Turbine
ESF Transformer	Grade	Transformer XNB01 *
ESF Transformer	Grade	Transformer XNB02 *

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

* Areas contain redundant systems or components which could be damaged.

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SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

    1) By performing a system functional test which includes simulated automatic actuation of the system, and:

        a) Verifying that the automatic valves in the flow path actuate to their correct positions on a Simulated Fire test signal, and

        b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

    2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

    3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air or water flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

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PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The following Halon Systems shall be OPERABLE:

Building	Elevation	Area Protected
Auxiliary	2026	North Electrical Penetration Room
Auxiliary	2026	South Electrical Penetration Room
Auxiliary	2026	Load Center and M. G. Sets Room *
Control	2000	ESF Switchgear Rooms *
Control	2016	Switchgear Rooms
Control	2047	Control Room Cable Trenches and Chases

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.3 Each of the above required Halon Systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct ~~position.~~

a ~~b.~~ At least once per 6 months by verifying Halon storage tank weight (or level) to be at least 95% of full charge weight and pressure to be at least 90% of full charge pressure, and

b ~~c.~~ At least once per 18 months by~~:~~

~~1)~~ Verifying the system, including associated Ventilation Systems fire dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, ~~and~~

~~2) Performance of a flow test through headers and nozzles to assure no blockage.~~

* Areas contain redundant systems or components which could be damaged.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.4 The fire hose stations given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, provide equivalent capacity backup hose protection to the unprotected area from the spare hose connection on the adjacent OPERABLE standpipe. If two standpipe hose connections are not available at the adjacent OPERABLE hose station(s), provide gated wye(s) to ensure continued OPERABILITY of the affected hose station. Where it can be demonstrated that the physical routing of the backup hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, or would require the blocking open of a fire door, the hose shall be stored at the point of origin and properly identified as to its intended use. The above action shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.4 Each of the fire hose stations given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station,

b. At least once per 18 months, by:

   1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2) Removing the hose for inspection and reracking, and

   3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

   1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

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TABLE 3.7-5

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Auxiliary	1974	1122	KC-HR-051
Auxiliary	1974	1122	KC-HR-047
Auxiliary	1974	1120	KC-HR-031
Auxiliary	1974	1120	KC-HR-025#
Auxiliary	1974	1101	KC-HR-023#
Auxiliary	1974	1101	KC-HR-040
Auxiliary	1974	1101	KC-HR-042
Auxiliary	1988	1201	KC-HR-024
Auxiliary	2000	1329	KC-HR-111
Auxiliary	2000	1320	KC-HR-048
Auxiliary	2000	1320	KC-HR-046#
Auxiliary	2000	1314	KC-HR-030
Auxiliary	2000	1321	KC-HR-029#
Auxiliary	2000	1301	KC-HR-035#
Auxiliary	2000	1301	KC-HR-039
Auxiliary	2000	1301	KC-HR-041#
~~Turbine~~	~~2000~~	~~4322~~	~~KC-HR-098~~
Auxiliary	2026	1408	KC-HR-049
Auxiliary	2026	1408	KC-HR-044
Auxiliary	2026	1408	KC-HR-032#
Auxiliary	2026	1408	KC-HR-026#
Auxiliary	2026	1401	KC-HR-034
Auxiliary	2026	1403	KC-HR-037#
Auxiliary	2047	1506	KC-HR-050
Auxiliary	2047	1513	KC-HR-043
Auxiliary	2047	1506	KC-HR-045
Auxiliary	2047	1501	KC-HR-038
Auxiliary	2047	1504	KC-HR-033
Auxiliary	2047	1502	KC-HR-027
Auxiliary	2064	1119	KC-HR-028#
Control	1974	3101	KC-HR-002#
Control	1974	3101	KC-HR-014#
Control	1984	3204	KC-HR-015#
Control	1984	3221	KC-HR-001#
Control	2000	3301	KC-HR-004#
Control	2000	3301	KC-HR-017#
Control	2000	3302	KC-HR-016#
Control	2016	3401	KC-HR-005
Control	2016	3401	KC-HR-019
Control	2016	3401	KC-HR-018

TABLE 3.7-4 (Continued)

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Control	2032	3501	KC-HR-006#
Control	2032	3501	KC-HR-020#
Control	2047	3604	KC-HR-007
Control	2047	3616	KC-HR-021
Control	2073	3801	KC-HR-008#
Control	2073	3801	KC-HR-022#
Reactor	2000	2201	KC-HR-120*
Reactor	2000	2201	KC-HR-131*
Reactor	2000	2201	KC-HR-124*
Reactor	2000	2201	KC-HR-129*
Reactor	2026	N.A.	KC-HR-121*
Reactor	2026	N.A.	KC-HR-132*#
Reactor	2026	N.A.	KC-HR-125*
Reactor	2026	N.A.	KC-HR-130*
Reactor	2047	N.A.	KC-HR-128*
Reactor	2047	N.A.	KC-HR-122*
Reactor	2047	N.A.	KC-HR-126*
Reactor	2068	N.A.	KC-HR-123*
Reactor	2068	N.A.	KC-HR-127*
Fuel	2000	6102	KC-HR-142#
Fuel	2000	6102	KC-HR-054#
Fuel	2000	6102	KC-HR-143
Fuel	2000	6104	KC-HR-057
Fuel	2026	6201	KC-HR-133
Fuel	2026	6203	KC-HR-052
Fuel	2047	6301	KC-HR-055#
Fuel	2047	6302	KC-HR-056#
Fuel	2047	6301	KC-HR-053#
ESW	2000	N.A.	KC-HR-140
ESW	2000	N.A.	KC-HR-141

TABLE NOTATIONS

#Secondary means of fire suppression to Water Sprays/Deluge or Halon System.

*Fire hose for station to be stored external to Reactor Building.

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3/4.7.11 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.11 All fire barrier penetrations (i~~ncluding~~ walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire-rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations ~~rated assemblies and/or sealing devices~~ inoperable, within 1 hour establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the inoperable fire barrier ~~assembly~~ and establish an hourly fire watch patrol. ~~or install a temporary noncombustible sealant material of at least an equal fire resistance rating to the inoperable assembly and sealing device and establish an hourly fire watch patrol.~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.1 At least once per 18 months the above required fire rated assemblies ~~barrier penetrations~~ and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type (electrical and mechanical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the Fire Door Supervision System for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door without electrical supervision is closed at least once per 7 days,

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test at least once per 18 months, and

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

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3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature limit of each area given ~~shown~~ in Table 3.7-6 shall not be exceeded for more ~~maintained within the limits indicated in Table 3.7-6.~~ than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above, and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	MAXIMUM TEMPERATURE LIMIT (°F)
1. ESW Pump Room A	119
2. ESW Pump Room B	119
3. Auxiliary Feedwater Pump Room A	119
4. Auxiliary Feedwater Pump Room B	119
5. Turbine Driven Auxiliary Feedwater Pump Room	147
6. ESF Switchgear Room I	87
7. ESF Switchgear Room II	87
8. RHR Pump Room A	119
9. RHR Pump Room B	119
10. CTMT Spray Pump Room A	119
11. CTMT Spray Pump Room B	119
12. Safety Injection Pump Room A	119
13. Safety Injection Pump Room B	119
14. Centrifugal Charging Pump Room A	119
15. Centrifugal Charging Pump Room B	119
16. Electrical Penetration Room A	101
17. Electrical Penetration Room B	101
18. Component Cooling Water Room A	119
19. Component Cooling Water Room B	119
20. Diesel Generator Room A	119
21. Diesel Generator Room B	119
22. Control Room	84

3/4.8 ELECTRICAL POWER SYSTEMS

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3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

   1) A separate day tank containing a minimum volume of 390 gallons of fuel,

   2) A separate Fuel Oil Storage System containing a minimum volume of 85,300 gallons of fuel, and

   3) A separate fuel transfer pump.

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1~~a~~. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specifications 4.8.1.1.1~~a~~. and 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

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ACTION (Continued)

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Specification 4.8.1.1.2a.4) within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Specification 4.8.1.1.1~~a~~ within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, ~~and~~

~~b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring manually unit power supply from the normal circuit to the alternate circuit~~.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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SURVEILLANCE REQUIREMENTS (Continued)

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

   1) Verifying the fuel level in the day tank,

   2) Verifying the fuel level in the fuel storage tank,

   3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

   4) Verifying the diesel starts from ambient condition and accelerates to at least ~~471~~ 514 rpm in less than or equal to 12 seconds.* The generator voltage and frequency* shall be ~~4160~~ 4000 ± ~~420~~ 320 volts and 60 ± 1.2 Hz within 12 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

      a) Manual, or

      b) Simulated loss-of-offsite power by itself, or

      c) Safety Injection test signal.

   5) Verifying the generator is synchronized, loaded to greater than or equal to 6201 kW in less than or equal to 60 seconds,* operates with a load greater than or equal to 6201 kW for at least 60 minutes, and

   6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

~~d. At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:~~

See insert A

   ~~1) A water and sediment content of less than or equal to 0.05 volume percent;~~

   ~~2) A kinematic visocity at 40°C of greater thatn or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;~~

---

* See insert following page

INSERT

*These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

Insert A

4.8.1.1.2.d. By sampling new fuel oil in accordance with ASTM D4057 prior to addition to storage tanks and:

(1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

(a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees,

(b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification,

(c) A flash point equal to or greater than 125°F, and

(d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

(2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

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SURVEILLANCE REQUIREMENTS (Continued)

3) ~~A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees;~~

4) ~~An impurity level of less than 2 mg. of insolubles per 100 ml when tested in accordanxe with ASTM-D2274-70; analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and~~

5) ~~The other properties specified in Table 1 of ASTM-D975- 1977 and Regulatory guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordanc~~~~y~~e ~~with ASTM-D975-1977; analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.~~

f. ~~e~~. At least once per 18 months, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2) Verifying the diesel generator capability to reject a load of greater than or equal to 1532 kW (ESW pump) while maintaining voltage at ~~4160~~ 4000 ± ~~420~~ 320 volts and frequency at 60 ± ~~1.2~~ 5.4 Hz,

3) Verifying the diesel generator capability to reject a load of 6201 kW without tripping. The generator.voltage shall not exceed 4784 volts during and following the load rejection,

4) Simulating a loss-of-offsite power by itself, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected shutdown loads through the shutdown sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~4160~~ 4000 ± ~~420~~ 320 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on a Safety Injection test signal without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes; and the offsite power source energizes the autoconnected emergency (accident) load through the LOCA sequencer. The generator voltage and frequency shall be

4000 320
~~4160~~ ± ~~420~~ volts and 60 ± 1.2 Hz within 12 seconds after the auto-start signal; the generator steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with a Safety Injection test signal, and

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected emergency (accident) loads through the LOCA sequencer and operates ~~operators~~ for greater than or equal to 5 minutes whiles its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~4160~~ ± ~~420~~ volts and 60 ± 1.2 Hz during this test; and 4000 ± 320

   c) Verifying that all automatic diesel generator trips, except high jacket coolant temperature, engine overspeed, low lube oil pressure, high crankcase pressure, start failure relay, and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 6821 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 6201 kW. The generator voltage and frequency shall be ~~4160~~ 4000 ± ~~420~~ 320 volts and 60 ± 1.2 ~~Hz~~, -3 Hz within 12 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within 4000 ± 320 volts and 60 ± 1.2 Hz ~~these limits~~ during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2a.6)b)*;

8) Verifying that the auto-connected loads to each diesel generator do not exceed 6635 kW;

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

---

*If Specification 4.8.1.1.2a.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 6201 kW for 1 hour or until operating temperature has stabilized.

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b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross-connection lines; and

12) Verifying that the automatic LOCA and Shutdown sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

g ~~f~~. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least ~~471~~ 514 rpm in less than or equal to ~~10~~ 12 seconds; and

h ~~g~~. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or equivalent, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.~~1~~2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
≤ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
≥ 4	At least once per 3 days

---

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the ~~OL issuance date shall be included in the computation of the "last 100 valid tests." Entry into this test schedule shall be made at the 31-day test frequency.~~ completion of the preoperational test requirements of Regulatory Guide 1.108, Revision 1, August 1977, shall be included in the computation of the "last 100 valid tests."

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A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

   1) A day tank containing a minimum volume of 390 gallons of fuel,

   2) A fuel storage system containing a minimum volume of 85,300 gallons of fuel, and

   3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the spent fuel pool. ~~and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2 square inch vent.~~ In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5)), and 4.8.1.1.3.

3/4.8.2 D.C. SOURCES

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OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt Battery Bank NK11 and NK13, and its associated Full Capacity Chargers NK21 and NK23, ~~or~~ and

b. 125-Volt Battery Bank NK12 and NK14, and its associated Full Capacity Chargers NK22 and NK24.

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

ACTION:

a. With one of the required battery banks and/or full capacity chargers inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. ~~With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Specification 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.~~

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to ~~122~~ 130.2 volts on float charge.

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SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

   1) The parameters in Table 4.8-2 meet the Category B limits,

   2) There is no visible corrosion at either terminals or connectors, or the cell-to-cell and terminal connection resistance of these items is less than ~~(~~150 x 10^{-6}~~)~~ ohm, and

   3) The average electrolyte temperature of at least every sixth cell is above 60°F.

c. At least once per 18 months by verifying that:

   1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

   2) The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

   3) The resistance of each cell-to-cell and terminal connection is less than or equal to ~~(~~150 x 10^{-6}~~)~~ ohm, and

   4) The battery charger will supply at least 300 amperes at ~~125~~ ~~134~~ 130.2 volts for at least 1 hour.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status Simulated ~~all of the actual~~ emergency loads for ~~200 minutes~~ the design duty cycle when the battery is subject to a battery service test;

e. At least once per 60 months, during shutdown, Once per 60 month interval by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battary that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

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TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates., and not overflowing
Float Voltage	≥ ~~2.05~~ 2.13 volts	≥ ~~2.05~~ 2.13 volts(6)	> ~~1.99~~ 2.07 volts
Specific Gravity(4)	≥ ~~1.205~~ 1.200(5)	≥ 1.195	Not more than ~~0.050~~ 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(5)

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

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ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt Battery Bank NK11 and NK13, and its associated full capacity Chargers NK21 and NK23, or

b. 125-Volt Battery Bank NK12 and NK14, and its associated full capacity Chargers NK22 and NK24.

APPLICABILITY: MODES 5 and 6.

ACTION:

and/or full capacity charger

~~a.~~ With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible, ~~and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2 square inch vent~~.

~~b. With the required full-capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Specification 4.8.2.1a.1) within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.~~

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and associated chargers shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

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3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open ~~(both)~~ between redundant busses within the unit ~~(and between units at the same station)~~:

See attached LCO items a. through j.

- ~~a. Division #1 A.C. Emergency Busses consisting of:~~
  - ~~1) 4160-Volt Emergency Bus #NB01, and~~
  - ~~2) 480-Volt Emergency Bus #NG01, NG03 and NG05E.~~
- ~~b. Division #2 A.C. Emergency Busses consisting of:~~
  - ~~1) 4160-Volt Emergency Bus #NB02, and~~
  - ~~2) 480-Volt Emergency Bus #NG02, NG04 and NG06E.~~
- ~~c. 120-Volt A.C. Vital Busses #N~~K~~N01 and N~~K~~N03 energized from their associated inverter connected to D.C. Busses #~~ ~~NN11~~ NK01 ~~and~~ ~~NN13~~ NK03,
- ~~d. 120-Volt A.C. Vital Busses #N~~K~~N02 and N~~K~~N04 energized from their associated inverter connected to D.C. Busses #~~ ~~NN12~~ NK02 ~~and~~ ~~NN14~~ NK04,
- ~~e. 125-Volt D.C. Busses NK01 and NK03 energized from Batteries NK11 and NK13, and~~
- ~~f. 125-Volt D.C. Busses NK02 and NK04 energized from Batteries NK12 and NK14.~~

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

ACTION:

a. With one of the required divisions of A.C. emergency busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. bus not energized from its associated battery bank~~,~~ or charger reenergize the D.C. bus from its associated battery bank and charger within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

a. Division #1 A.C. Emergency Busses consisting of:

   1) 4160-Volt Emergency Bus #NB01, and
   2) 480-Volt Emergency Busses #NG01, NG03 and NG05E.

b. Division #2 A.C. Emergency Busses consisting of:

   1) 4160-Volt Emergency Bus #NB02, and
   2) 430-Volt Emergency Busses #NG02, NG04 and NG06E.

c. 120-Volt A.C. Vital Bus #NN01 energized from its associated inverter connected to D.C. Bus #NK01,

d. 120-Volt A.C. Vital Bus #NN02 energized from its associated inverter connected to D.C. Bus #NK02,

e. 120-Volt A.C. Vital Bus #NN03 energized from its associated inverter connected to D.C. Bus #NK03,

f. 120-Volt A.C. Vital Bus #NN04 energized from its associated inverter connected to D.C. Bus #NK04,

g. 125-Volt D.C. Bus #NK01 energized from Battery #NK11 and Charger #NK21,

h. 125-Volt D.C. Bus #NK0 energized from Battery #NK12 and Charger #NK22,

i. 125-Volt D.C. Bus #NK03 energized from Battery #NK13, and Charger #NK23

j. 125-Volt D.C. Bus #NK04 energized from Battery #NK14 and Charger #NK24

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

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ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, one or the following divisions of electrical busses shall be energized in the specified manner:

See attached LCO items a. & b.

a. ~~One division of A.C. emergency busses consisting of one 4160-volt and one 480-volt A.C. emergency bus,~~

b. ~~Two 120-volt A.C. vital busses energized from their associated inverters connected to their respective D.C. busses, and~~

c. ~~Two 125-volt D.C. busses energized from their associated battery banks.~~

APPLICABILITY: MODES 5 and 6.

ACTION:

With out ~~any~~ one of the above required divisions of electrical busses ~~not~~ energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to energize at least one division of the required ~~electrical~~ busses in the specified manner. ~~as soon as possible, and within 8 hours depressurize and vent the RCS through at least a 2 square inch vent.~~

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

a. Division 1, consisting of:

  1) 4160-volt emergency bus #NB01 and

  2) 480-volt emergency busses #NG01, NG03, and NG05E, and

  3) 120-volt A.C. vital busses #NN~~21~~01 and NN~~23~~03 energized from their associated inverter connected to D.C. busses #NK01 and NK03, and

  4) 125-volt D.C. busses #NK01 and NK03 energized from batteries #NK11 and NK13, ~~or~~ and Chargers #NK21 and NK23, or

b. Division 2, consisting of:

  1) 4160-volt emergency bus #NB02, and

  2) 480-volt emergency busses #NG02, NG04 and NG06E, and

  3) 120-volt A.C. vital busses #NN~~22~~02 and NN~~24~~04 energized from their associated inverter connected to D.C. busses #NK02 and NK04, and

  4) 125-volt D.C. busses #NK02 and NK04 energized from batteries #NK12 and NK14, and Chargers #NK22 and #NK24

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be OPERABLE.

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

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices given in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

  1) By verifying that the 13.8 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers ~~of each voltage level~~, and performing the following:

    a) A CHANNEL CALIBRATION of the associated protective relays,

    b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-1, and

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage ~~400-volt~~ circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. insert ~~The functional test shall consist of injecting a current input at the specified Setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed and the response time is less than or equal to the specified value.~~ Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and

3) By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a nondestructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

Insert

Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer.

Use revised Table 3.8-1 from Callaway specs. attached behind Wolf Creek Table 3.8-1 rather than Wolf Creek table.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
13.8-kV Switchgear			
Primary (P) 252PA0107	3600 (50)/ 372 (51) & 840 (51)	0.1	Reactor coolant pump DPBB01A
P-252PA0108	3163(50)/372 (51) & 840 (51)	0.1	Reactor coolant pump DPBB01B
P-252PA0205	3163 (50)/372(51) & 840 (51)	0.1	Reactor coolant pump DPBB01C
P-252PA0204	3163 (50/372 (51) & 840 (51)	0.1	Reactor coolant pump DPBB01D
480-Volt Load Center			
P-52NG0304 B-52NG0301	1200 (Inst.) 4320 (S.T.)	0.05 0.5	Hydrogen recombiner SGS01A
P-52NG404 B-52NG0401	1200 (Inst.) 4320 (S.T.)	0.05 0.5	Hydrogen recombiner SGS01B
P-52PG2102 Through 52PG2112 B-350 A Fuse	375 (Inst.)	0.025 N.A.	Pressurizer backup heater

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
P-52PG2202 Through 52PG2212 B-350 A Fuse	375 (Inst.)	0.025 N.A.	Pressurizer Backup Heaters
P-52NG01TAF1 B-52NG0108	2000 ~~1450~~ (Inst.) 2400 (S.T.)	0.016 0.19	Containment Cooler DSGN01A
P-52NG03TAF B-52NG0305	2000 ~~1450~~ (Inst.) 2400 (S.T.)	0.016 0.19	Containment Cooler DSGN01C
P-52NG02TAF1 B-52NG0208	2000 ~~1450~~ (Inst.) 2400 (S.T.)	0.016 0.19	Containment Cooler DSGN01B
P-52NG04TAF1 B-52NG0405	2000 ~~1450~~ (Inst.) 2400 (S.T.)	0.016 0.19	Containment Cooler DSGN01D
480 V-Motor Control Center			
P-52NG01BDF3 B-40A Fuse	75 (Inst.)	0.016 N.A.	RHR Loop Inlet Iso Vlv EJHV8701B
P-52NG02BHR2 B-15A Fuse	10 (Inst.)	0.016 N.A.	ESW from Ctmt Air Coolers Iso Vlv EFHV46
P-52NG02BDF2 B-40A Fuse	45 (Inst.)	0.016 N.A.	CCW to Ctmt Iso Vlv EGHV60

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
P-52NG02BDF3 B-15A Fuse	29 (Inst.)	0.016 N.A.	Ctmt Iso Vlv Ret from Thrm Barrier Cooling Coil EGHV62
P-52NG02BEF2 B-60A Fuse	170 (Inst.)	0.016 N.A.	Sump to RHR Pump Vlv EJHV8811B
P-52NG02BEF3 B-15A Fuse	29 (Inst.)	0.016 N.A.	Ctmt Recirc Sump Iso Vlve ENHV7
P-52NG02BGF3 B-60A Fuse	565 ~~320~~ (Inst.)	0.016 N.A.	Accumulator Iso Vlv EPHV8808B
P-52NG02BHF2 B-60A Fuse	565 ~~320~~ (Inst.)	0.016 N.A.	Accumulator Iso Vlv EPHV8808D
P-52NG02BFF3 B-15A Fuse	10 (Inst.)	0.016 N.A.	H_2 Control System Make-up Air Valve KAHV30
P-52NG01BBF3 B-15A Fuse	29 (Inst.)	0.016 N.A.	RC Pump Seal Water Iso Vlv BGHV8112
P-52NG01BFF3 B-60A Fuse	170 (Inst.)	0.016 N.A.	Sump to RHR Pump Vlv EJHV8811A
P-52NG01BEF3 B-15A Fuse	29 (Inst.)	0.016 N.A.	Ctmt Recirc Sump Iso Vlv ENHV1

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
P-52NG01BGF3 B-60A Fuse	565 ~~320~~ (Inst.)	0.016 N.A.	Accumulator Iso Vlv EPHV8808A
P-52NG01BGF2 B-60A Fuse	565 ~~320~~ (Inst.)	0.016 N.A.	Accumulator Iso Vlv EPHV8808C
P-52NG01BFF2 B-15A Fuse	10 ~~9~~ (Inst.)	0.016 N.A.	Ctmt Air to Aux Bldg ESF Filter Iso Vlv GSHV20
P-52NG01BBR2 B-15A Fuse	29 (Inst.)	0.016 N.A.	React Bldg Discharge Iso Vlve LFFV95
P-52NG02BBF3 B-40A Fuse	75 ~~72~~ (Inst.)	0.016 N.A.	RHR Loop Inlet Iso Vlv BBPV8702B
P-52NG02BCF2 B-40A Fuse	75 ~~84~~ (Inst.)	0.016 N.A.	RHR Loop Inlet Iso Vlv BBPV8702A
P-52NG02BHF3 B-15A Fuse	10 (Inst.)	0.016 N.A.	ESW to Ctmt Air Coolers Iso Vlv EFHV34
P-52NG01BCF2 B-15A Fuse	10 (Inst.)	0.016 N.A.	ESW to Ctmt Air Coolers Iso Vlv EFHV33
P-52NG01BDF2 B-15A Fuse	10 (Inst.)	0.016 N.A.	ESW from Ctmt Air Coolers Iso Vlv EFHV45

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
P-52NG01BEF2 B-40A Fuse	75 (Inst.)	0.016 N.A.	RHR Loop Inlet Iso Vlv EJHV8701A
P-52NG03CDF4 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Valve BBHV13
P-52NG03CHF1 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV14
P-52PG19NAF4 B-100A Fuse	400 (Inst.)	0.016 N.A.	Reactor Cavity Cooling Fan PCGN02A
P-52PG19NCF3 B-60A Fuse	260 (Inst.)	0.016 N.A.	Ctmt Atmospheric Control System Fan DCGR01A
P-52PG19NGF2 B-40 Fuse	675 ~~300~~ (Inst.)	0.017 N.A.	RCP A Space Heater
P-52PG19NGF3 B-40 Fuse	675 ~~300~~ (Inst.)	0.017 N.A.	RCP B Space Heater
P-52PG19NEF1 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP A Oil Lift Pump
P-52PG19NGR3 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP B Oil Lift Pump
P-52PG19NFF1 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
P-52PG19NFF2 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05C
P-52PG19NAF2 B-40A Fuse	84 (Inst.)	0.016 N.A.	Instrument Tunnel Sump Pump DPLF07A
P-52NG03CBF4 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV15
P-52NG03CDF2 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV16
P-52PG20NBF5 B-100A Fuse	320 (Inst.)	0.016 N.A.	Reactor Cavity Cooling Fan DCGN02B
P-52PG20NFF4 B-60A Fuse	260 (Inst.)	0.016 N.A.	Ctmt Atmospheric Control System Fan DCGR01B
P-52PG20NBF1 B-40A Fuse	615 ~~300~~ (Inst.)	0.017 N.A.	RCP C Space Heater
P-52PG20NCF1 B-40A Fuse	615 ~~300~~ (Inst.)	0.017 N.A.	RCP D Space Heater
P-52PG20NFF3 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP C Oil Lift Pump

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
P-52PG20NFF2 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP D Oil Lift Pump
P-52PG20NER2 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05B
P-52PG20NGF4 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05D
P-52PG20MDR2 B-40A Fuse	84 (Inst.)	0.016 N.A.	Instrument Tunnel Sump DPLF07B
P-52PG1904 B-600A Fuse	1440 (Inst.)	0.03 N.A.	Polar Crane AKE 13
CRDM Control Rod Drive Power			
P-10A Fuse B-30A Fuse	- - - - - -	N.A. N.A.	Gripper Coils (106 fused circuits)
P-50A Fuse B-150A Fuse	- - - - - -	N.A. N.A.	Lift Coils (53 fused circuits)

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(50) - Protective Relay Instantaneous Unit
(51) - Protective Relay Inverse Time Unit
Inst. - Instantaneous Protection
S.T. - Short Time Protection

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
13.8-kV Switchgear	
P-252PA0107 B-252PA0110/252PA0101	Reactor Coolant Pump DPBB01A
P-252PA0108 B-252PA0110/252PA0101	Reactor Coolant Pump DPBB01B
P-252PA0205 B-252PA0211/252PA0202	Reactor Coolant Pump DPBB01C
P-252PA0204 B-252PA0211/252PA0202	Reactor Coolant Pump DPBB01D
480-V Load Center	
P-125A Fuse B-52NG0304	Hydrogen Recombiner SGS01A
P-125A Fuse B-52NG0404	Hydrogen Recombiner SGS01B
P-52PG2102 Through 52PG2111 B-250 A Fuse	Pressurizer Backup Heaters
P-52PG2202 Through 52PG2211 B-250 A Fuse	Pressurizer Backup Heaters
P-52NG01TAF1 B-52NG0108	Containment Cooler DSGN01A
P-52NG03TAF1 B-52NG0305	Containment Cooler DSGN01C
P-52NG02TAF1 B-52NG0208	Containment Cooler DSGN01B
P-52NG04TAF1 B-52NG0405	Containment Cooler DSGN01D
P-52PG2007 B-52PG2001	PG20P MCC

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVEPCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Load Center (Continued)	
P-52PG2402 B-500A Fuse	Pressurizer Heater
P-52PG2403 B-500A Fuse	Pressurizer Heater
P-52PG2404 B-500A Fuse	Pressurizer Heater
P-52PG2405 B-500A Fuse	Pressurizer Heater
P-52PG2406 B-500A Fuse	Pressurizer Heater
P-52PG2407 B-500A Fuse	Pressurizer Heater
480-V Motor Control Center	
P-52NG01BDF3 B-40A Fuse	RHR Loop Inlet Iso Vlv EJHV8701B
P-52NG02BHR2 B-15A Fuse	ESW from Ctmt Air Coolers Iso Vlv EFHV46
P-52NG02BDF2 B-30A Fuse	CCW to Ctmt Iso Vlv EGHV60
P-52NG01BHF3 B-40A Fuse	CCW Containment Isolation Valve EGHV132
P-52NG01BDF1 B-15A Fuse	PRT to Containment Valve BBHV8037A
P-52NG01BBR3 B-30A Fuse	Pressurizer Relief Valve BBHV8000A
P-52PG19NEF5 B-100A Fuse	Reactor Coolant Drain Tank Pump DPHB02A
P-52PG19NCR3 B-150A Fuse	Lighting Transformer XQA26

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
P-52PG19NCR5 B-15A Fuse	Machine Rm Exhaust Fan DCGN04
P-5A Fuse B-52PG19GDF3	Flux Mapping Motor Starters SR06A
P-5A Fuse B-52PG19GDF6	Flux Mapping Motor Starters SR06B
P-52NG03CLF2 B-15A Fuse	RCP thermal barrier return isolation Vlv BBHV16
P-52NG01BHF4 B-40A Fuse	CCW Containment Isolation Valve EGHV130
P-52NG02BBF4 B-250A Fuse	Hydrogen Mixing Fan DCGN03B
P-52NG02BCF3 B-250A Fuse	Hydrogen Mixing Fan DCGN03D
P-52NG01BBF4 B-250A Fuse	Hydrogen Mixing Fan DCGN03A
P-52NG01BCF3 B-250A Fuse	Hydrogen Mixing Fan DCGN03C
P-52NG02BJF5 B-150A Fuse	CRDM Cooling Fan DCGN01B
P-52PG20GAR2 B-150A Fuse	CRDM Cooling Fan DCGN01A
P-52PG20NFR3 B-15A Fuse	Pressurizer Cooling Fan DCGN05
P-52PG20NEF5 B-100A Fuse	Reactor Coolant Drain Tank Pump DPHB02B
P-52PG20NEF1 B-150A Fuse	Lighting Transformer XQA28

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
P-52NG02BJF1 B-40A Fuse	Standby Lighting
P-52NG02BJF2 B-40A Fuse	Standby Lighting
P-52NG02BHR3 B-15A Fuse	PRT to Containment Vlv BBHV8037B
P-52NG02BDF1 B-30A Fuse	Pressurizer Relief Valve BBHV8000B
P-52NG01BJF5 B-150A Fuse	CRDM Cooling Fan DCGN01D
P-52PG19GBR2 B-150A Fuse	CRDM Cooling Fan DCGN01C
P-52NG02BDF3 B-15A Fuse	Ctmt Iso Vlv Ret from Thrm Barrier Cooling Coil EGHV62
P-52NG02BEF2 B-60A Fuse	Sump to RHR Pump Vlv EJHV8811B
P-52NG02BEF3 B-15A Fuse	Ctmt Recirc Sump Iso Vlv ENHV7
P-52NG02BGF3 B-60A Fuse	Accumulator Iso Vlv EPHV8808B
P-52NG02BHF2 B-60A Fuse	Accumulator Iso Vlv EPHV8808D
P-52NG02BFF3 B-15A Fuse	H_2 Control System Make-up Air Vlv KAHV30
P-52NG01BBF3 B-15A Fuse	RC Pump Seal Water Iso Vlv BGHV8112
P-52NG01BFF3 B-60A Fuse	Sump to RHR Pump Vlv EJHV8811A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
P-52NG01BEF3 B-15A Fuse	Ctmt Recirc Sump Iso Vlv ENHV1
P-52NG01BGF3 B-60A Fuse	Accumulator Iso Vlv EPHV8808A
P-52NG01BGF2 B-60A Fuse	Accumulator Iso Vlv EPHV8808C
P-52NG01BFF2 B-15A Fuse	Ctmt Air to Aux Bldg ESF Filter Iso Vlv GSHV20
P-52NG01BBR2 B-15A Fuse	React Bldg Discharge Iso Vlv LFFV95
P-52NG02BBF3 B-40A Fuse	RHR Loop Inlet Iso Vlv BBPV8702B
P-52NG02BCF2 B-40A Fuse	RHR Loop Inlet Iso Vlv BBPV8702A
P-52NG02BHF3 B-15A Fuse	ESW to Ctmt Air Coolers Iso Vlv EFHV34
P-52NG01BCF2 B-15A Fuse	ESW to Ctmt Air Coolers Iso Vlv EFHV33
P-52NG01BDF2 B-15A Fuse	ESW from Ctmt Air Coolers Iso Vlv EFHV45
P-52NG01BEF2 B-40A Fuse	RHR Loop Inlet Iso Vlv EJHV8701A
P-52NG03CDF4 B-15A Fuse	RCP Thermal Barrier CCW Iso Valve BBHV13
P-52NG03CHF1 B-15A Fuse	RCP Thermal Barrier CCW Iso Vlv BBHV14
P-52PG19NAF4 B-100A Fuse	Reactor Cavity Cooling Fan DCGN02A
P-52PG19NCF3 B-60A Fuse	Ctmt Atmospheric Control System Fan DCGR01A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
P-52PG19NGF2 B-40 Fuse	RCP A Space Heater
P-52PG19NGF3 B-40 Fuse	RCP B Space Heater
P-52PG19NEF1 B-40A Fuse	RCP A Oil Lift Pump
P-52PG19NGR3 B-40A Fuse	RCP B Oil Lift Pump
P-52PG19NFF1 B-15A Fuse	Ctmt Normal Sump Pump DPLF05A
P-52PG19NFF2 B-15A Fuse	Ctmt Normal Sump Pump DPLF05C
P-52PG19NAF2 B-25A Fuse	Instrument Tunnel Sump Pump DPLF07A
P-52NG03CBF4 B-15A Fuse	RCP Thermal Barrier CCW Iso Vlv BBHV15
P-52NG03CLF2 B-15A Fuse	RCP Thermal Barrier CCW ISo Vlv BBHV16
P-52PG20NBF5 B-100A Fuse	Reactor Cavity Cooling Fan DCGN02B
P-52PG20NFF4 B-60A Fuse	Ctmt Atmospheric Control System Fan DCGR01B
P-52PG20NBF1 B-40A Fuse	RCP C Space Heater
P-52PG20NCF1 B-40A Fuse	RCP D Space Heater
P-52PG20NFF3 B-40A Fuse	RCP C Oil Lift Pump
1EPR08C P-3A Fuse RP139 B-3A Fuse	Accumulator Tank A Isol Vlv EPHV8808A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
1EPR08D P-3A Fuse RP139 B-3A Fuse	Accumulator Tank C Isol Vlv EPHV8808C
4EPR08A P-3A Fuse RP140 B-3A Fuse	Accumulator Tank B Isol Vlv EPHV8808B
4EPR08B P-3A Fuse RP140 B-3A Fuse	Accumulator Tank D Isol Vlv EPHV8808D
1EPK09B P-3A Fuse RL018 B-3A Fuse	Accumulator Tank B Vent Vlv EPHV8950B
IEPK09D P-3A Fuse RL018 B-3A Fuse	Accumulator Tank C Vent Vlv EPHV8950D
1EPK09F P-3A Fuse RL018 B-3A Fuse	Accumulator Tank D Vent Vlv EPHV8950F
4EPK09A P-3A Fuse RL018 B-3A Fuse	Accumulator Tank A Vent Vlv EPHV8950A
4EPK09C P-3A Fuse RL018 B-3A Fuse	Accumulator Tank B Vent Vlv EPHV8950C
4EPK09E P-3A Fuse RL018 B-3A Fuse	Accumulator Tank.C Vent Vlv EPHV8950E
4GTK03B P-3A Fuse RL020 B-3A Fuse	Cont. Mini Purge Isol Vlv GTHZ11
4GTK03D P-3A Fuse RL020 B-3A Fuse	Cont. Purge Isol Vlv GTHZ8
5BBA01A P-15A Fuse PA0107 B-15A Fuse	RCP Breaker Control
5BBA01B P-15A Fuse PA0108 B-15A Fuse	RCP Breaker Control
6BBA01C P-15A Fuse PA0205 B-15A Fuse	RCP Breaker Control
6BBA01D P-15A Fuse PA0204 B-15A Fuse	RCP Breaker Control

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
480-V Motor Control Center (Continued)	
5BBG02A P-2A Fuse PG19NEF1 B-1A Fuse	RCP Oil Lift Pump Control
P-52PG20NFF2 B-40A Fuse	RCP D Oil Lift Pump
P-52PG20NER2 B-15A Fuse	Ctmt Normal Sump Pump DPLF05B
P-52PG20NGF4 B-15A Fuse	Ctmt Normal Sump Pump DPLF05D
P-52PG20NDR2 B-25A Fuse	Instrument Tunnel Sump Pump DPLF07B
P-52PG1904 B-600A Fuse	Polar Crane HKE13
CRDM Control Rod Drive Power	
P-10A Fuse B-30A Fuse	Gripper Coils (106 fused circuits)
P-50A Fuse B-150A Fuse	Lift Coils (52 fused circuits)
Low Voltage Power and Control	
6HBK04B P-3A Fuse HB115 B-3A Fuse	RCDT Heat Exchanger return to RCDT Vlv HBHV7144
6HBK05A P-3A Fuse HB115 B-3A Fuse	RCDT Discharge Vlv HBHV7143
1HBK19A P-3A Fuse RL021 B-3A Fuse	Containment Isolation Vlv HBHV7176
4KAG04A P-2A Fuse NG02BFF3 B-1A Fuse	Hydrogen Purge Makeup Air Supply Vlv KAHV30
5KCQ15S P-3A Fuse KC274A B-3A Fuse	Fire Protection Discharge Vlv KCXV261

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
6KCQ15S P-3A Fuse KC274A B-3A Fuse	Fire Protection Discharge Vlv KCXV262
5KCQ19X P-3A Fuse KC274A B-3A Fuse	Fire Protection Detector KCHPS261-002
5KCQ19Y P-3A Fuse KC274A B-3A Fuse	Fire Protection Detector KCHPS262-002
6KES01A P-5A Fuse KE124 B-15A Fuse	Fuel Transfer Panel KE124
6KES01A P-5A Fuse KE125 B-15A Fuse	Fuel Transfer Panel KE125
5LFG06A P-2A Fuse PG19NFF1 B-1A fuse	Containment Normal Sump Pump A DPLF05A
5LFG06C P-2A Fuse PG19NFF2 B-1A Fuse	Containment Normal Sump Pump C DPLF05C
6LFG06B P-2A Fuse PG20NER2 B-1A Fuse	Containment Normal Sump Pump B DPLF05B
6LFG06D P-2A Fuse PG20NGF4 B-1A Fuse	Containment Normal Sump Pump D DPLF05D
5GNG03A P-5A Fuse NG01BJF5 B-3A Fuse	CRDM Cooling Fan D Discharge Isolation Damper GNHZ44
5GNG03C P-5A Fuse PG19GBR2 B-3A Fuse	CRDM Cooling Fan C Discharge Isolation Damper GNHZ43
1EMK04B P-3A Fuse RL017 B-3A Fuse	SIS Test Line Vlv EMHV8824
1EMK04D P-3A Fuse RL017 B-3A Fuse	SIS Test Line Vlv EMHV8881
4EMK04C P-3A Fuse RL017 B-3A Fuse	BIT Test Line Vlv EMHV8843

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
4EMK04E P-3A Fuse RL017 B-3A Fuse	SI Test Line Vlv EMHV8871
5EMK05A P-3A Fuse RL017 B-3A Fuse	SI Test Line Vlv EMHV8889A
5EMK05C P-3A Fuse RL017 B-3A Fuse	SI Test Line Vlv EMHV8889C
5EMK05E P-3A Fuse RL018 B-3A Fuse	BIT Test Line Vlv EMHV8882
6EMK05B P-3A Fuse RL017 B-3A Fuse	SI Test Line Vlv EMHV8889D
6EMK05D P-3A Fuse RL017 B-3A Fuse	SI Test Line Vlv EMHV8889B
1ENG02A P-2A Fuse NG01BEF3 B-1A Fuse	Containment Spray Sump Isol Vlv ENHV1
4ENG02B P-2A Fuse NG02BEF3 B-1A Fuse	Containment Spray Sump Isol Vlv ENHV7
1EPG02A P-2A Fuse NG01BGF3 B-1A Fuse	Accumulator Tank Isolation Valve EPHV8808A
1EPG02B P-2A Fuse NG01BGF2 B-1A Fuse	Accumulator Tank Isolation Valve EPHV8808C
1EPK02C P-3A Fuse RL018 B-3A Fuse	Accumulator Tank Isolation Valve EPHV8808B Indication
1EMK04A P-3A Fuse RL017 B-3A Fuse	SI Test Line Vlv EMHV8823
4EJG04B P-2A Fuse NG02AFR3 B-1A Fuse	RHR to charging/SI pump suctions EJHV8804B
1EJG05A P-2A Fuse NG01BEF2 B-1A Fuse	RHR Shutdown Suction Line Isol Valve EJHV8701A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
1EJG05B P-2A Fuse NG01BDF3 B-1A Fuse	RHR Shutdown Suction Line Isolation Valve EJHV8701B
1EJG06A P-2A Fuse NG01BFF3 B-1A Fuse	Cont Recirc Sump Isolation Valve EJHV8811A
4EJG06B P-2A Fuse NG02BEF2 B-1A Fuse	Cont Recirc Sump Isolation Valve EJHV8811B
1EJK07A P-3A Fuse RL017 B-3A Fuse	Test Line Isol Vlv Hot Leg Inj Line Solenoid EJHCV8825
1EJK07C P-3A Fuse RL017 B-3A Fuse	RHR Test Line Vlv EJHCV8890A
4EJK07B P-3A Fuse RL017 B-3A Fuse	RHR Test Line Vlv EJHCV8890B
P-1EJY13A 3A Fuse RL011 B-1RLY01G 15A Breaker NG01ACR119	Ctmt Sump Sample Isolation Vlv EJHV21
P-4EJY13B 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Ctmt Sump Sample Isolation Vlv EJHV22
P-4BMY01D 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.C Cnt to Nuc Sample Sys Vlv BMHV22
P-4BMY02A 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.A Tube Sheet Sample Vlv BMHV35
P-4BMY02B 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.B Tube Sheet Sample Vlv BMHV36

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-4BMY02C 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.C Tube Sheet Sample Vlv BMHV37
P-4BMY02D 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G.D Tube Sheet Sample Vlv BMHV38
1BNG03A P-2A Fuse NG01ACR2 B-1A Fuse	RHR Pump RWST Suction Valve BNHV8812A
4BNG03B P-2A Fuse NG02AFF4 B-1A Fuse	RHR Pump RWST Suction Valve BNHV8812B
1EFG09A P-2A Fuse NG01BCF2 B-1A Fuse	Cont Cooler Isolation Valve EFHV33
1EFG09C P-2A Fuse NG01BDF2 B-1A Fuse	Cont Cooler Isolation Valve EFHV45
4EFG09B P-2A Fuse NG02BHF3 B-1A Fuse	Cont Cooler Isolation Valve EFHV34
4EFG09D P-2A Fuse NG02BHR2 B-1A Fuse	Cont Cooler Isolation Valve EFHV46
4EGG06A P-2A Fuse NG02BDF2 B-1A Fuse	RC Pump CCW Return Cont Isol Vlv EGHV60
4EGG10A P-2A Fuse NG02BDF3 B-1A Fuse	Cont Isol Vlv CCW Return from RC pump Ther Barr Vlv EGHV62
1EGG17A P-2A Fuse NG01BHF4 B-1A Fuse	EGHV-60 Bypass Valve Cont Isol Vlv EGHV130
1EGG17B P-2A Fuse NG01BHF3 B-1A fuse	EGHV-62 Bypass Valve Cont Isol Vlv EGHV132
1EJG04A P-2A Fuse NG03CMF4 B-1A Fuse	RHR to Charging/SI Pump Suction Vlv EJHV8804A

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
1BBK35A P-3A Fuse RL001 B-3A Fuse	Excess Letdown Path to PRT Isol Vlv BBHV8157A
4BBK35B P-3A Fuse RL001 B-3A Fuse	Excess Letdown Path to PRT Isol Vlv BBHV8157B
1BBK37A P-3A Fuse RL021 B-3A Fuse	Pressurizer Safety Vlv BBHV8010A
1BBK37B P-3A Fuse RL021 B-3A Fuse	Pressurizer Safety Vlv BBHV8010B
4BBK37C P-3A Fuse RL021 B-3A Fuse	Pressurizer Safety Vlv BBHV8010C
1BBG39A P-2A Fuse NG01BBR3 B-1A Fuse	Pressurizer PORV Isolation Valve BBHV8000A
5GRK02A P-3A Fuse RL020 B-3A Fuse	Filtration Unit Damper GRPDZ5
6GRK02B P-3A Fuse RL020 B-3A Fuse	Filtration Unit Damper GRPDZ15
1GTK03A P-3A Fuse RL020 B-3A Fuse	Ctmt Minipurge Isol Valve GTHZ5
P-1GSY01D 3A Fuse RL011 B-1RLY01G 15A breaker NG01ACR119	Hydrogen Analyzer Ctmt Isol Vlv GSHV13
P-1GSY01E 3A Fuse RL011 B-1RLY01G 15A Breaker NG01ACR119	Hydrogen Analyzer Ctmt Isol Vlv GSHV14
P-1GSY10A 3A Fuse RL020 B-1RLY01A 15A Breaker NG01ACR123	Ctmt Atmosphere Monitor Isol Vlv GSHV31

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-1GSY10B 3A Fuse RL020 B-1RLY01A 15A Breaker NG01ACR123	Containment Atm Monitor Isol Vlv GSHV34
P-6SQY02B 15A Breaker PG20GBR238 B-6SQY02B 30A Fuse PG20GBR1	Loose Parts Simulator
P-6BBY16A 15A Breaker PG20GBR239 B-6BBY16A 30A Fuse PG20GBR1	Reactor Coolant Sys Level Alarm
P-6GNY09F 15A Breaker PG20NBR225 B-6GNY09F 30A Fuse PG20NBR1	CRDM Cooling Fan Space Heater
P-6GNY09H 15A Breaker PG20NBR226 B-6GNY09H 30A Fuse PG20NBR1	CRDM Cooling Fan Space Heater
P-6GNY09K 15A Breaker PG20NBR223 B-6GNY09K 30A Fuse PG20NBR1	Reactor Cavity Cooling Fan Space Heater
P-4EPY02C 15A Breaker NG02BAR114 B-4EPY02C 30A Fuse NG02BGR4	Accumulator Isolation Valve Space Heater
P-4EPY02D 15A Breaker NG02BAR115 B-4EPY02D 30A Fuse NG02BGR4	Accumulator Isolation Valve Space Heater
P-4GNY09B 15A Breaker NG02BAR110 B-4GNY09B 30A Fuse NG02BGR4	Containment Cooler Fan Space Heater

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-4GNY09D 15A Breaker NG02BAR111 B-4GNY09D 30A Fuse NG02BGR4	Containment Cooler Fan Space Heater
P-4GNY09M 15A Breaker NG02BAR120 B-4GNY09M 30A Fuse NG02BGR4	Hydrogen Mixing Fan Space Heater
P-4GNY09P 15A Breaker NG02BAR121 B-4GNY09P 30A Fuse NG02BGR4	Hydrogen Mixing Fan Space Heater
P-1GNY09A 15A Breaker NG01BAR119 B-1GNY09A 30A Fuse NG01BER3	Containment Cooler Fan Space Heater
P-1GNY09C 15A Breaker NG01BAR120 B-1GNY09C 30A Fuse NG01BER3	Containment Cooler Fan Space Heater
P-1GNY09L 15A Breaker NG01BAR122 B-1GNY09L 30A Fuse NG01BER3	Hydrogen Mixing Fan Space Heater
P-1GNY09N 15A Breaker NG01BAR123 B-1GNY09N 30A Fuse NG01BER3	Hydrogen Mixing Fan Space Heater
P-5GNY09E 15A Breaker PG19NHF228 B-5GNY09E 30A Fuse PG19NHF1	CRDM Cooling Fan Space Heater
P-1EPY02A 15A Breaker NG01BAR116 B-1EPY02A 30A Fuse NG01BER3	Accumulator Iso Valve Space Heater

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-1EPY02B 15A Breaker NG01BAR117 B-1EPY02B 30A Fuse NG01BER3	Accumulator ISO Valve Space Heater
P-4GSY10C 3A Fuse RL020 B-4RLY01A 15A Breaker NG02ACR130	Ctmt Atm Monitor Isol Vlv GSHV36
P-4GSY10D 3A Fuse RL020 B-4RLY01A 15A Breaker NG02ACR130	Ctmt Atm Monitor Isol Vlv GSHV39
P-5GNY09G 15A Breaker PG19NHF229 B-5GNY09G 30A Fuse PG19NHF1	CRDM Cooling Fan Space Heater
P-5GNY09J 15A Breaker PG19NHF225 B-5GNY09J 30A Fuse PG19NHF1	Reactor Cavity Cooling Fan Space Heater
5SFY11AA P-30A Fuse PN0711 B-60A Fuse	Rod Position Panel SF109A
5SFY11AB P-30A Fuse PN0710 B-60A Fuse	Rod Position Panel SF109B
1LFG08A P-2A Fuse NG01BBR2 B-1A Fuse	Normal sump Ctmt Isol Vlv LFFV95
5LFG15A P-2A Fuse PG19NAF2 B-1A Fuse	Instrument Tunnel Sump Pump DPLF07A
6LFG15B P-2A Fuse PG20NDR2 B-1A Fuse	Instrument Tunnel Sump Pump DPLF07B
1GSG03A P-2A Fuse NG01BFF2 B-1A Fuse	Hydrogen Purge Ctmt Isolation Vlv GSHV20
4BBG39B P-2A Fuse NG02BDF1 B-1A Fuse	Pressurizer PORV Isol Vlv BBHV8000B

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
1BBK40A P-30A Fuse NK5108 B-30A Fuse	PZR PORV BBPCV455A
4BBK04B P-30A Fuse NK4421 B-30A Fuse	PZR PORV BBPCV456A
5BGK04B P-3A Fuse RL001 B-3A Fuse	Alternate Charging Path Isol Valv BGHV8147
6BGK04A P-3A Fuse RL001 B-3A Fuse	Normal Charging Path Isol Valv BGHV8146
P-5LFY10A 3A Fuse RL023 B-5RLY01H 15A Breaker PG19GCR217	Containment Cooler Drain Valve LFLV97
P-5LFY10C 3A Fuse RL023 B-5RLY01H 15A Breaker PG19GCR217	Containment Cooler Drain Valve LFLV99
P-6LFY10B 3A Fuse RL023 B-6RLY01G 15A Breaker PG20GBR217	Containment Cooler Drain Valve LFLV98
P-6LFY10D 3A Fuse RL023 B-6RLY01G 15A Breaker PG20GBR217	Containment Cooler Drain Valve LFLV100
P-6LFY17A 3A Fuse RL023 B-6RLY01G 15A Breaker PG20GBR217	Refueling Pool Stand Pipe Discharge Valve LFLV122
P-5LFY20A 15A Breaker PG19NHF224 B-5LFY20A 30A Fuse PG19NHF1	Instrument Tunnel Sump Moisture Sensor TLVF01

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-6LFY20B 15A Breaker PG20NBR216 B-6LFY20B 30A Fuse PG20NBR1	Instrument Tunnel Sump Moisture Sensor TVLF02
P-5SDY06C 15A Breaker PG19NHF215 B-5SDY06C 30A Fuse PG19NHF1	Local Radiation Monitor Power Supplies SPRIA39-42
P-1SJY01D 3A Fuse RL011 B-1RLY01G 15A Breaker NG01ACR119	Press. Ctmt Isol Vlv SJHV128
P-4SJY01A 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Press. Liq/HL 1&3 Sample Clr Vlv SJHV5
1GTK03C P-3A Fuse RL020 B-3A Fuse	Ctmt Purge Isol Vlv GTHZ7
P-1GSY01E 3A Fuse RL011 B-1RLY01G 15A Breaker NG01ACR119	Hydrogen Analyzer Ctmt Sample Vlv GSHV14
P-1GSY01F 3A Fuse RL011 B-1RLY01G 15A Breaker NG01ACR119	Hydrogen Anal Samp Return to Ctmt Vlv GSHV18
P-4GSY01A 3A Fuse RL011 B-4RLY01G NG02ACR140 15A Breaker	Hydrogen Anal Ctmt Sample Vlv GSHV4
P-4GSY01B 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Hydrogen Anal Ctmt Sample Vlv GSHV5

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-4GSY01C 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Hydrogen and Samp Return to Ctmt Vlv GSHV9
5BBG02B P-2A Fuse PG19NGR3 B-1A Fuse	RCP Oil Lift Pump Control
6BBG02C P-2A Fuse PG20NFF3 B-1A Fuse	RCP Oil Lift Pump Control
6BBG02D P-2A Fuse PG20NFF2 B-1A Fuse	RCP Oil Lift Pump Control
1BBG03A P-2A Fuse NG03CDF4 B 1A Fuse	RCP Thermal Barrier Cooler Isol Vlv BBHV13
1BBG03B P-2A Fuse NG03CHF1 B-1A Fuse	RCP Thermal Barrier Cooler Isol Vlv BBHV14
1BBG03C P-2A Fuse NG03CBF4 B-1A Fuse	RCP Thermal Barrier Cooler Isol Vlv BBHV15
1BBG03D P-2A Fuse NG03CLF2 B-1A Fuse	RCP Thermal Barrier Cooler Isol Vlv BBHV16
5BBK05A P-3A Fuse RL001 B-3A Fuse	RCP Seal Water Return Vlv BBHV8141A
5BBK05B P-3A Fuse RL001 B-3A Fuse	RCP Seal Water Return Vlv BBHV8141B
6BBK05C P-3A Fuse RL001 B-3A Fuse	RCP Seal Water Return Vlv BBHV8141C
6BBK05D P-3A Fuse RL001 B-3A Fuse	RCP Seal Water Return Vlv BBHV8141D
6BBK07A P-3A Fuse RL021 B-3A Fuse	PRT Discharge to RCDT Vlv BBHV8031
6BBK07B P-3A Fuse RL021 B-3A Fuse	Reactor Makeup Water to PRT Vlv BBHV8045

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
6BBK08A P-3A Fuse HB115 B-3A Fuse	RCDT Ht. Exch. to PRT Vlv BBHV7141
6BBK09A P-3A Fuse RL021 B-3A Fuse	RV Flange Leakoff Line to RCDT Vlv BBHV8032
1BBK30C P-3A Fuse RL021 B-3A Fuse	Reactor Vessel Head Vent Vlv BBHV8002A
4BBK30D P-3A Fuse RL021 B-3A Fuse	Rx Vessel Head Vent Vlv BBHV8002B
1BBG31A P-2A Fuse NG01BDF1 B-1A Fuse	PRT Emergency Drain Line Vlv BBHV8037A
4BBG31B P-2A Fuse NG02BHR3 B-1A Fuse	PRT Emergency Drain Line Vlv BBHV8037B
1EPK02D P-3A Fuse RL018 B-3A Fuse	Accumulator Tank D Isol Vlv EPHV8808D Indication
1EPK02E P-3A Fuse RL018 B-3A Fuse	Accumulator Tank A Isol Vlv EPHV8808A Indication
1EPK02F P-3A Fuse RL018 B-3A Fuse	Accumulator Tank C Isol Vlv EPHV8808C Indication
4EPG02C P-2A Fuse NG02BGF3 B-1A Fuse	Accumulator Tank B Isol Vlv EPHV8808B
4EPG02D P-2A Fuse NG02BHF2 B-1A Fuse	Accumulator Tank D Isol Vlv EPHV8808D
4EPK02A P-3A Fuse RL018 B-3A Fuse	Accumulator Tank A Isol Vlv EPHV8808A Indication
4EPK02B P-3A Fuse RL018 B-3A Fuse	Accumulator Tank C Isol Vlv EPHV8808C Indication
4EPK02G P-3A Fuse RL018 B-3A Fuse	Accumulator Tank B Isol Vlv EPHV8808B Indication
4EPK02H P-3A Fuse RL018 B-3A Fuse	Accumulator Tank D Isol Vlv EPHV8808D Indication

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
5EPK03A P-3A Fuse RL018 B-3A Fuse	Accumulator Nitrogen Supply Vlv EPHV8875A
5EPK03B P-3A Fuse RL018 B-3A Fuse	Accumulator Nitrogen Supply Vlv EPHV8875C
5EPK03C P-3A Fuse RL018 B-3A Fuse	Accumulator Test Line Vlv EPHV8877A
4BBK30B P-3A Fuse RL021 B-3A Fuse	Reactor Vessel Head Vent Vlv BBHV8001B
5EPK03D P-3A Fuse RL018 B-3A Fuse	Accumulator Test line Vlv EPHV8877C
5EPK03E P-3A Fuse RL018 B-3A Fuse	Accumulator Water Fill Line Vlv EPHV8878A
5EPK04A P-3A Fuse RL018 B-3A Fuse	Accumulator Water Fill Line Vlv EPHV8878C
5EPK04B P-3A Fuse RL018 B-3A Fuse	Accumulator Test Line Vlv EPHV8879A
5EPK04C P-3A Fuse RL018 B-3A Fuse	Accumulator Test Line Vlv EPHV8879C
6EPK04D P-3A Fuse RL018 B-3A Fuse	Accumulator Test Line Vlv EPHV8879B
6EPK04E P-3A Fuse RL018 B-34 Fuse	Accumulator Test Line Vlv EPHV8879D
6EPK05A P-3A Fuse RL018 B-3A Fuse	Accumulator Nitrogen Supply Vlv EPHV8875B
6EPK05B P-3A Fuse RL018 B-3A Fuse	Accumulator Nitrogen Supply Vlv EPHV8875D
6EPK05C P-3A Fuse RL018 B-3A Fuse	Accumulator Test Line Vlv EPHV8877B
6EPK05D P-3A Fuse RL018 B-3A Fuse	Accumulator Test Line Vlv EPHV8877D

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
6EPK05E P-3A Fuse RL018 B-3A Fuse	Accumulator Water FIll Vlv EPHV8878B
6EPK05F P-3A Fuse RL018 B-3A Fuse	Accumulator Water Fill Vlv EPHV8878D
P-4SJY01B 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Press. Vapor. Cont. Iso. Space Vlv. SJHV12
P-4SJY01C 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Accums Sample Cont Isol Vlv SJHV18
P-5SJY03B 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV16
P-5SJY03C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV17
P-5SJY04B 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV14
P-5SJOY4C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV15
P-1SJY06B 3A Fuse RP332 B-1RPY09F 15A Breaker NG01BAR140	HL Sample 3 Vlv SJHV4
P-4SJY06A 3A Fuse RP333 B-4RPY09F 15A Breaker NG02BAR140	HL Sample 1 Vlv SJHV3

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-5SJY06C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Press Liquid Space Samp Isol Vlv SJHV20
P-4BMY01A 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. A Out to Nuc Sample Sys Vlv BMHV19
P-4BMY01B 3A Fuse RL024 B-4RLY01H 15A Breaker NG2ACR127	S.G. B Out to Nuc Sample Sys Vlv BMHV20
P-4BMY01C 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. C Out to Nuc Sample Sys Vlv BMHV21
P-5GNY08A 3A Fuse RL020 B-5RLY01L 15A Breaker PG19GCR230	CRDM Cooling Discharge Damper GNHZ71
P-5GNY08C 3A Fuse RL020 B-5RLY01L 15A Breaker PG19GCR230	CRDM Cooling Discharge Damper GNHZ73
P-6GNY08A 3A Fuse RL020 B-6RLY01J 15A Breaker PG20GBR222	CRDM Cooling Discharge Damper GNHZ72
P-6GNY08C 3A Fuse RL020 B-6RLY01J 15A Breaker PG20GBR222	CRDM Cooling Discharge Damper GNHZ74
5BGK10A P-3A Fuse RL001 B-3A Fuse	Normal Letdown Isolation Vlv BGLCV459

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
5BGK10B P-3A Fuse RL001 B-3A Fuse	Normal Letdown Isolation Vlv BGLCV460
5BGK19A P-3A Fuse RL001 B-3A Fuse	Pressurizer Spray Vlv BGHV8145
6BGK20A P-3A Fuse RL001 B-3A Fuse	Excess Letdown Line Isolation Vlv BGHV8143
5BGK35A P-3A Fuse RL001 B-3A Fuse	Letdown Orifice Isolation Vlv BGHV8149A
5BGK35B P-3A Fuse RL001 B-3A Fuse	Letdown Orifice Isolation Vlv BGHV8149B
5BGK35C P-3A Fuse RL001 B-3A Fuse	Letdown Orifice Isolation Vlv BGHV8149C
1BGK36A P-3A Fuse RL001 B-3A Fuse	Letdown Containment Isolation Vlv BGHV8160
1BGG38A P-2A Fuse NG01BBF3 B-1A Fuse	Seal Water Ctmt Isolation Vlv BGHV8112
1BGK48C P-3A Fuse RL001 B-3A Fuse	Excess Letdown/RCS Isolation Vlv BGHV8153A
1BGK48D P-3A Fuse RL001 B-3A Fuse	Excess Letdown/RCS Isolation Vlv BGHV8154A
4BGK48A P-3A Fuse RL001 B-3A Fuse	Excess Letdown/RCS Isolation Vlv BGHV8153B
4BGK48B P-3A Fuse RL001 B-3A Fuse	Excess Letdown/RCS Isolation Vlv BGHV8154B
4BBG12A P-2A Fuse NG02BCF2 B-1A Fuse	RHR Pump Suction Isolation Valve BBPV8702A
4BBG12B P-2A Fuse NG02BBF3 B-1A Fuse	RHR Loop 2 Inlet Isolation Valve BBPV8702B
1BBK13A P-3A Fuse RL021 B-3A Fuse	Ctmt Isol Nitrogen Supply to PRT Vlv BBHV8026

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
5BBK14C P-3A Fuse RK021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV180
5BBK14D P-3A Fuse RK021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV181
6BBK14A P-3A Fuse RL021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV178
6BBK14B P-3A Fuse RL021 B-3A Fuse	RCP Standpipe Makeup Vlv BBLCV179
5BBK15B P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
5BBK15C P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
6BBK15D P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
6BBK15E P-3A Fuse RL021 B-3A Fuse	Reactor Coolant Loops Instrumentation
5BBK19A P-3A Fuse RL002 B-3A Fuse	Pressurizer Spray Valve BBPCV455B
5BBK19B P-3A Fuse RL002 B-3A Fuse	Pressurizer Spray Valve BBPCV455C
1BBK30A P-3A Fuse RL021 B-3A Fuse	Reactor Vessel Head Vent Vlv BBHV8001A
6GNG03B P-5A Fuse NG02BJF5 B-3A Fuse	CRDM Cooling Fan B Discharge Isolation Damper GNHZ42
6GNG03D P-5A Fuse PG20GAR2 B-3A Fuse	CRDM Cooling Fan A Discharge Isolation Damper GNHZ41
5GNG04A P-6A Fuse PG19NAF4 B-4A Fuse	Cavity Cooling Fan Discharge Damper GNHZ47

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
6GNG04B P-6A Fuse PG20NBF5 B-4A Fuse	Cavity Cooling Fan Discharge Damper GNHZ48
1HBK03A P-3A Fuse RL021 B-3A Fuse	RCDT Vapor Space CTMT Isol Vlv HBHV7126
6HBK04A P-3A Fuse HB115 B-3A Fuse	RCDT Vapor Space CTMT Isol Vlv HBHV7127
5EPY07B P-3A Fuse RP043 B-15A CB-1	Accumulator Tank Discharge Valve Position Switch EPHV8808DA EPHV8808BA
6EPY07A P-3A Fuse RP044 B-15A CB-1	Accumulator Tank Discharge Valve Position Switch EPHV8808AA EPHV8808CA
6GTY12A P-15A Breaker PG20GBR134 B-20A Fuse	CTMT Minipurge Exhaust Isolation Damper GTHZ41
6GTY12A P-15A Breaker PG20GBR134 B-20A Fuse	CTMT Minipurge Exhaust Isolation Damper GTHZ42
P-5SRY09A 5A Fuse SR057 B-5SRY09A 20A Breaker PG19GEF6	In-Core Neutron Monitoring Drive Unit Heater SR01A, B
P-5SRY09A 5A Fuse SR057 B-5SRY09A 20A Breaker PG19GEF6	In-Core Neutron Monitoring Drive Unit Heater SR01C, D

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MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection and bypass devices, integral with the motor starter of each valve listed in Table 3.8-2, shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

   1) Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

   2) Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

   1) All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years, and

   2) All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE and not bypassed, at least once per 6 years.

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TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)

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3/4.9 REFULEING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

~~4.9.1.3 Valves BG-V178 and BG-V601 shall be verified locked closed and secured in position at least once per 31 days.~~

---

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

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

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Be capable of being closed by an OPERABLE automatic containment purge isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge isolation valves per the applicable portions of Specification 4.6.3.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

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3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine used for movement of fuel assemblies having:

  1) A minimum capacity of 4800 pounds,

  2) Automatic overload cutoffs with the following Setpoints:

    a) Primary - less than or equal to 250 ~~pounds~~ above the indicated suspended weight for wet conditions and 350 ~~100~~ pounds above the indicated suspended weight for dry conditions, and

    b) Secondary - less than or equal to 150 pounds above the Primary overload cutoff.

  3) An automatic load reduction trip with a Setpoint of less than or equal to 250 pounds below the suspended weight for wet conditions ~~and 350 pounds below the suspended weight for~~ or dry conditions.

b. The auxiliary hoist used for latching and unlatching drive rods and thimble plug handling operations having:

  1) A minimum capacity of 3000 pounds, and

  2) at least A 1000 pound load indicator which shall be used to monitor lifting loads for these operation.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for refueling machine and/or auxiliary hoist OPERABILITY not satisified, suspend use of any inoperable refueling machine crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior

Specification 3.9.6.b.2)

Justification -

This specification was revised to be consistent with the load test requirements in 4.9.6.2.

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to ~~the start of such operations~~ removal of the reactor vessel head by performing a load test of at least 125% of the secondary automatic overload cutoff and demonstrating an automatic load cutoff when the refueling machine load exceeds the Setpoints of Specification 3.9.6a.2) and by demonstrating an automatic load reduction trip when the load reduction exceeds the Setpoint of Specification 3.9.6a.3).

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to ~~the start of such operations~~ removal of the reactor vessel head by performing a load test of at least 1250 pounds.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2250 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility.

APPLICABILITY: With fuel assemblies in the spent fuel storage facility.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2250 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

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3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operation [illegible] ing an increase in the reactor decay heat load or a reduction in boron [illegible] tion of the Reactor Coolant System and immediately initiate correcti [illegible] to return the required RHR loop to OPERABLE and operating status as [illegible] possible. Close all containment penetrations providing direct access from [illegible] containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

---

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

---

*Prior to initial criticality, the RHR loop may be removed from operation for up to 1 hour per 2-hour ~~8-hour~~ period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

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3/4.9.9 CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation System inoperable, close each of the purge valves providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specifications 3.0.3 and 3.0 4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge isolation occurs on manual initiation and on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies within the containment when the fuel assemblies being moved are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY: During movement of control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor vessel.

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REFUELING OPERATIONS

3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel ~~storage~~ pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel ~~storage~~ pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel ~~storage~~ pool.

Specification 3.9.11

Justification -

This specification was revised to be consistent with Wolf Creek nomenclature.

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3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 Spent fuel assemblies stored in Region 2 shall be subject to the following conditions:

a. The combination of initial enrichment and cumulative exposure shall be within the acceptable domain of Figure 3.9-1, and

b. No spent fuel assemblies shall be placed in Region 2, nor shall any storage location be changed in designation from being in Region 1 to being in Region 2, while refueling operations are in progress.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The burnup of each spent fuel assembly stored in Region 2 shall be ascertained by [redacted] analysis of its burnup history and independently verified, prior to storage in Region 2. A complete record of such analysis shall be kept for the time period that the spent fuel assembly remains in Region 2 of the spent fuel pool.

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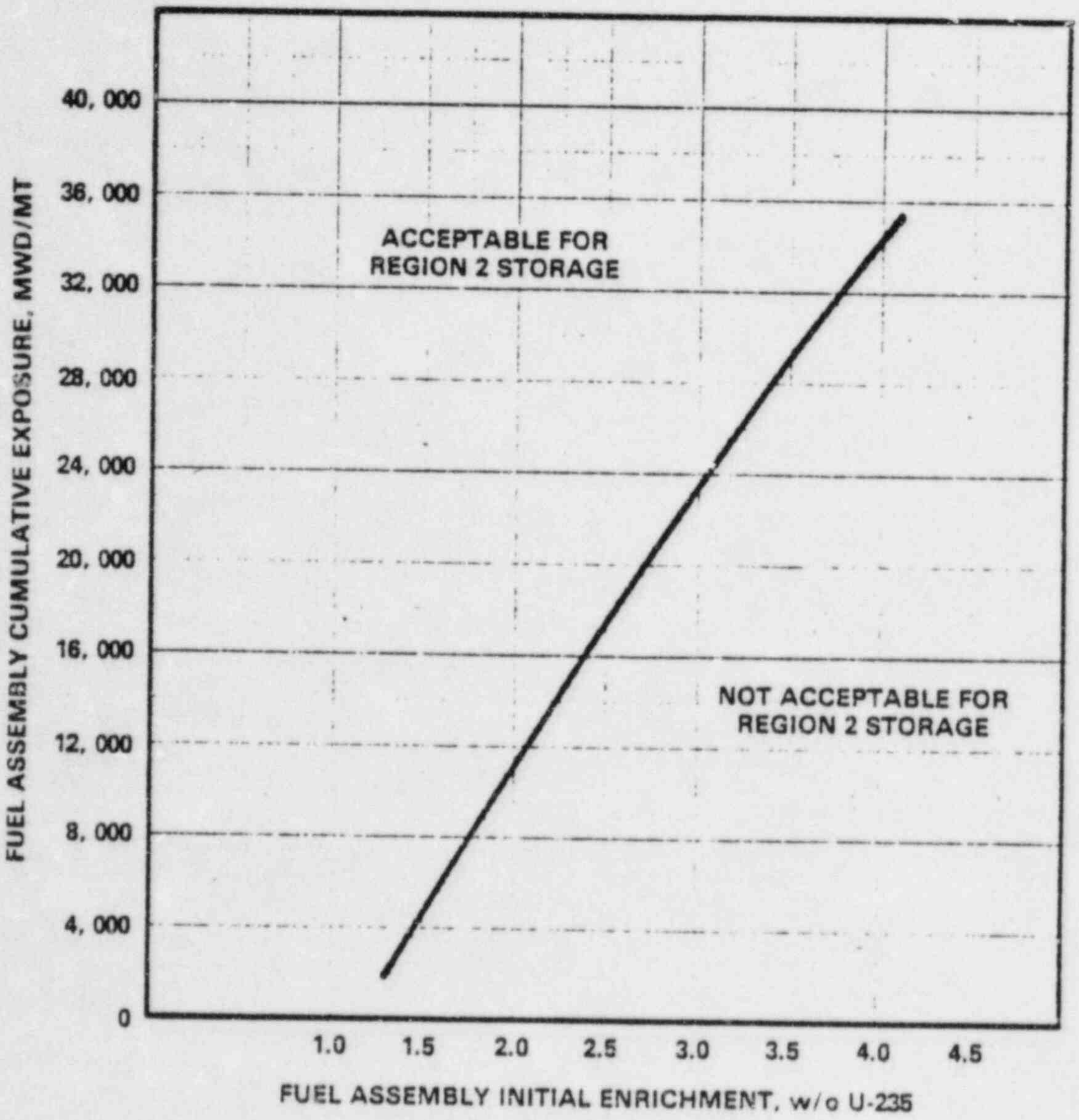

FIGURE 3.9-1

MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

3/4.9.13 EMERGENCY EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.13 Two independent Emergency Exhaust Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the Spent fuel ~~storage~~ pool.

ACTION:

a. With one Emergency Exhaust System inoperable, fuel movement within the fuel storage areas or crane operation with loads over the fuel storage areas may proceed provided the OPERABLE Emergency Exhaust System is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Emergency Exhaust System OPERABLE, suspend all operations involving movement of fuel within the fuel storage areas or crane operation with loads over the fuel storage areas until at least one Emergency Exhaust System is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13 The above required Emergency Exhaust Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months, or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

   1) Verifying that the Emergency Exhaust ~~cleanup~~ System satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 9000 cfm ± 10%;

Specification 3.9.13

Justification -

This specification was revised to be consistent with Wolf Creek nomenclature.

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2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revisison 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.4 inches Water Gauge while operating the system at a flow rate of 9000 cfm ± 10%.

2) Verifying that on a Spent Fuel Pool High Radioactivity test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks and isolates the normal fuel building exhaust flow to the auxiliary/fuel building exhaust fan;

3) Verifying that the system maintains the Fuel Building at a negative pressure of greater than or equal to 1/4 inches Water Gauge relative to the outside atmosphere during system operation; and

4) Verifying that the heaters dissipate 37 ± 3 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ± 10%.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

all capital letters

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2, below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.1.3.7~~, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

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3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6, may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

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3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. ~~3.10.4 The limitations of~~ Specification 3.4.1.1 ~~may be suspended~~ during the performance of STARTUP and PHYSICS TESTS provided:

1). The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

2). The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint, during the performance of startup and PHYSICS TESTS immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during STARTUP and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided only one shutdown or control bank is withdrawn from the fully inserted position at a time.

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements and during surveillance of digital rod position indicators for OPERABILITY.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analysis shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Batch Waste Release Tanks(2) a. Waste Monitor Tank b. Secondary Liquid Waste Monitor Tank	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	5x10^{-7}
			I-131	1x10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1x10^{-5}
	P Each Batch	M Composite(4)	H-3	1x10^{-5}
			Gross Alpha	1x10^{-7}
	P Each Batch	Q Composite(4)	Sr-89, Sr-90	5~~1~~x10^{-8}
			Fe-55	1x10^{-6}
2. Continuous Releases(5) Steam Generator Blowdown	Daily ~~Continuous~~(6) Grab Sample	W Composite(~~6~~) 4	Principal Gamma Emitters(3)	5x10^{-7}
			I-131	1x10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1x10^{-5}
	Daily ~~Continuous~~(6) Grab Sample	M Composite(~~6~~) 4	H-3	1x10^{-5}
			Gross Alpha	1x10^{-7}
	Daily ~~Continuous~~(6) Grab Sample	Q Composite(~~6~~) 4	Sr-89, Sr-90	5x10^{-8}
			Fe-55	1x10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

(2)A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed ~~by a method described in the ODCM~~ to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(3)The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4)A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released. Prior to analysis, all samples taken for the composite shall be thoroughly mixed in order for the composite samples to be representative of the effluent release.

(5)A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

See attached note 6

~~(6)To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.~~

TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(6)Samples shall be taken at the initiation of effluent flow and at least once per ~~day~~ thereafter while the release is occurring. To be representative of the liquid effluent, the sample ~~flow rate~~ shall be proportioned to the ~~rate of flow of~~ the effluent stream. The ratio of sample ~~flow rate~~ to effluent ~~flow rate~~ shall be maintained constant for all samples taken for the composite sample.

24 hours   discharge volume   discharge volume   volume   volume

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Clean Drinking Water Act.*

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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*The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river-sited plants this is 3 miles downstream only.

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-4) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

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LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 150 Curies, excluding tritium and dissolved or entrained noble gases.

a. Reactor Makeup Water Storage Tank,
b. Refueling Water Storage Tank,
c. Condensate Storage Tank, and
d. Outside temporary tanks, excluding demineralizer vessels and the liners being used to solidify radioactive wastes.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added ~~to the tank.~~ and within 7 days following any addition of radioactive material to the tank.

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3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131 and 133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS, RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Waste Gas Decay Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1~~2~~ $\times10^{-4}$
2. Containment purge or vent(3)	P Each PURGE(3) Grab Sample	P Each PURGE(3)	Principal Gamma Emitters(2)	1~~2~~ $\times10^{-4}$
		M	H-3 (oxide)	1×10^{-6}
3.~~2~~. Unit Vent	M(3), (4)(5) Grab Sample	M(3) ~~M~~	Principal Gamma Emitters(2)	1×10^{-4}
		M(4)	H-3 (oxide)	1×10^{-6}
~~4.3.~~ ~~Spent Fuel~~ ~~Radwaste Building Vent Exhaust~~	~~M~~(5) ~~Grab Sample~~	M ~~M~~	~~Principal Gamma Emitters~~(2)	~~1×10^{-4}~~
		~~M(5)~~	~~H-3 (oxide)~~	~~1×10^{-6}~~
4. Radwaste Building Vent	M Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
5~~4~~. All Release Types as listed in ~~1, 2, and 3 above.~~ ~~2 and 3.~~ ~~4 and 5.~~ 1,2,3, and 4	Continuous(5)	W(7) Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous(6)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(6)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(6)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

Table 4.11-2

Justification -

The Spent Fuel Building Exhaust does not belong in this table. The Spent Fuel Building Exhausts into the Unit Vent which is the release point for performing dose calculations. Sampling the Spent Fuel Building for Principal Gamma Emitters does not contribute pertinent information to the dose calculation. The Spent Fuel Building Exhaust is monitored for atmospheric radioactivity. Any unusual activity from the Spent Fuel Building Exhaust will be detected and recorded by the monitors. This item has therefore been deleted from the table.

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt= ~~for plant effluents is~~ the elapsed time between the midpoint of sample collection and the time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2)The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3)Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour period.

(4)Tritium grab samples shall be taken and analyzed at least once per 24 hours when the refueling canal is flooded.

(5)Tritium grab samples shall be taken and analyzed at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool. Grab samples need to be taken only when spent fuel is in the spent fuel pool.

(6)The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7)Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. For unit vent, sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, STARTUP or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

RADIOACTIVE EFFLUENTS

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DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative does contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

DOSE - IODINE-131 AND 133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131 and 133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ, and

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limits and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131 and 133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the WASTE GAS HOLDUP SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity ~~when~~ the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action~~s~~(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and WASTE GAS HOLDUP SYSTEM shall be considered OPERABLE by meeting Specification 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to ~~2%~~ 3% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than ~~2%~~ 3% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a. above.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

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LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ~~2~~ 2.5 x 10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and, within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit ~~at least once per 24 7 days hours when radioactive materials are being added to the tank.~~ per the following:

See insert

Insert

a. Whenever the Xe-133 concentration in the reactor coolant system is less than 4 µCi/cc, a gas storage tank will have the above determination performed once the tank is full and isolated from service.

b. Whenever, the Xe-133 concentration in the reactor coolant system is greater than 4 µCi/cc but less than 40 µCi/cc, each gas storage tank receiving radioactive materials shall be determined to meet the above limit at least once per 7 days and within 7 days following any addition of radioactive material to the tank.

c. Whenever the Xe-133 concentration in the reactor coolant system is greater than 40 µCi/cc, each gas storage tank receiving radioactive materials shall be determined to meet the above limit at least once per 24 hours.

Specification 3.11.2.6

Justification -

This specification has been revised for the following reason. Accidental releases from a Waste Gas Decay Tank can occur through valve packing, instrument fittings and pressure relief valves. The instrument fittings and valve packing leaks are contained in the Waste Gas Decay Tank area serviced by a 500 cfm ventilation sweep. This ventilation flow is monitored by GH-RE-23. GH-RE-23 can be used to estimate concentration of the leakage. Accidental releases through the pressure relief valves will be through the Radwaste Building Ventilation Exhaust. This type of a release is monitored by GH-RE-10. Therefore, accidental releases can be accounted for without taking weekly (or daily) grab samples.

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3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared out-of-service, restore the equipment to operable status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

RADIOACTIVE EFFLUENTS

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6.

c. ~~With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM.~~

See Insert AA

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*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

See Insert AA

~~The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.~~

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

Insert AA

With milk or fresh leafy vegetable samples temporarily unavailable from a routine sampling location, a sample from an alternative location (identified in the ODCM) will be substituted, noting the reason for the unavailability in the Annual Radiological Environmental Operating Report. When changes in sampling locations are permanent, the sample schedule in the ODCM will be updated to reflect the new routine and alternative sampling locations, and this revision will be described in the Annual Radiological Environmental Operating Report.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[(1)]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation[(2)]	40 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY An outer ring of stations, one in each meteorological sector in the 6- to 8-km (3-to 5-mi) range from the site The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly.

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TABLE 3.12-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground level D/Q. One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q. One sample from a control location, as for example 15 to 30 km (10 to 20 mi) distant and in the least prevalent wind direction.(3)	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;(4) and gamma isotopic analysis(5) of composite (by location) quarterly.
3. Waterborne			
a. Surface(6)	One sample upstream. and sample downstream.(6)	Monthly grab sample ~~Composite sample over 1-month period.(7)~~	Gamma isotopic analysis(5) monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources only if likely to be affected.(8)	Quarterly.	Gamma isotopic(5) and tritium analysis quarterly.
c. Drinking	One sample of each of one to three of the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period(7) when I-131 analysis is performed, monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.(9) Composite for gross beta and gamma isotopic analyses(5) monthly. Composite for tritium analysis quarterly.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)			
d. Sediment from shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis(5) semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 5 km distance having the highest dose potential. If there are none, then, one sample from milking animals in each of three areas between 5 to 8 km (3 to 5 mi) distant where doses are calculated to be greater than 1 mrem per yr(9). One sample from milking animals preferably at a control location~~,~~ 15 to 30 km (10 to 20 mi) distant and in the least prevalent wind direction.	Semimonthly when animals are on pasture, monthly at other times.	Gamma isotopic(5) and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
b. Fish ~~and Invertebrates~~	One sample of one to three ~~each commercially and~~ recreationally important species in vicinity of plant discharge area. One sample of similar ~~same~~ species in areas not influenced by plant discharge.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis(5) on edible portions.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
c. Food Products	One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged.	At time of harvest.(10)	Gamma isotopic analyses(5) on edible portion.
	Samples of one to three different kinds of available broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic(5) and I-131 analysis.
	Control ~~One~~ sample of each of the similar broad leaf vegetation preferably grown 15 to 30 km (10 to 20 mi) distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic(5) and I-131 analysis.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1)Specific parameters of distance and direction sector from the centerline of one unit, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. ~~Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979.~~ Deviations will occur ~~are permitted~~ from the required sampling schedule ~~if~~ specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operation Report pursuant to Specification 6.9.1.6. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the selection of the new location(s) for obtaining samples.

(2)One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., ~~at an ocean site,~~ some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

(3)The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

(4)Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(5)Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(6)The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. ~~"Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.~~

(7)A composite sample is one in which the quantity (aliquot) of liquid sampled is ~~proportional to the quantity of flowing liquid~~ constant over the sampling period and in which the method of sampling employed results in a specimen that is representative of the average liquid concentration ~~flow~~. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(8)Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(9)The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(10)If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

	REPORTING LEVELS				
ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
~~Zn-65~~	~~300~~		~~20,000~~		
Zr-Nb-95	400**				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200**			300**	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCI/ℓ may be used.

** Total activity, parent plus daughter activity

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1)

LOWER LIMIT OF DECTECTION (LLD)(2,3)

ANALYSIS	WATER (pCi/ℓ)	AIRBORNE PARTICULATE OR GAS (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/ℓ)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
~~Zn-65~~	~~30~~		~~260~~			
Zr-Nb-95	15**					
I-131	1(4)	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15**			15**		

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*~~If no drinking water pathway exists~~ for surface water samples, a value of 3000 pCi/l may be used.

** Total activity, parent plus daughter activity

TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1)This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6..

(2)Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$\mathrm{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between sample collection, or end of the sample collection period, and time of counting (s).

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

(4)LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m² (500 ft²) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this Land Use Census was conducted. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

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*Broad leaf vegetation sampling of available ~~at least three different kinds of~~ vegetation may be performed at the SITE BOUNDARY in each of two different-direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1, Part 4.c. shall be followed, including analysis of control samples.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

- a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

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BASES FOR
SECTIONS 3.0 AND 4.0
LIMITING CONDITIONS FOR OPERATION
AND
SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and [illegible] ~~those~~ stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act, shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS subsystems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Containment Spray System is inoperable. Under the requirements of Specification 3.0.3 if both the required Containment Spray Systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that allowed ~~provided~~ for operation in subsequent lower OPERATIONAL MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example, if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN. However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE, and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when STARTUP with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under ~~this~~ these criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements. Items may be determined inoperable during use, during surveillance tests or in accordance with this specification. Therefore, ACTION statements are entered when the Surveillance Requirements should have been performed rather than at the time it is discovered that the tests were not performed.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other

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applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition ~~for~~ of Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

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 subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} Δk/k/°F. The MTC value of -3.2×10^{-4} Δk/k/°F 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 making these corrections to the limiting MTC value of -4.1×10^{-4} Δk/k/°F.

The Surveillance Requirements 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.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boration Systems ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature ~~above 200°F~~ equal to or greater than 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 ~~10,117~~ gallons of 7000 ppm borated water from the boric acid storage tanks or 83,745 ~~72,096~~ gallons of 2000 ppm borated water from the RWST. With the RCS average temperature less than 350°F, only one boron injection flow path is required.

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boration System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV, or an RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either ~~2713~~ 2468 gallons of 7000 ppm borated water from the boric acid storage tanks or ~~12117~~ 14,016 gallons of 2000 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boration System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

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MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (b) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$	Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;
$F^N_{\Delta H}$	Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and
$F_{xy}(Z)$	Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4.2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position,

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

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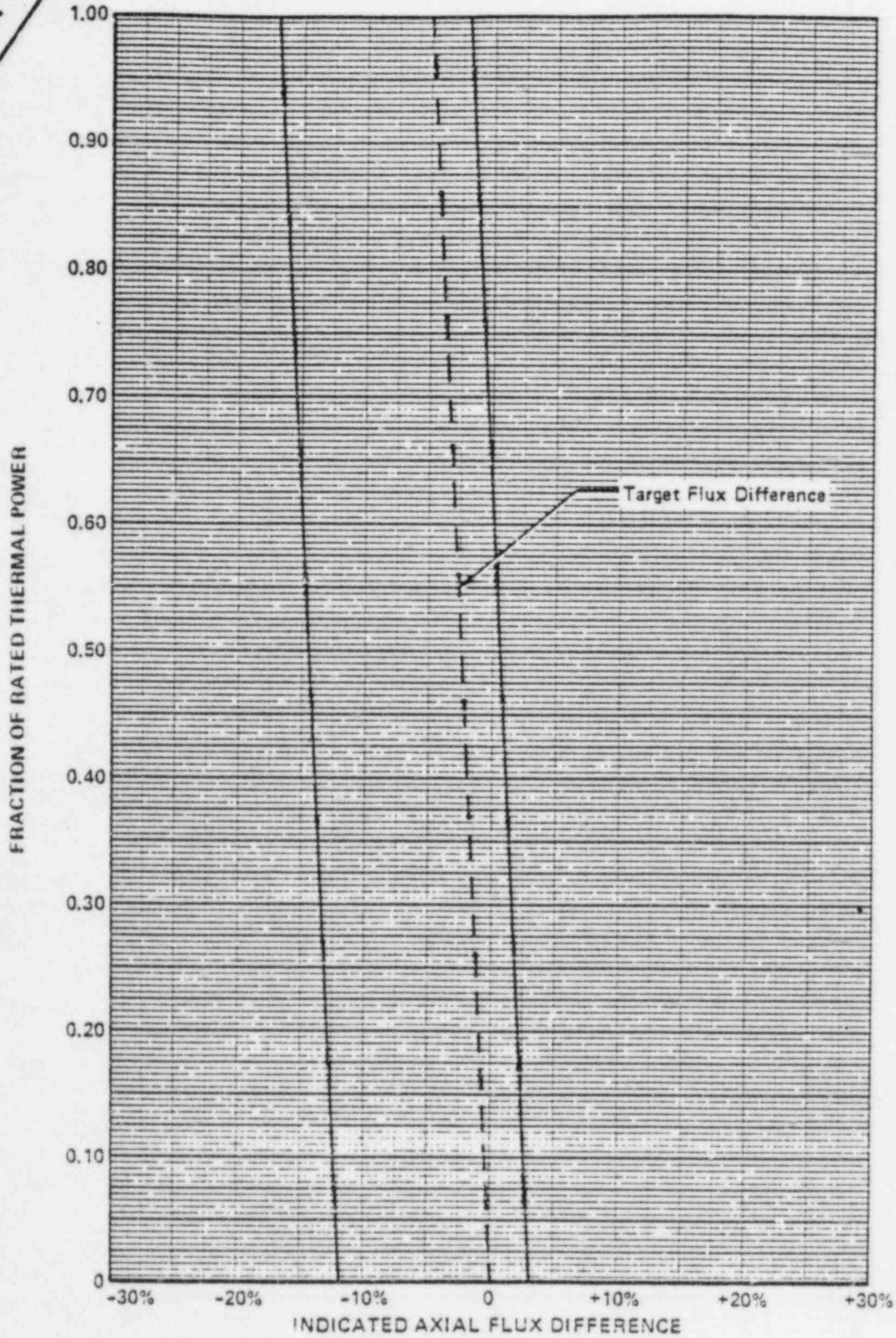

FIGURE B 3/4.2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

~~Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% when the analysis is performed with the approved interim methods. The margin used to partially offset rod bow penalties is 9.1%. This margin breaks down as follows:~~

See insert, following page

~~1) Design limit DNBR (1.6)%~~

~~2) Grid spacing K_s (2.9)%~~

~~3) Thermal Diffusion Coefficent (1.2)%~~

~~4) DNBR Multiplier (1.7)%~~

~~5) Pitch Reduction (1.7)%~~

~~The margin used to partially offset rod bow penalties is (5.9)% with the remaining (3.2)% used to trade off against measured flow which may be as much as (2)% lower then thermal design flow plus uncertainties.~~

~~The penalties applied to $F^N_{\Delta H}$ to account for rod bow as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 with the difference being due to the amount of margin each unit uses to partially offset rod bow penalties.~~

INSERT

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28
2) Grid Spacing (K_s) of 0.046 vs. 0.059
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059
4) DNBR Multiplier of 0.86 vs. 0.88
5) Pitch reduction

The applicable values of rod bow penalties are referenced in the FSAR.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of ~~2%~~ 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The Quadrant Power Tilt Ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective ACTION is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore

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QUADRANT POWER TILT RATIO (Continued)

flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of 592.5°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595°F and 2205 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either

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REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statisitical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trips (3) Feedwater System isolat~~ion~~, (4) s~~tartup of~~ the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolat~~ion~~, (7) steam line isolat~~ion~~, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

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Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates safety injection actuation on low pressurizer pressure and low steamline pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steamline pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated ACTION will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and acutation of Emergency Exhaust or Ventilation Systems.

Controll Room Emergency

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

Neutron Flux

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3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 3, and 19 and Appendix R of 10 CFR Part 50. [illegible] that a fire will not preclude achieving safe shutdown. The Remote Shutdown System transfer switches, power circuits, and control circuits are independent of areas where a fire could damage systems normally used to shutdown the reactor.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision ~~3,~~ 2 "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," ~~May 1983~~ December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection System ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977.

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3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for the prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the reactor Coolant system and avoid or mitigate damage to reactor Coolant system components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 o Appendix A to 10 CFR Part 50. ~~The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.~~

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3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1 x 10^{-6} μCi/cc are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Although the orientation of the turbine is such that number of potentially damaging missiles which could impact and damage safety-related components, equipment, or structures is minimal, protection from excessive turbine overspeed is required.

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within ~~1~~ 6 hours.

In MODE 3, ~~a single~~ two reactor coolant loops provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that ~~two~~ three loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by ~~either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2) by~~ restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

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3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip ~~until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load)~~ and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

~~The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions~~. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

See insert, following page

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of ~~53%~~ 48% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Results from WCAP-10043 have been used to establish plugging limit.

Insert

Unscheduled inservice inspections are performed on each steam generator following; 1) primary to secondary tube leaks; 2) seismic occurrence greater than the Operating Basis Earthquake; 3) a loss of coolant accident requiring actuation of the Engineered Safety Features, which for this specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulator Guide 1.121 which unplugged steam generator tubes must be capable of withstanding.

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STEAM GENERATORS (Continued)

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be ~~promptly~~ reported to the Commission pursuant to Specification 6.9.~~1~~2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that stema generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow from ~~supplied to~~ the reactor coolant pump seals exceeds 8 gpm per RC pump ~~40 gpm with the modulating valve in the supply line fully open~~ at a nominal RCS pressure of 2235 psig. This limitation ensures ~~that in the event of a LOCA, the Safety Injection flow will not be less than assumed in the safety analyses.~~ adequate performance of the RC pump seals.

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OPERATIONAL LEAKAGE (Continued)

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since those valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Wolf Creek Generating Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

SPECIFIC ACTIVITY (Continued)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1 microCurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1 microCurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective ACTION. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

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PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 583°F, ~~and~~

5. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 16 effective full power years (EFPY) of service life. The 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials" or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the

PRESSURE/TEMPERATURE LIMITS (Continued)

surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = ~~is~~ the stress intensity factor caused by membrane (pressure) stress, PRESSURE/TEMPERATURE LIMITS (Continued)

K_{It}: ~~is~~ the stress intensity factor caused by the thermal gradients,

K_{IR}: ~~is provided by the code as~~ a function of temperature relative to the RT_{NDT} of the material, as provided by the Code,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	Cu (%)	P (%)	T_{NDT} (°F)	50 FT-LB 35 Mil Temp (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF NMWD* (FT-LB)	AVG. UPPER SHELF MWD** (FT-LB)
Closure Head Dome	R2516-1	A533B, CL.1	0.12	0.010	-40	60	0	112	----
Closure Head Torus	R2515-1	A533B, CL.1	0.11	0.009	-20	<40	-20	119	----
Closure Head Flange	R2504-1	A508, CL.2	----	0.013	20	<80	20	139	----
Vessel Flange	R2501-1	A508, CL.2	----	0.012	20	<80	20	102	----
Inlet Nozzle	R2502-1	A508, CL.2	----	0.010	-20	<40	-20	147	----
Inlet Nozzle	R2502-2	A508, CL.2	----	0.009	-20	<40	-20	137	----
Inlet Nozzle	R2502-3	A508, CL.2	0.11	0.010	-20	<40	-20	156	----
Inlet Nozzle	R2502-4	A508, CL.2	0.11	0.010	-30	<30	-30	156	----
Outlet Nozzle	R2503-1	A508, CL.2	----	0.006	-10	<50	-10	126	----
Outlet Nozzle	R2503-2	A508, CL.2	----	0.009	0	<60	0	129	----
Outlet Nozzle	R2503-3	A508, CL.2	----	0.007	0	<60	0	136	----
Outlet Nozzle	R2503-4	A508, CL.2	----	0.007	0	<60	0	114	----
Nozzle Shell	R2004-1	A533B, CL.1	0.05	0.010	-40	70	10	118	----
Nozzle Shell	R2004-2	A533B, CL.1	0.04	0.011	-40	80	20	121	----
Nozzle Shell	R2004-3	A533B, CL.1	0.04	0.008	-50	60	0	133	----
Inter. Shell	R2005-1	A533B, CL.1	0.04	0.008	-20	<40	-20	127	156
Inter. Shell	R2005-2	A533B, CL.1	0.04	0.007	-30	40	-20	127	143
Inter. Shell	R2005-3	A533B, CL.1	0.05	0.007	-30	40	-20	135	164
Lower Shell	R2508-1	A533B, CL.1	0.09	0.009	-40	60	0	87	118
Lower Shell	R2508-2	A533B, CL.1	0.06	0.008	-10	70	10	100	127
Lower Shell	R2508-3	A533B, CL.1	0.07	0.008	-20	100	40	86	127
Bottom Head Torus	R2517-1	A533B, CL.1	0.11	0.010	-80	30	-30	92	----
Bottom Head Dome	R2518-1	A533B, CL.1	0.12	0.009	-60	0	-60	154	----
Inter. & Lower Shell Long. Weld Seams	G2.06	SAW	0.04	0.006	-50	<10	-50	150	----
Inter. to Lower Shell Long. Weld Seams	F3.16	SAW	0.05	0.007	-50	10	-50	98	----

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* NMWD - Normal to Major Working Directions

** MWD - Major Working Directions

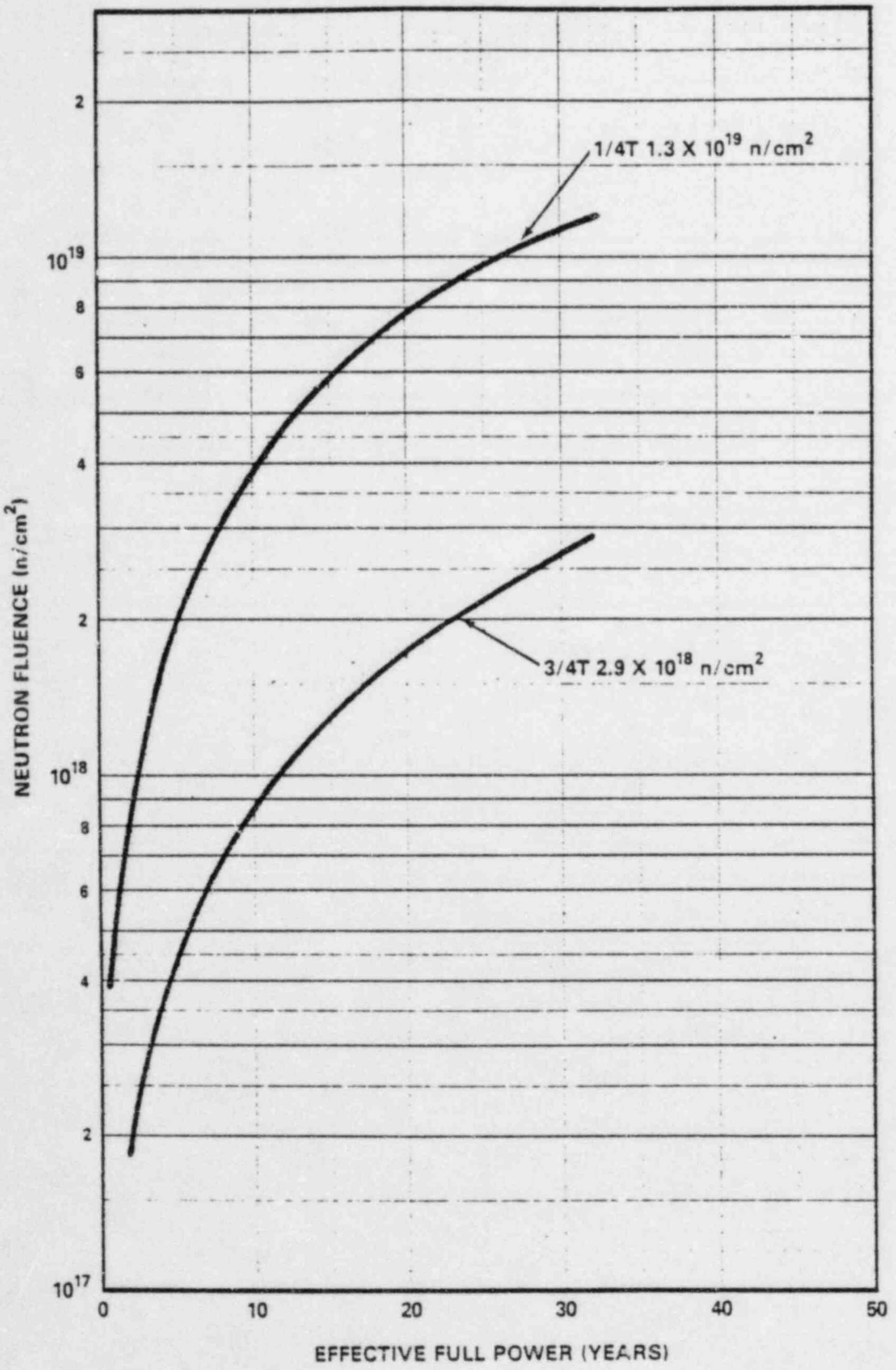

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION EFFECTIVE FULL POWER LIFE

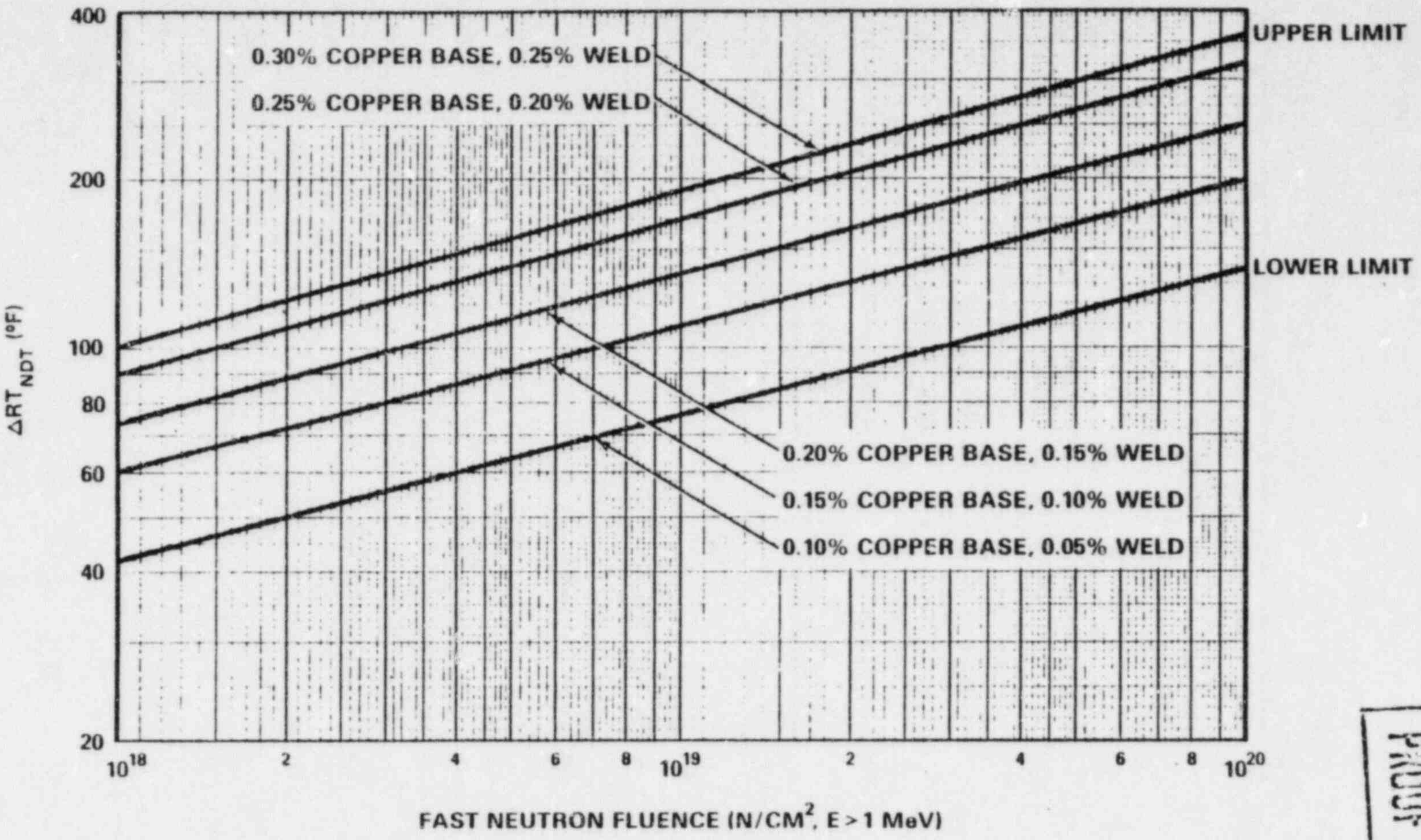

FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER ON SHIFT OF RT_{NDT} FOR

REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F

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PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

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PRESSURE/TEMPERATURE LIMITS (Continued)

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction relief valves, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

insert >

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

insert

RHR RCS suction isolation valves 8701A and B are interlocked with an "A" train wide range pressure transmitter and valves 8702A and B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701B and 8702A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths.

In addition to opening RCS vents to meet the requirement of Specification 3.4.9.3c., it is acceptable to remove a pressurizer Code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

COLD OVERPRESSURE

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; 3) instrument uncertainties; and 4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, technical specifications require lockout of both safety injection pumps and all but one centrifugal charging pump while in MODES 4, 5 and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature. Exceptions to these mode requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centifugal charging pump OPERABLE and no safety injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis LOCA's occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and safety injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic safety injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of safety injection (one centrifugal charging pump, and one safety injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORV's without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of safety injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

insert (continued)

Although COMS is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both safety injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent safety injection occur above 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F ~~one~~ RCPs two and all pressurizer safety valves are required to be OPERABLE. Operation of an RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the pressurizer safety valves provide acceptable and redundant overpressure protection.

The Maximum Allowed PORV setpoint for the Cold Overpressure Mitigation System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H and in accordance with the schedule in Table 4.4-5.

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3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumlators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2, 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

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ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump ~~and one Safety Injection pump~~ to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR Suction relief valve.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure~~s~~, that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensures that a failure of one valve will not cause an intersystem LOCA. The Surveillance Requirement to vent the ECCS pump casings and accessible, if it can be reached without personnel hazard or high radiation dose, discharge piping ensures against inoperable pumps caused by gas binding or water hammer in ECCS piping.

~~3/4.5.5 BORON INJECTION SYSTEM~~

~~The OPERABILITY of the Boron Injection System as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.~~

~~The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.~~

3/4.5.~~6~~5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water

REFUELING WATER STORAGE TANK (Continued)

volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the ~~safety~~ accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

insert >

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

Insert

For reduced pressure tests, the leakage characteristics yielded by measurements L_{tm} and L_{am} shall establish the maximum allowable test leakage rate L_t of not more than L_a (L_{tm}/L_{am}). In the event L_{tm}/L_{am} is greater than 0.7, L_t shall be specified as equal to $L_a (P_t / P_a)^{1/2}$

Bases 3/4.6.1.2

Justification -

The bases has been revised to be consistent with 10CFR50, Appendix J.

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3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig, and (2) the containment peak pressure does not exceed the design pressure of 60 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 48 psig. The limit of ~~2~~ 1.5 psig for initial positive containment pressure will limit the total pressure to ~~50~~ 49.5 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained in accordance with safety analysis requirements for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 50 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," ~~January 1976~~, and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979. April 1979

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerance on cracking, the results of the engineering evaluation and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be closed and blank flanged during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed and blank flanged during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 36-inch containment valves cannot be inadvertently opened, the valves are blank flanged ~~sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.~~

The use of the containment mini-purge lines is restricted to the 18-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 18-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation will be limited to ~~500~~ 2000 hours during a calendar year. The total time the Containment Purge (vent) System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons, e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, ~~may~~ should be used to support the additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4 in any calendar year regardless of the allowable hours.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the

SPRAY ADDITIVE SYSTEM (Continued)

solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. The educator flow test of 52 gpm with RWST water is equivalent to 40 gpm NaOH solution. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the Purge System) is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. The Hydrogen Purge Subsystem discharges directly to the Emergency Exhaust System. Operation of the Emergency Exhaust System with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a ~~LOCA," March 1971~~.

Loss-of-Coolant Accident" Revision 2, November 1978.

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3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary Coolant System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 102% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, (1971 Edition). The total relieving capacity for all valves on all of the steam lines is (18.28×10^6) lbs/h which is 115% of the total secondary steam flow of 15.85×10^6 lbs/h at 102% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109).$$

For three loop operation:

$$SP = \frac{(X) - (Y)(U)}{X} \times (*).$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

U = Maximum number of inoperable safety valves per operating steam line,

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SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. (This value left blank pending NRC approval of three loop operation),

X = Total relieving capacity of all safety valves per steam line in lbs/hour.

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Testing of

on a [illegible] of the recirculation flow with
ensuring a discharge pressure of greater than
equal to 1535 psig verifies the capability of each pump to deliver

Each electric motor-driven auxiliary feedwater pump ~~is capable of delivering~~ a total feedwater flow of 575 gpm at a pressure of 1221 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 1145 gpm at a pressure of 1221 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power and then a cooldown to 350°F at 50°F per hour. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses. Each independent CCW loop contains two 100% capacity pumps and, therefore, the failure of one pump does not affect the OPERABILITY of that loop.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

The OPERABILITY of the Essential Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analysis.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available either to: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

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ULTIMATE HEAT SINK (Continued)

from the Essential Service Water Pumps

The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

charcoal

3/4.7.7 EMERGENCY EXHAUST SYSTEM

The OPERABILITY of the Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

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Bases 3/4.7.5

Justification -

The words "from the Essential Service Water Pumps" were added to the bases to clarify that fact that the safety analysis for the Ultimate Heat Sink was based on maximum temperature and minimum water level at the Essential Service Water Pumps not the Circulating Water Pumps.

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3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. ~~Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system~~.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer. Snubbers may also be classified and grouped by inaccessible or accessible for visual inspection purposes. Therefore, each snubber type may be grouped for inspection in accordance with accessibility.

A list of individual snubbers with detailed information of snubber location and size and of systems affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Safety Review Committee. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing. Since the visual

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SNUBBERS (Continued)

inspections are augmented by functional testing program, the visual inspection need not be a hands on inspection, but shall require visual scrutiny sufficient to assure that fasteners or mountings for connecting the snubbers to supports or foundations shall have no visible bolts, pins or fasteners missing, or other visible signs of physical damage such as cracking or loosening.

To provide assurance of snubber functional reliability, one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or
2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or
3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Insert >

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Insert

For mechanical snubbers the force required to initiate or maintain motion of the snubber is not great enough to overstress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.

Bases 3/4.7.8

Justification

The insert was added to the bases to clarify that drag force below a specified range is not required to determine a snubbers ability to reduce the effects of seismic events. The snubbers need only allow the component pipe to move without overstressing the component/pipe.

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SEALED SOURCE CONTAMINATION (Continued)

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, spray, and/or sprinklers, Halon, and fire hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method,

insert

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.11 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection of and the extinguishing of the fire. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

insert

or by ultrasonic measurement corrected for temperature using equipment calibrated to standards traceable to NBS. The term "simulated fire" test signal is interpreted to mean actuation of an automatic Fire Protection System by any of the release mechanisms provided, e.g., fire detectors, hand pull stations, fusible link/mechanical, manual, hydro/mechanical, etc.

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FIRE BARRIER PENETRATIONS (Continued)

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±3°F.

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3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources", December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability are ~~is~~ available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

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A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than ~~2.05~~ 2.13 volts and ~~0.010~~ 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than ~~2.05~~ 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than ~~0.050~~ 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than ~~1.99~~ 2.07 volts, ensures the battery's capability to perform its design function.

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3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

~~The OPERABILITY of the motor operated valves thermal overload protection and bypass devices ensures that these devices will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.~~

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3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on Keff of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations. ~~The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.~~

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The OPERABILITY of this system ensures the containment purge penetrations will be automatically isolated upon detection of high radiation levels within containment. The OPERABILITY of this system is required to restrict the release of radioactive materials from the containment atmosphere to the environment.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

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3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies stored in Region 2 of the spent fuel pool ensure inadvertent criticality will not occur.

3/4.9.13 EMERGENCY EXHAUST SYSTEM

The limitations on the Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity with automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

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3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM-SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements.

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3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

~~This specification applies to the release of radioactive materials in liquid effluents from all units at the site~~.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I ~~that~~ conformance with the guides

which specify that

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DOSE (Continued)

of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

~~This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing~~

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LIQUID RADWASTE TREATMENT SYSTEM (Continued)

~~units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. ~~Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM.~~ The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000mrems/year to the skin. These release rate limits also restrict, at all times the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

~~This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.~~

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DOSE RATE (Continued)

The required detection capabilities for radioactive materials in ~~liquid~~ gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in ~~liquid~~ gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive materials in ~~liquid~~ gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1," July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon either the historical average real time atmospheric conditions.

~~This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

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3/4.11.2.3 DOSE - IODINE-131 AND 133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon either the historical or real time average atmospheric conditions. The release rate specifications for Iodine-131 and 133, tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

~~This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The

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GASEOUS RADWASTE TREATMENT SYSTEM (Continued)

requirement that the appropriate portions of this system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the ~~Liquid Radwaste Treatment~~ Systems were specified as a suitable fraction of the dose design objectives set forth in Section II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

~~This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.~~

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen. ~~automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.)~~ Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. ~~to a quantity that is less than the quantity that provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem, the annual dose limit in 10 CFR Part 20.~~

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the

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GAS STORAGE TANKS (Continued)

tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to, waste type, waste pH, waste/liquid/SOLIDIFICATION agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and the radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and from outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program given in the ODCM are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities or other acceptable sources shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identfied and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To

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LAND USE CENSUS (Continued)

determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m².

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

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SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4. The definition of UNRESTRICTED AREA used in implementing the Radiological Effluent Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the LIMITING CONDITIONS FOR OPERATION to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

lower case

lower case

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet,

b. Nominal inside height = 205 feet,

c. Nominal thickness of concrete walls = 4 feet,

d. Nominal thickness of concrete dome = 3 feet,

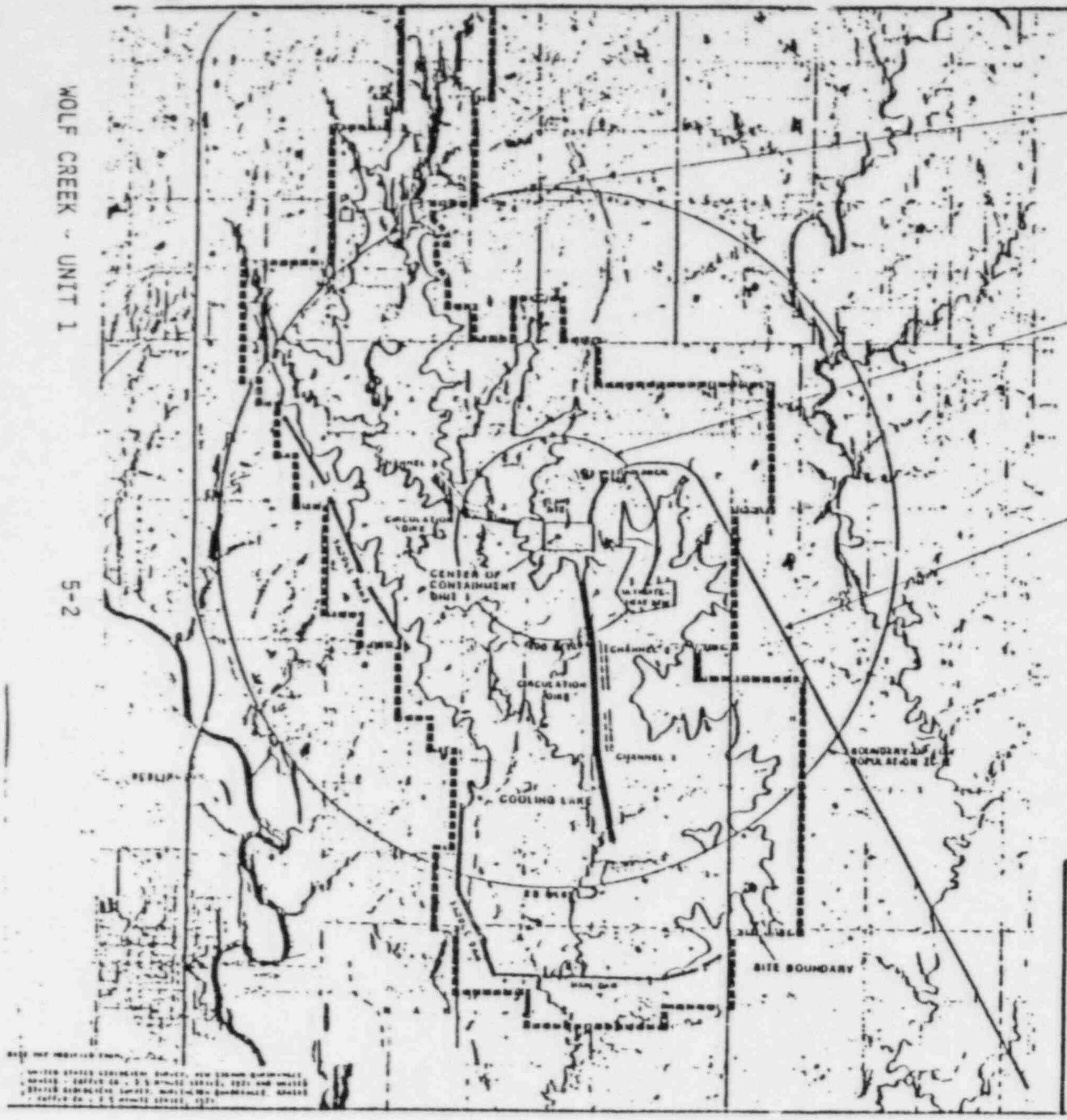
e. Nominal thickness of concrete base slab = 10 feet,

f. Nominal thickness of steel liner = 0.25 inch, and

g. Net free volume = 2.5 x 10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 60 psig and a temperature of 320°F.



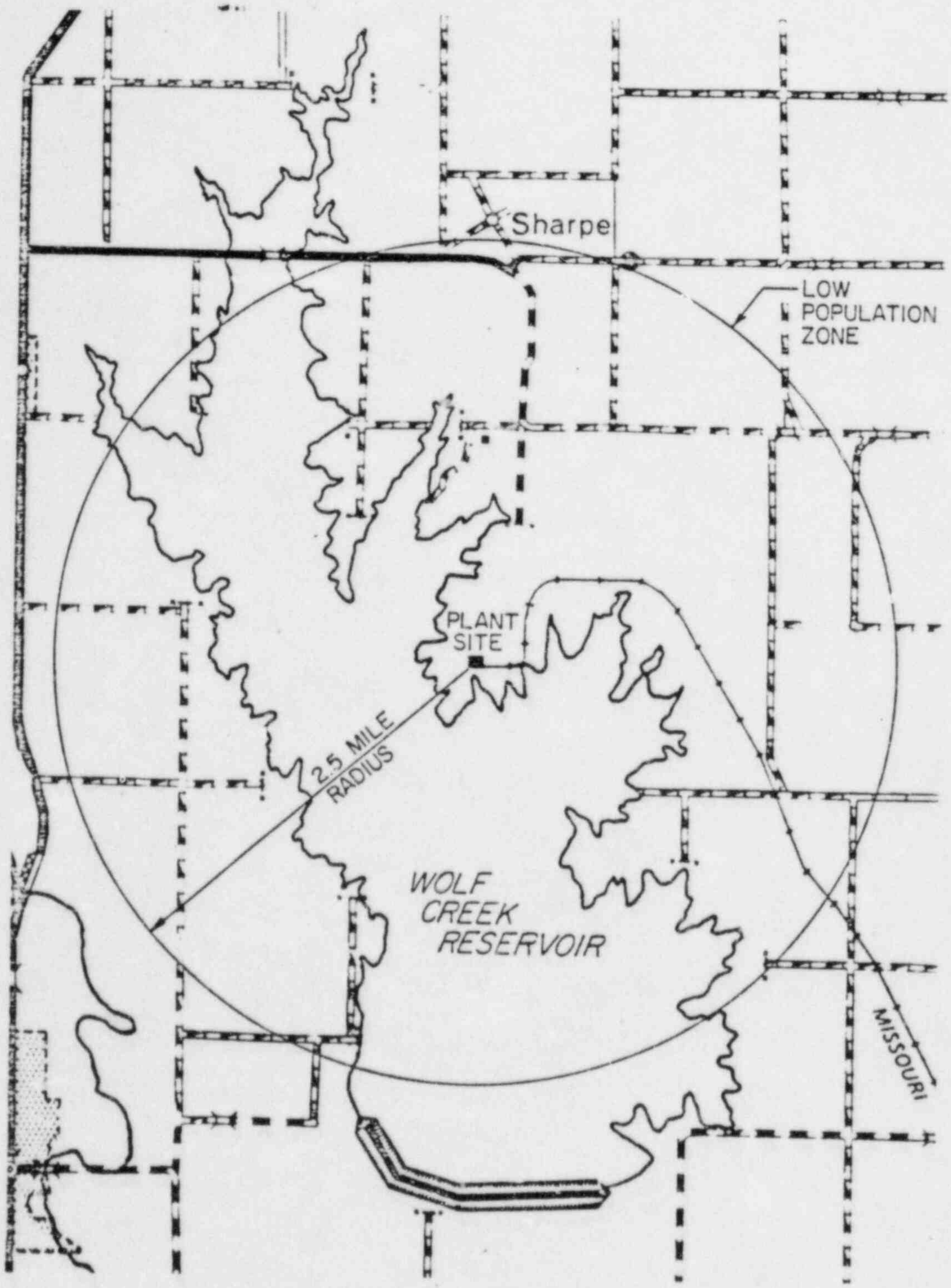
Note:

1. THE EXCLUSION -RESTRICTED AREA IS A 1200 METER RADIUS CIRCLE CENTERED AROUND UNIT 1 CONTAINMENT.
2. UNITED TELEPHONE COMPANY AND RURAL ELECTRIC COOPERATIVE 69-KV EASEMENTS IN THE EXCLUSION AREA ARE NOT SHOWN.

WOLF CREEK GENERATING STATION
UNIT NO. 1
FINAL SAFETY ANALYSIS REPORT

FIGURE 5.1-1

EXCLUSION AREA



WOLF CREEK GENERATING STATION
UNIT NO. 1
FINAL SAFETY ANALYSIS REPORT

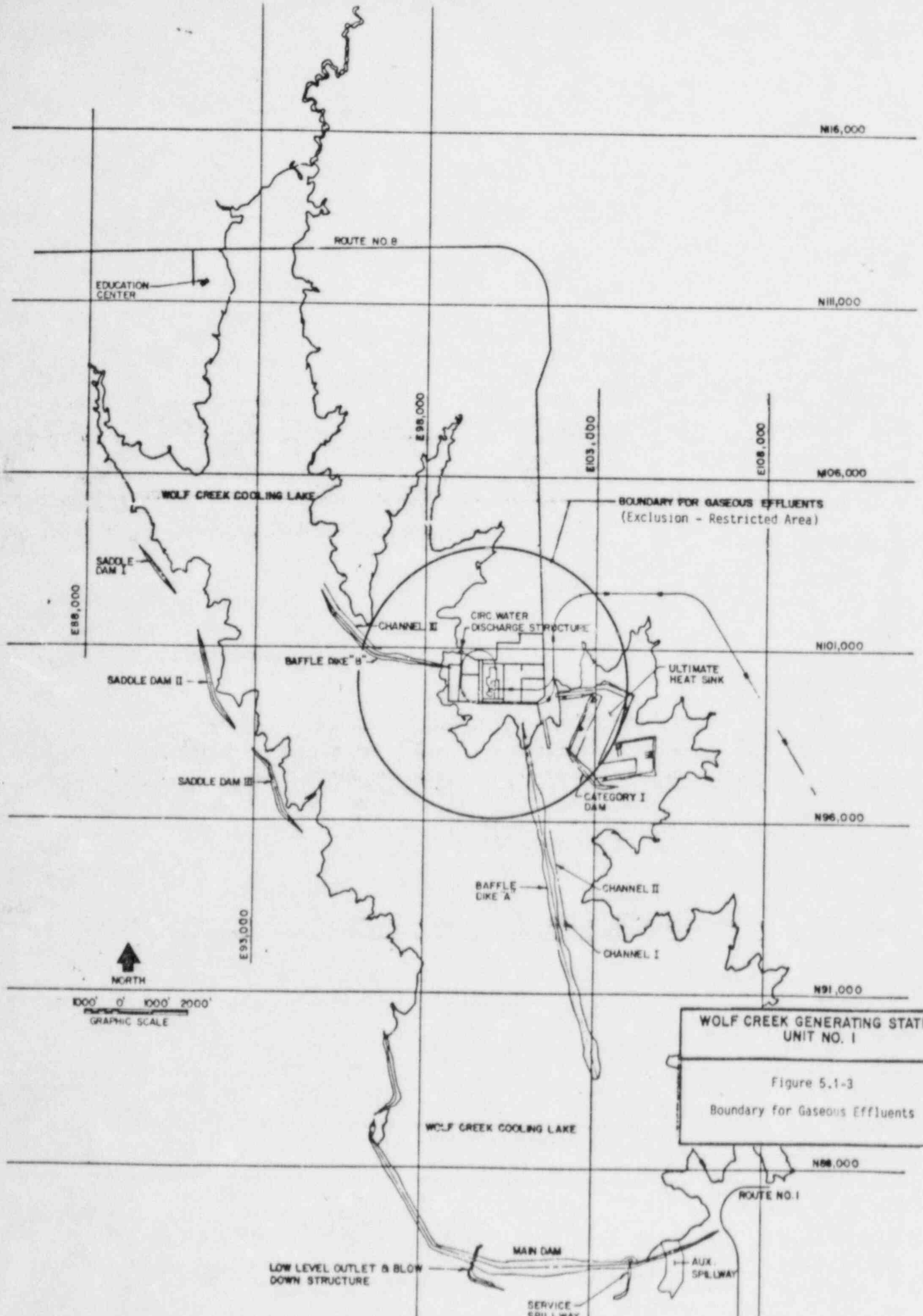
FIGURE 5.1-2

LOW POPULATION ZONE

Figure 5.1-2

Justification -

This figure was revised for clarity.



WOLF CREEK GENERATING STATION
UNIT NO. 1

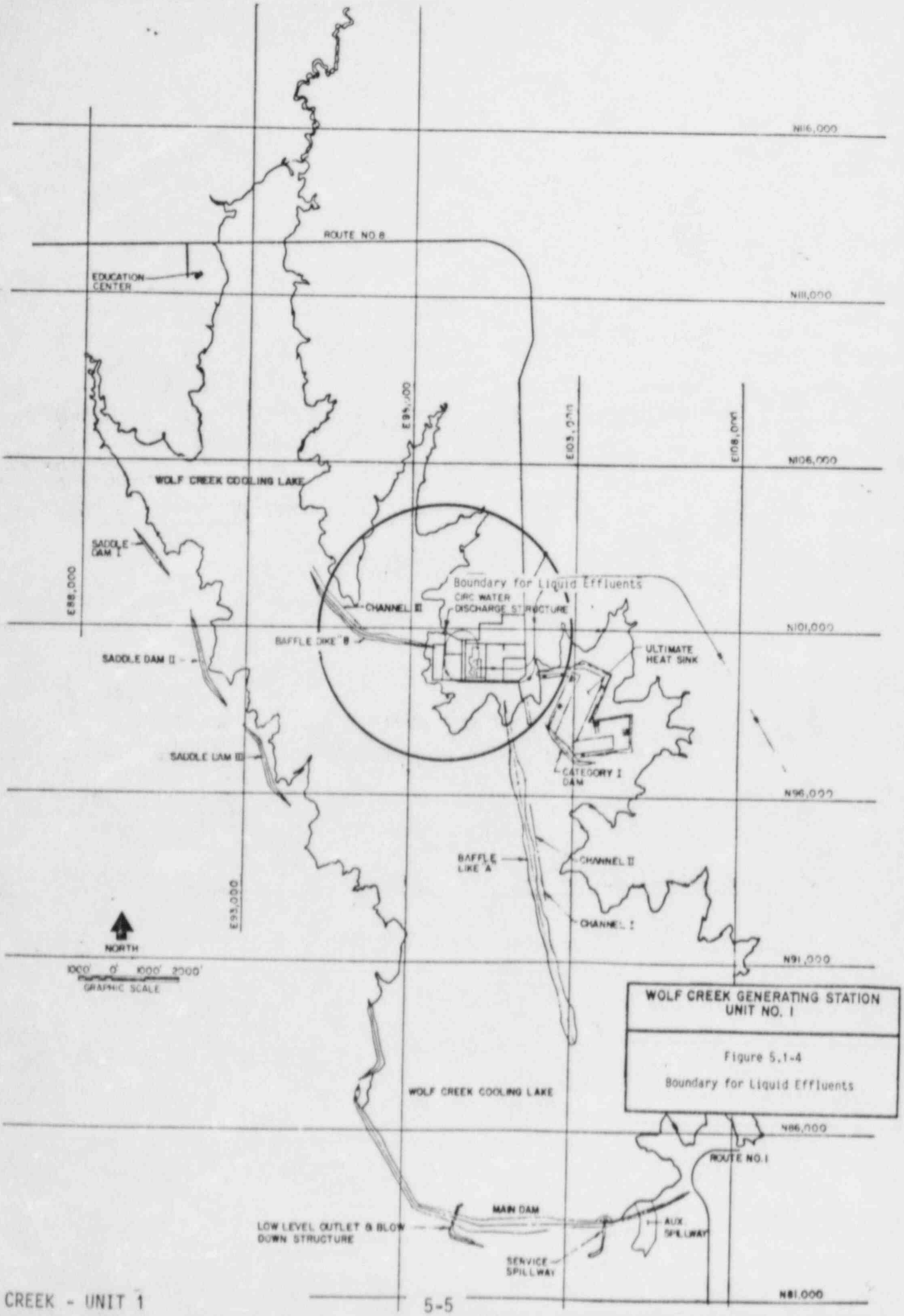
Figure 5.1-3

Boundary for Gaseous Effluents

Figure 5.1-3

Justification -

The figure was revised for clarity.



WOLF CREEK GENERATING STATION UNIT NO. 1

Figure 5.1-4

Boundary for Liquid Effluents

Figure 5.1-4

Justification -

The figure was revised for clarity.

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of ~~143.7~~ 144 inches and contain a maximum total weight of ~~1735~~ 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.10 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading a. ' shall have a maximum enrichment of 3.50 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length and no part-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. All control rods shall be hafnium, clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total volume of the Reactor Coolant System, including pressurizer and surge line, is 12,135 ± 100 cubic feet at a nominal T_{avg} of 557°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

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5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.6% $\Delta k/k$ for uncertainties as described in Section 4.3 of the FSAR. This is based on new fuel with an enrichment of 3.50 weight percent U-235 in Region 1 and on spent fuel with combination of initial enrichment and discharge exposures, shown in Figure 5.6-1, in Region 2; and

b. A nominal 9.14 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 2040 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1344 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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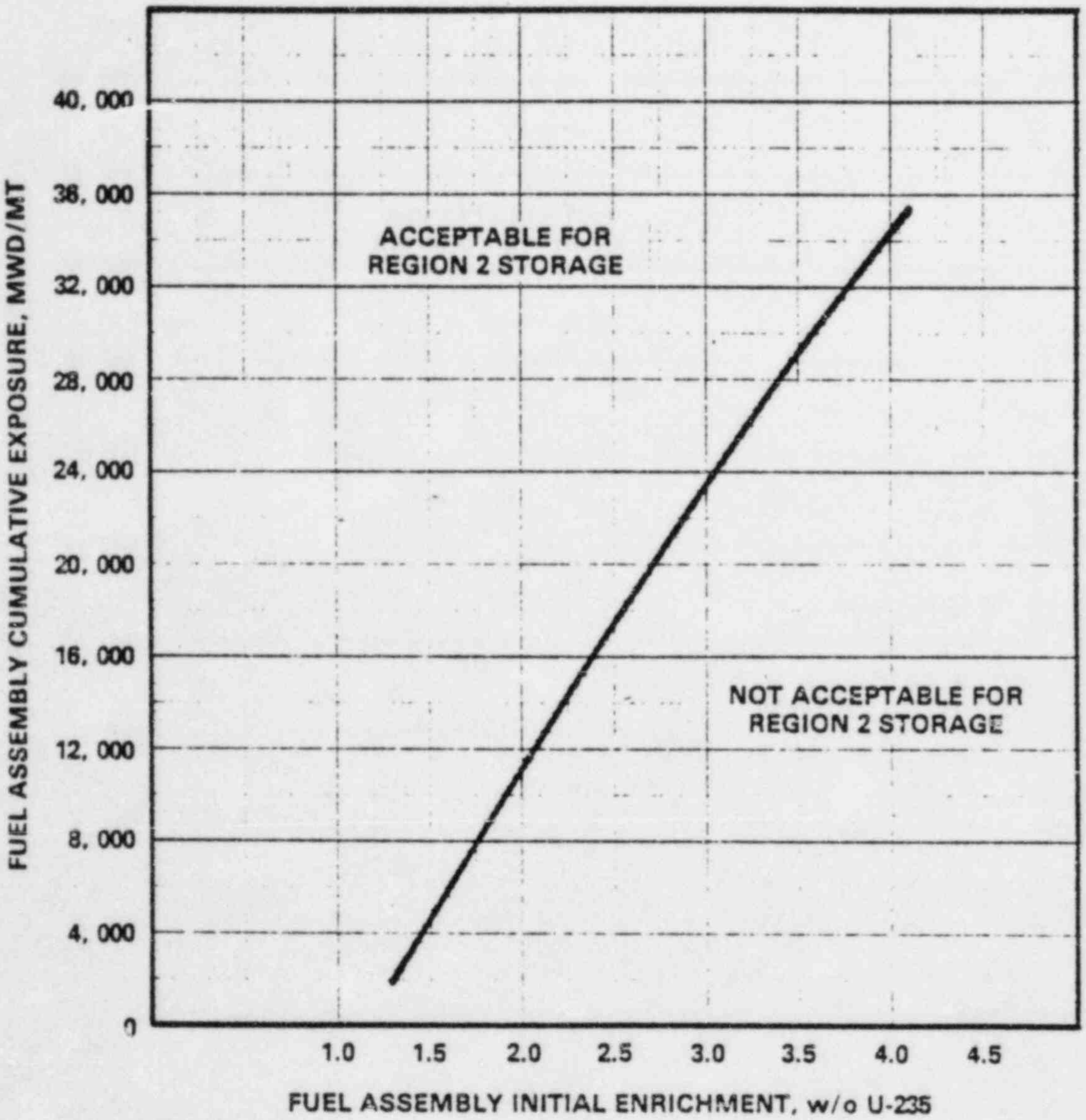

FIGURE 5.6-1

MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at < 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to > 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray acutation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3106 psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line.
	5 hydrostatic pressure tests.	Pressurized to ≥ 1350 psig.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall Unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Supervising Operator, under the Shift Supervisor, shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President-Nuclear shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The Unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the Unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the control room;

c. An individual from the Health Physics Group*, qualified in radiation protection procedures, shall be on site when fuel is in the reactor;

d. ALL CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

e. A site Fire Brigade of at least 5 members* shall be maintained ~~onsite~~ within the Site Boundary at all times. The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the Unit and any personnel required for other essential functions during a fire emergency; and

---

*May be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

<u>Section 6.2.2.e</u>

<u>Justification</u>

Section 6.2.2.e was revised to read as follows:

> ....shall be maintained within the <u>Site Boundary</u> at all times....

The reasons for the change are as follows:

(1) the term Site Boundary is defined in Section 1.29 (page 1-5) of the Technical Specifications.

(2) Several important structures at Wolf Creek are located outside of the "Protected Area" which is often used by regulatory authorities as the boundary for the term "Onsite." These structures include: Fuel Oil Storage, Circulating Water Screenhouse, Switchyard, and support buildings.

(3) The ESW pumphouse though located in a "Protected Area" is located some distance (1500-1800 feet) from the Power Block.

However, KG&E (Wolf Creek) does not intend to send our Fire Brigade to grass fires or fires at remote "Site Boundary" properties such as the Makeup Water Screenhouse, Makeup Water Discharge Structure, Low Level Outlet Works, and Education Center. A liaison from our brigade may be sent to assist local fire department (working under contract) with such fires, not the entire brigade.

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UNIT STAFF (Continued)

f. Administrative procedures shall be developed and implemented to limit the working hours of Unit Staff who perform safety-related functions; e.g., Senior Operators, Operators, Health Physicists, Auxiliary operators, and key maintenance personnel.

The amount of overtime worked by Unit Staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

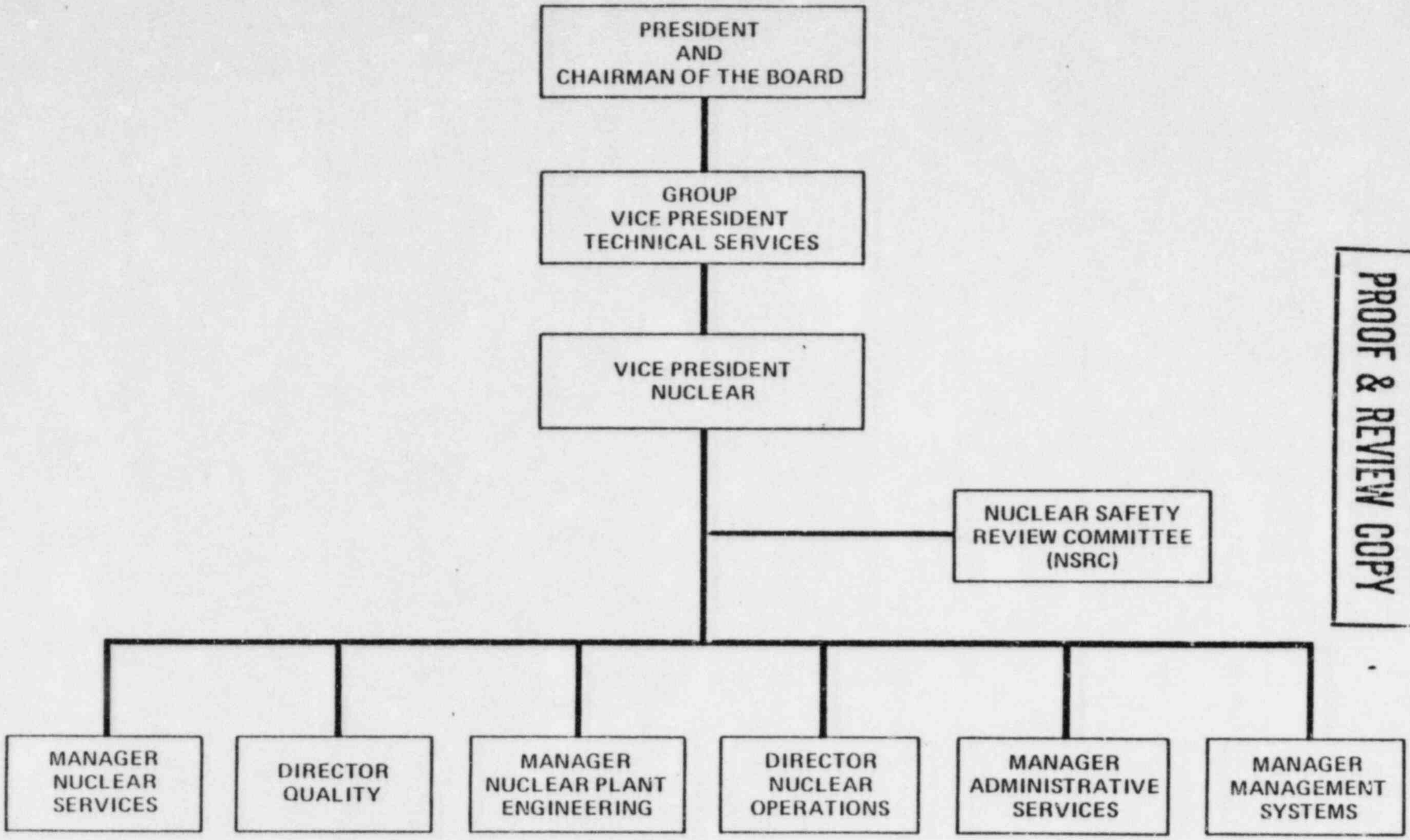
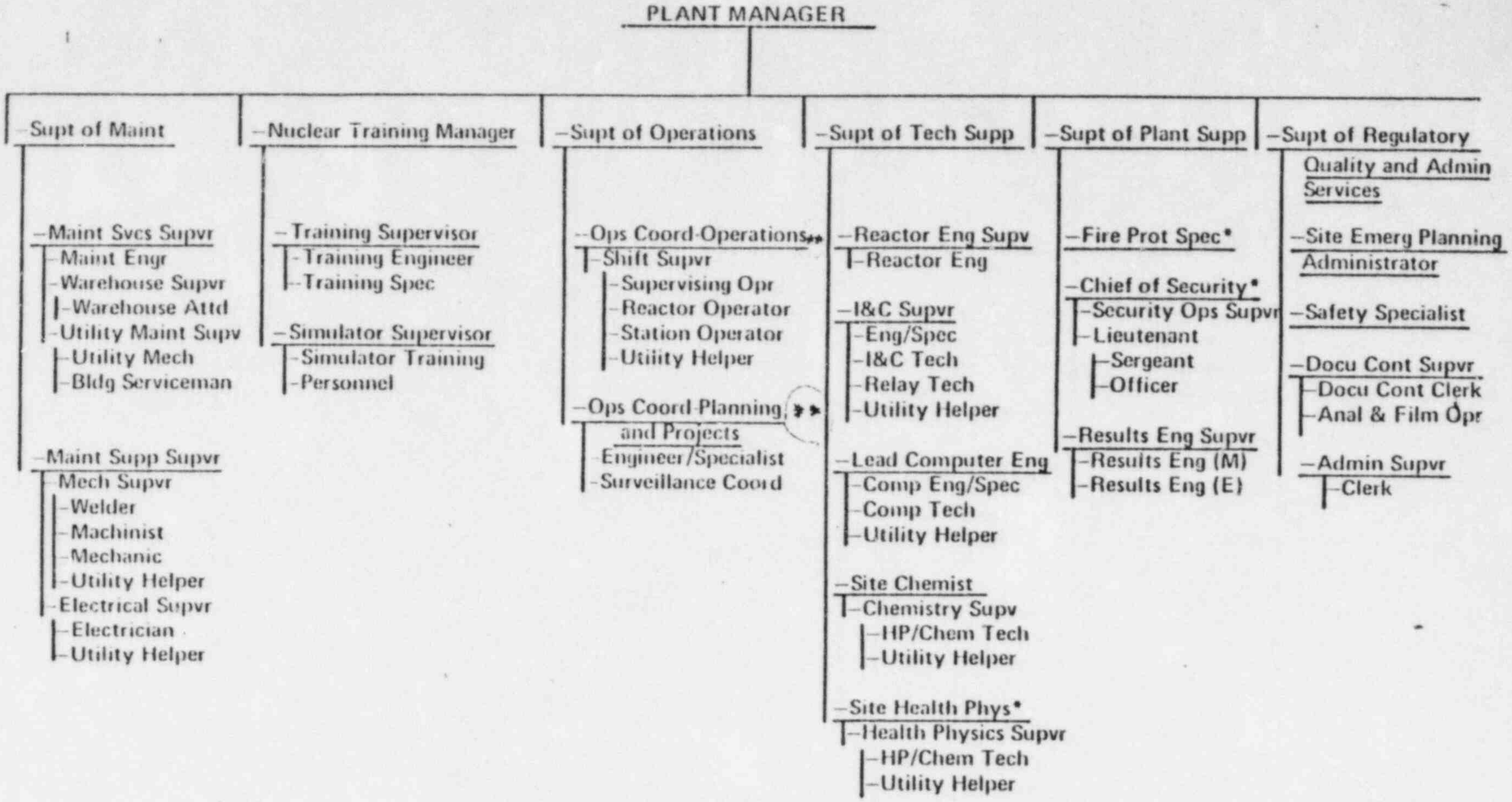


FIGURE 6.2-1

OFFSITE ORGANIZATION



*For technical matters of an immediate nature, the respective individual reports directly to the Plant Manager.

** At least one of the Operations Coordinators will be licensed

FIGURE 6.2-2

UNIT ORGANIZATION

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Figure 6.2-2

Justification -

The "**" note was added to this figure because one of the two Operations Coordinators may not be licensed.

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1	1
SRO	1	None
RO	2	1
SO	4	1
STA	1*	None
CHM	1	None

SS	Shift Supervisor with a Senior Operator license on Unit 1
SRO	Individual with a Senior Operator license on Unit 1
RO	Individual with an Operator license on Unit 1
SO	Station Operator
STA	Shift Technical Advisor
CHM	Chemistry Personnel

The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the Unit is in MODE 5 or 6, an individual with a valid Operator license (other than the Shift Technical Advisor) shall be designated to assume the control room command function.

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*The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, REPORTABLE EVENTS ~~Licensee Event Reports~~ and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the ~~Plant~~ Manager, Nuclear Safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to ~~Plant~~ Manager, Nuclear Safety.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor (STA)** shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the Unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the Unit Staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978, except for the Site Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 for a Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

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*Not responsible for sign-off function.

**The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5 of ANSI/ANS 3.1-1978 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ISEG or another plant group.

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The PSRC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PSRC shall be composed of the:

Chairman:	Plant Manager
Member:	Superintendent of Operations
Member:	Superintendent of Technical Support
Member:	Superintendent of Maintenance
Member:	Instrument and Control Supervisor
Member:	Reactor Engineering Supervisor
Member:	Health Physicist
Member:	Chemist
Member:	Results Engineering Supervisor
Member:	Superintendent of Plant Support
Member:	Superintendent of Regulatory, Quality and Administrative Services

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however no more than two alternates shall participate as voting members in PSRC activities at any one time.

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MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PSRC shall be responsible for:

a. Review of: (1) all procedures required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety;

b. Review of all proposed changes, tests and experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Review of all proposed changes to Technical Specifications or the Operating License;

d. Review of all safety evaluations performed under the provision of Section 50.59(a)(1), 10 CFR, for changes, tests and experiments;

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Director, Nuclear Operations, and to the Nuclear Safety Review Committee (NSRC);

f. Review of all REPORTABLE EVENTS;

g. Review of reports of operating abnormalities, deviations from expected performance of plant equipment and of unanticipated deficiencies in the design or operation of structures, systems or components that affect nuclear safety;

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman, NSRC;

i. Review of the plant Security Plan and implementing procedures and shall submit recommended changes to the NSRC;

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the NSRC;

RESPONSIBILITIES (Continued)

k. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL and the Radwaste Treatment Systems, and

l. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Plant Manager and to the Nuclear Safety Review Committee.

6.5.1.7 The PSRC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a. through d. above,

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through e. above constitutes an unreviewed safety question, and

c. Provide written notification within 24 hours to the Director Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The PSRC shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility ~~and authority~~ provisions of these Technical Specifications. Copies shall be provided to the Director~~,~~ Nuclear Operations, and the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The NSRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

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FUNCTION (Continued)

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NSRC shall report to and advise the Vice President-Nuclear on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The NSRC shall be composed of the:

Chairman:	Manager Nuclear Services
Member:	Manager Nuclear Plant Engineering
Member:	Manager Quality Assurance (Home Office)
Member	Director Nuclear Operations
Member:	Manager Licensing
Member:	Vice President-Engineering
Member:	Manager Nuclear Safety

ALTERNATES

6.5.2.3. All alternate members shall be appointed in writing by the NSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the Unit.

REVIEW

6.5.2.7 The NSRC shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, systems or facilities, and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, systems, or facilities which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design of operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PSRC.

AUDITS

6.5.2.8 Audits of Unit activities shall be performed under the cognizance of the NSRC. These audits shall encompass:

a. The conformance of Unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire Unit Staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months;

AUDITS (Continued)

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA or ISEG personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The ODCM and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months;

k. The Emergency Plan and implementing procedures at least once per 12 months;

l. The Security Plan and implementing procedures at least once per 12 months; and

m. Any other area of Unit operation considered appropriate by the NSRC or the Vice President-Nuclear.

RECORDS

6.5.2.9 Records of NSRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, reviewed by participating members and forwarded to the Vice President-Nuclear within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.7 above, shall be prepared, reviewed by participating members and forwarded to the Vice President-Nuclear within 14 days following completion of the review; and

RECORDS (Continued)

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the Vice President-Nuclear and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 of 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PSRC and submitted to the NSRC and the Vice President-Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear and the NSRC shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective ACTION taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC and the Vice President-Nuclear within 14 days of the violation; and

d. Critical operation of the Unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978,

b. The ~~applicable~~ emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter 82-33,

c. Security Plan implementation,

d. Emergency Plan implementation,

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PROCEDURES AND PROGRAMS (Continued)

e. Process Control Program implementation,

f. ODCM implementation, and

g. Quality Assurance Program implementation for effluent and environmental monitoring.

Major Procedures, supported by appropriate Minor Procedures (such as checkoff lists, operating instructions, data sheets, alarm responses, etc.), shall be provided for the above activities.* A Major Procedure is a procedure which controls safety-related activities, and establishes one or more basic controls, overall responsibilities, authority assignments or administrative and operational ground rules at the Wolf Creek plant. Major Procedures are written to meet the requirements of ANSI N18.7-1976/ANS 3.2 and generally are supported by Minor Procedures which provide delineation of details such as for valve lineups, calibration procedures, operating instructions, data sheets, alarm responses, and other procedures identified as "supporting." Major Procedures require signature approval in all cases by the Plant Manager or a Call Superintendent in his absence. A Minor Procedure is a procedure which controls safety-related activities in support of a Major Procedure. It addresses a specific topic or sub-topic established by its 'parent' Major Procedure, expanding on it by providing working level instructions. Minor Procedures are not permitted to contradict requirements contained in their governing Major Procedure. Minor Procedures require signature approval by the Plant Manager, or a Call Superintendent in his absence, only at Revision '0.'

6.8.2 Approval of Procedures

a. All Major Procedures of the categories listed in Specification 6.8.1 and modifications to the intent thereof shall be reviewed by the PSRC and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in Administrative Procedures.

b. Minor Procedures (checkoff lists, operating instructions, data sheets, alarm responses, chemistry and analytical procedures, technical instructions, special and routine maintenance procedures, laboratory manuals, etc.) shall, prior to initial use, be approved by the PSRC or a Subcommittee thereof.

See insert >

6.8.3 Changes to Procedures

a. Temporary changes to Major Procedures, of the categories listed in Specification 6.8.1 which do not change the intent or generate an unreviewed safety question of the original or subsequent approved procedure, may be made provided such changes to operating procedures are approved by the Shift Supervisor (SRO licensed) and one of the Call Superintendents. For temporary changes to Major Procedures under the jurisdiction of Maintenance, Instrumentation and Control, Reactor Engineering, Chemistry, or Health Physics which do not change the intent or generate an unreviewed safety question, changes may be made upon approval of the Cognizant Group Leader and a Call Superintendent.

Insert

6.8.1

* With the exception of Corporate Emergency Plan implementing procedures. Corporate Emergency Plan implementing procedures shall be provided but shall not be designated as major or minor procedures.

Insert

6.8.2

c. Corporate Emergency Plan implementing procedures shall be reviewed by appropriate Corporate and plant personnel and approved by the Vice-President-Nuclear as set forth in General Procedures.

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PROCEDURES AND PROGRAMS (Continued)

All temporary changes to Major Procedures (made by a Call Superintendent and either a Cognizant Group Leader or the Shift Supervisor) shall subsequently be reviewed by the PSRC and approved by the Plant Manager within 14 days, except that temporary changes to Major Procedures made during a refueling outage may be reviewed and approved at any time prior to initial criticality of the reload core. All permanent changes to Major Procedures shall be made in accordance with Specification 6.8.2.a.

b. All temporary or permanent changes to Minor Procedures (checkoff lists, alarm responses, data sheets, operating instructions, etc.) shall be approved by the Shift Supervisor, and shall be subsequently reviewed and approved by the Operations PSRC Subcommittee. All temporary or permanent changes to other Minor Procedures under the jurisdiction of Maintenance, Instrumentation and Control, Reactor Engineering, Chemistry, or Health Physics, shall be approved by a Cognizant Group Leader and shall be subsequently reviewed and approved by the appropriate PSRC Subcommittee.

See insert

6.8.4 The following programs shall be established, implemented, and maintained:

a. ~~Primary~~ Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the appropriate portions of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, RHR System, and the Nuclear Sampling System (PASS only). The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

Insert

6.8.3

c. Temporary changes to Corporate Emergency Plan implementing procedures may be made provided that 1) the intent of the original procedure is not altered, 2) the change is approved by the Emergency Planning Coordinator, and 3) the change is documented, reviewed by appropriate Corporate and plant personnel and approved by the Vice-President-Nuclear within 14 days of the implementation.

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PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective ACTION for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective ACTION.

d. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

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STARTUP REPORT (Continued)

6.9.1.2 The Startup Report shall address each of the tests identified in the Final Safety Analysis Report FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective ACTIONS that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption of commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS

6.9.1.4 Annual Reports covering the activities of the Unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. Tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,* e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions; and

b. Documentation of all challenges to the PORVs or safety valves.

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*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 Routine Annual Radiological Environmental Operating Reports covering the operation of the Unit during the previous calendar year shall be submitted by May 1 of each year. The initial report shall be submitted by May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activites for the report period, including a comparison with preoperational studies, with operational controls and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective actions being taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by Specification 3.12.1 and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

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*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT* 6

6.9.1.7 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the Unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the Unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid waste (as defined by 10 CFR Part 60), type of container (e.g., LSA, Type A, Type B, Large Quantity), and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the Unit or Station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figures 5.1-3 and 5.1-4) during the report period using historical average atmospheric conditions. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the ODCM.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance

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~~*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate Radwaste Systems, the submittal shall specify the releases of radioactive material from each unit.~~

**In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT REPORT (Continued)

with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and the ODCM, pursuant to Specifications 6.13 and 6.14, respectively, as well as any major changes to Liquid, Gaseous, or Solid Radwaste Treatment Systems, pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations identified by the Land Use Census, pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following information: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORT

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the NRC Regional Office, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.9 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator with a copy to Director of Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T . P_{Rel}$) vs Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, it shall be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

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6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those Unit components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

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RECORD RETENTION (Continued)

g. Records of training and qualification for current members of the Unit Staff;

h. Records of in-service inspections performed pursuant to these Technical Specifications;

i. Records of Quality Assurance activities required by the QA Manual;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the PSRC and the NSRC;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.8 including the date at which the service life commences and associated installation and maintenance records;

m. Records of secondary water sampling and water quality; and

n. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device: or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

HIGH RADIATION AREA (Continued)

otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by Health Physics management personnel in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor/Operating Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed-circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

PROCESS CONTROL PROGRAM (PCP) (Continued)

3) Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

b. Shall become effective upon review and acceptance by the PSRC.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PSRC.

b.. Shall become effective upon review and acceptance by the PSRC.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the ~~ORC~~ PSRC. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional and supplemental information;

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*Licensees may choose to submit the information called for in this specification as part of the annual FSAR update.

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS (Continued)

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems.

4) An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PSRC.

2. Shall become effective upon review and acceptance by the PSRC.