UNION ELECTRIC COMPANY
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DONALD F. SCHNELL
VICE PRESIDENT

April 9, 1984

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

Dear Mr. Denton:

ULNRC-792

DOCKET NUMBER 50-483
CALLAWAY PLANT, UNIT 1
CALLAWAY TECHNICAL SPECIFICATIONS

References: 1) D. G. Eisenhut letter to D. F. Schnell dated March 8, 1984
2) ULNRC-787 dated April 5, 1984

Attachments: 1) Summary Listing of Attached Specifications
2) Specifications Resolved Since Final Draft
3) Specifications Being Appealed
4) Specifications Which Have Not Been Resolved

Reference 1 transmitted the Callaway Technical Specifications in final draft form. Union Electric has reviewed this final draft and has included as attachments to this letter those changes which are necessary so that the Callaway Technical Specifications accurately reflect the plant design and operating program.

Attachment 1 is a summary listing of the specifications contained in Attachments 2, 3 and 4. Attachment 2 contains those specifications which have been revised since the final draft and for which UE and NRC have reached mutual agreement on the wording and substance. Attachment 3 contains those specifications which are being appealed as noted in Reference 2.

Attachment 4 contains those specifications which have not been resolved. Tables 3.3-10 and 3.3-11 include, as item 19, Reactor Coolant Radiation Level Instrumentation. These are footnoted as not needed until startup following the first refueling. These instruments are not in the Callaway design. The SNUPPS submittal on Regulatory Guide 1.97, Revision 2 was made by SLNRC 82-031 dated July 6, 1982 and subsequently incorporated in SNUPPS FSAR Revision 10 in September, 1982. The review of this information is still in progress and a license condition will assure proper resolution.

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The second unresolved specification (3/4 5.5) concerns the boron concentration in the Boron Injection Tank (BIT). A proposed specification change with justification was submitted by SLNRC 84-0046 dated March 20, 1984. This is under review by NRC.

The Callaway Technical Specifications reflect an acceptance criteria of 12 seconds for the diesel generator start time from ambient conditions. FSAR text changes which support this will be provided by letter prior to fuel load and will be incorporated in the next SNUPPS FSAR revision.

Except as noted herein, in my judgement, the Callaway Technical Specifications accurately reflect the plant design and operating program as described in the FSAR and other information on our docket.

Very truly yours,

Donald F. Schnell

Donald F. Schnell

DFS/cfs

Attachments

cc: J. J. Holonich

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

Donald F. Schnell, of lawful age, being first duly sworn upon oath says that he is Vice President-Nuclear and an officer of Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Donald F. Schnell
Donald F. Schnell
Vice President
Nuclear

SUBSCRIBED and sworn to before me this 9th day of April, 1984.

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ATTACHMENT 1

Summary Listing of Attached Specifications

Item	Page	Agree	Open	Issue
1	III	x		Editorial
2	IV		x	Editorial
3	VII	x		Editorial
4	IX		x	Awaiting NRC Review Deletion of BIT
5	X	x		Editorial
6	XI	x		Editorial
7	XV		x	Awaiting NRC Review Deletion of BIT
8	2-2	x		Editorial
9	2-5 thru 2-11	x	x(p2-7)	Incorporation of Callaway Setpoints
10	B2-5	x		Editorial
11	3/4 1-1	x		Editorial
12	3/4 1-4	x		Editorial
13	3/4 1-6	x		Editorial
14	3/4 1-7		x	Awaiting NRC Review Boration Systems
15	3/4 1-10	x		Editorial
16	3/4 1-11		x	Awaiting NRC Review Boration Systems
17	3/4 1-12		x	Awaiting NRC Review Boration Systems
18	3/4 1-13		x	Awaiting NRC Review Boration Systems
19	3/4 1-14		x	Editorial Boration Systems
20	3/4 1-15	x		Editorial
21	3/4 1-13 thru 3/4 1-24		x	Numbering changes due to boration systems
22	3/4 1-15	x		Editorial
23	3/4 1-21	x		Editorial
24	3/4 1-22	x		Editorial
25	3/4 2-1	x		Editorial
26	3/4 2-8	x		Flow Uncertainties
27	3/4 2-9	x		Flow Uncertainties
28	3/4 2-10	?		Flow Uncertainties
29	3/4 2-11	x		Editorial
30	3/4 3-5	x		Editorial
31	3/4 3-10	x		Editorial
32	3/4 3-13	x		Inclusion of new action statement
33	3/4 3-14	x		Editorial
34	3/4 3-17	x		Editorial
35	3/4 3-19	x		Inclusion of solid state load sequencer
36	3/4 3-21	x		Inclusion of action statement for SSLS
36	3/4 3-22 thru 3-27	x		Inclusion of Callaway Setpoints
37	3/4 3-27	x		Inclusion of solid state load sequencer
38	3/4 3-30	x		Editorial
39	3/4 3-31	x		Editorial
40	3/4 3-32	x		Editorial
41	3/4 3-33 thru 3/4 3-37	x		Editorial
42	3/4 3-37	x		Inclusion of solid state load sequencer
43	3/4 3-29	x		Radiation Monitoring Instrumentation setpoints

Item	Page	Agree	Open	Issue
44	3/4 3-40	x		Radiation monitoring instrumentation setpoints
45	3/4 3-44		x	Seismic switch trip setpoints
46	3/4 3-45	x		Exception to analog channel operational test
47	3/4 3-52	x		Editorial
48	3/4 3-53 thru 3/4 3-56		x	Reg Guide 1.97 instrumentation
49	3/4 3-61	x		Inclusion of ESF transformers/Editorial
50	3/4 3-64	x		Editorial
51	3/4 3-66	x		Editorial
52	3/4 3-67	x		Changes to clarify note 2
53	3/4 3-69	x		Inclusion of instrumentation 3.b thru 3.e
54	3/4 3-70		x	Editorial
55	3/4 3-71	x		Editorial
56	3/4 3-72	x		Inclusion of instrumentation 3.b thru 3.e
57	3/4 3-73		x	Editorial
58	3/4 3-74	x		Changes to clarify note 3
59	3/4 3-76	x		Inclusion of Turbine overspeed protection program
60	3/4 4-1	x		Editorial
61	3/4 4-3	x		Editorial
62	3/4 4-13	x		Deletion of "Condition IV" from C.3 and C.4
63	3/4 4-14	x		Editorial
64	3/4 4-18	x		Editorial
65	3/4 4-26	x		Editorial
66	3/4 4-30	x		Editorial
67	3/4 4-31	x		Editorial
68	3/4 4-34 3/4 4-35		x	RHR suction relief valves for cold over-pressure protection appeal issue
69	3/4 5-3	x		Editorial
70	3/4 5-5	x		Editorial
71	3/4 5-6	x		RHR System flowrate
72	3/4 5-8	x		Editorial
73	3/4 5-9	x		Editorial
74	3/4 5-10		x	Deletion of boron injection tank spec
75	3/4 5-11		x	Editorial based on Item 74
76	3/4 6-2	x		Editorial
77	3/4 6-3		x	Verbal agreement reached new words need review
78	3/4 6-5		x	UE verifying revised numbers
79	3/4 6-6	x		Revised containment internal pressure setpoints
80	3/4 6-8 thru 6-10		x	Containment vessel structural integrity - appeal
81	3/4 6-11	x		Increase cont ventilation limit to 2000 hrs.
82	3/4 6-12	x		Increase limit to .05 La
83	3/4 6-16		x	Containment isolation valves - appeal
84	3/4 6-20	x		Editorial
85	3/4 6-21	x		Editorial
86	3/4 6-22	x		Editorial
87	3/4 6-26	x		Editorial
88	3/4 6-29	x		Inclusion of breathing air supply valves
89	3/4 6-30	x		Revision of hydrogen analyzer surveillance
90	3/4 6-31	x		Deletion of hydrogen purge
91	3/4 7-7	x		Editorial
92	3/4 7-8	x		Editorial

Item	Page	Agree	Open	Issue
93	3/4 7-15	x		Editorial
94	3/4 7-27	x		Increasing volume of fire suppression tanks
95	3/4 7-30	x		Inclusion of ESF transformers in specification
96	3/4 7-33	x		Editorial
97	3/4 7-34	x		Editorial
98	3/4 7-35	x		Editorial
99	3/4 7-36		x	Editorial
100	3/4 7-37	x		Editorial
101	3/4 7-38	x		Editorial
102	3/4 8-2	x		Moved proposed surveillances to other specs
103	3/4 8-3	x		Change voltage valves/include new fuel spec
104	3/4 8-4	x		Inclusion of new fuel oil spec
105	3/4 8-5	x		Change voltage values
106	3/4 8-6	x		Editorial
107	3/4 8-7			Added statement for DG Start counting
108	3/4 8-8		x	RHR suction relief issue - appeal
109	3/4 8-10	x		Editorial
110	3/4 8-12		x	RHR suction relief issue - appeal
111	3/4 8-18 thru 3/4 8-24	x		Editorial and changes to values in Table 3.8-1
112	3/4 9-15	x		Change to incorporate SER requirement
113	3/4 11-4	x		Change to note 6
114	3/4 11-7	x		Added demin vessels to Loo
115	3/4 11-9	x		Editorial
116	3/4 11-11	x		Editorial
117	3/4 11-16	x		Change 31 days to 7 days for surveillance
118	3/4 12-4	x		Editorial
119	3/4 12-8	x		Change to note 7
120	3/4 12-14	x		Editorial
121	B3/4 1-2		x	Boration system under review by NRC
121	B3/4 1-3		x	Boration system under review by NRC
122	B3/4 2-4	x		Rod bow penalty
123	B3/4 2-5		x	RCS Flow Rate - Ventury Fouling
124	B3/4 4-3	x		Clarification on S/G Turg inspection
125	B3/4 4-15		x	RHR suction relief valves - appeal
126	B3/4 5-1	x		Editorial
127	B3/4 5-2		x	Deletion of BIT under review by NRC
128	B3/4 6-2	x		Containment internal pressure
129	B3/4 6-3	x		Containment purge limit increase to 2000 hrs.
130	B3/4 6-4	x		Editorial
131	B3/4 12-2	x		Editorial
132	5-1	x		Editorial
133	6-1	x		Editorial
134	6-3,4	x		Organizational chart changes
135	6-6	x		Editorial
136	6-9	x		Turbine overspeed protection program
137	6-13	x		Temporary change review
138	6-15	x		Turbine overspeed protection program/editorial
139	6-16	x		Turbine overspeed protection program
140	6-19	x		Editorial

ATTACHMENT 2

Specifications Resolved Since Final Draft

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INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

DRAFT

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DRAFT

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

PLANT SYSTEMS (Continued)

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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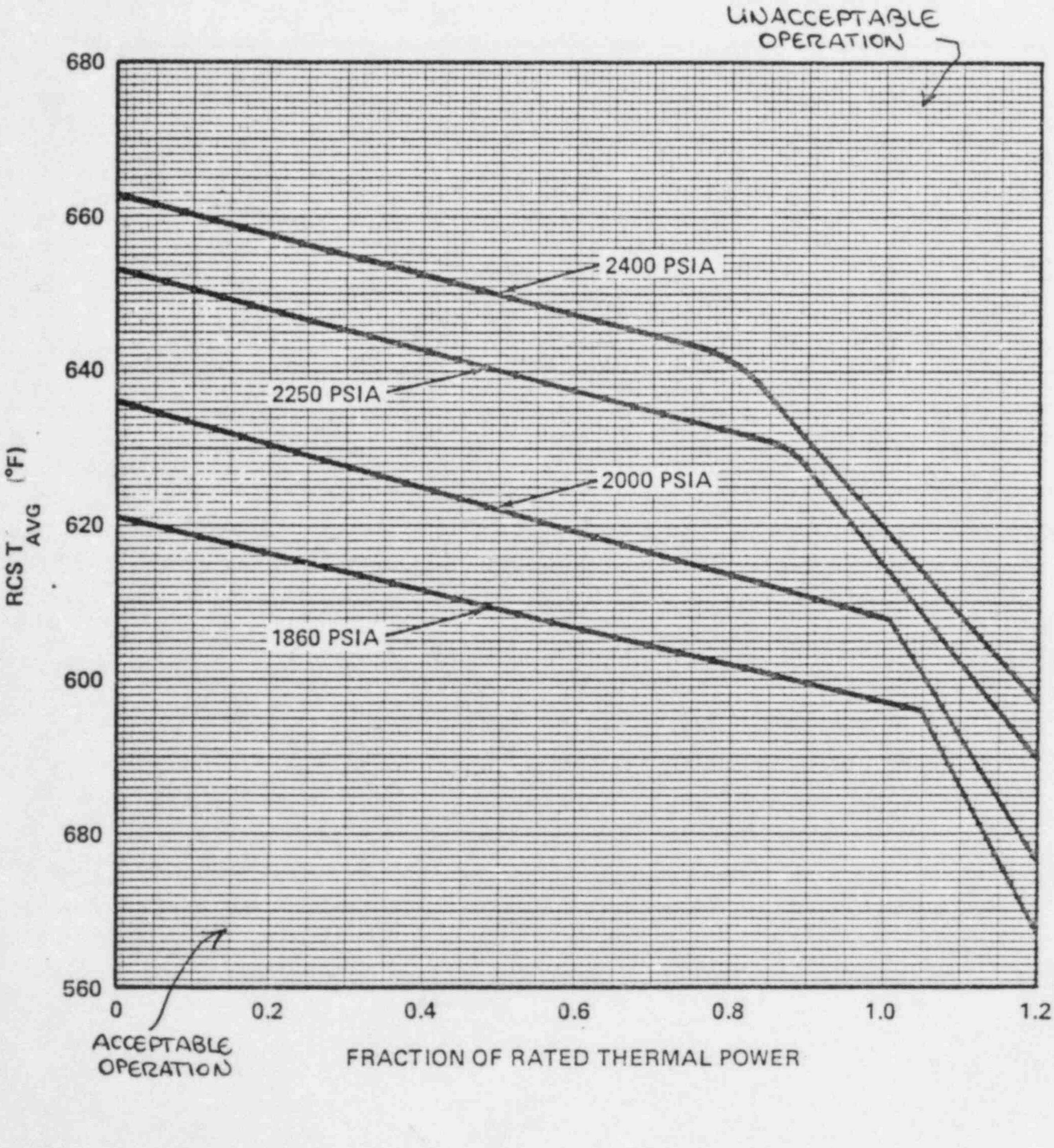

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	SENSOR ERROR 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	$\leq$109% of RTP*	$\leq$112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	$\leq$25% of RTP*	$\leq$28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	$\leq$4% of RTP* with a time constant $\geq$2 seconds	$\leq$6.3% of RTP* with a time constant $\geq$2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	$\leq$4% of RTP* with a time constant $\geq$2 seconds	$\leq$6.3% of RTP* with a time constant $\geq$2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	$\leq$25% of RTP*	$\leq$35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	6.1	2.76	1.8	See Note 1	See Note 2
8. Overpower ΔT	4.6	1.3	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.21	2.0	$\geq$1885 psig	$\geq$1874 psig
10. Pressurizer Pressure-High	7.5	4.96	1.0	$\leq$2385 psig	$\leq$2400 psig
11. Pressurizer Water Level-High	8.0	2.18	2.0	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.0	1.5	$\geq$90% of loop design flow**	$\geq$89.2% of loop design flow**

*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
13. Steam Generator Water Level Low-Low	23.5	21.18	2.0	≥23.5% of narrow range instrument span	≥22.0% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	7.7	1.33	0	≥10584 Volts A.C	≥10356 Volts A.C.
15. Underfrequency - Reactor Coolant Pumps	3.3	0	0	≥57.2 Hz	≥57.1 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥598.94 psig	≥539.42 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.

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.	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ 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 ~~≤~~12.4% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	≤10% of RTP* Turbine Impulse Pressure Equivalent	≤12.4% of RTP* Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤48% of RTP*	≤51.3% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤50% of RTP*	≤53.3% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	~~≥~~10% of RTP*	≥6.7 to 12.4% of RTP*
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	≤10% of RTP* Turbine Impulse Pressure Equivalent	≤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.

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

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_1 = 1.10;

K_2 = 0.0137/°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}, $\tau_4 = 28$ s, $\tau_5 = 4$ 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, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T' $\leq$ 588.5°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.000671;

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 2.8% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

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_4	=	1.085;
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}, τ_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, τ_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 2.8% of ΔT span.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, 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 to 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 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 Releases."

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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

*See Special Test Exception Specification 3.10.1.

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MODERATOR TEMPERATURE COEFFICIENT

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.

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MINIMUM TEMPERATURE FOR CRITICALITY

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.

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CHARGING PUMPS - OPERATING

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 SHUTDOWN within the next 30 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.

*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°F.

3/4.2 POWER DISTRIBUTION LIMITS

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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 cumulated 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 RATE 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.

*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.

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 ~~undetected feedwater venturi fouling of 0.1% and for~~ measurement uncertainties of 1.Ø% 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.

PENALTIES ~~OF 0.1%~~ FOR ~~UNDETECTED FEEDW. TER VENTURI FOULING AND~~ MEASUREMENT UNCERTAINTIES OF 2.0% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F_{\Delta H}^{N}$ ARE INCLUDED IN THIS FIGURE

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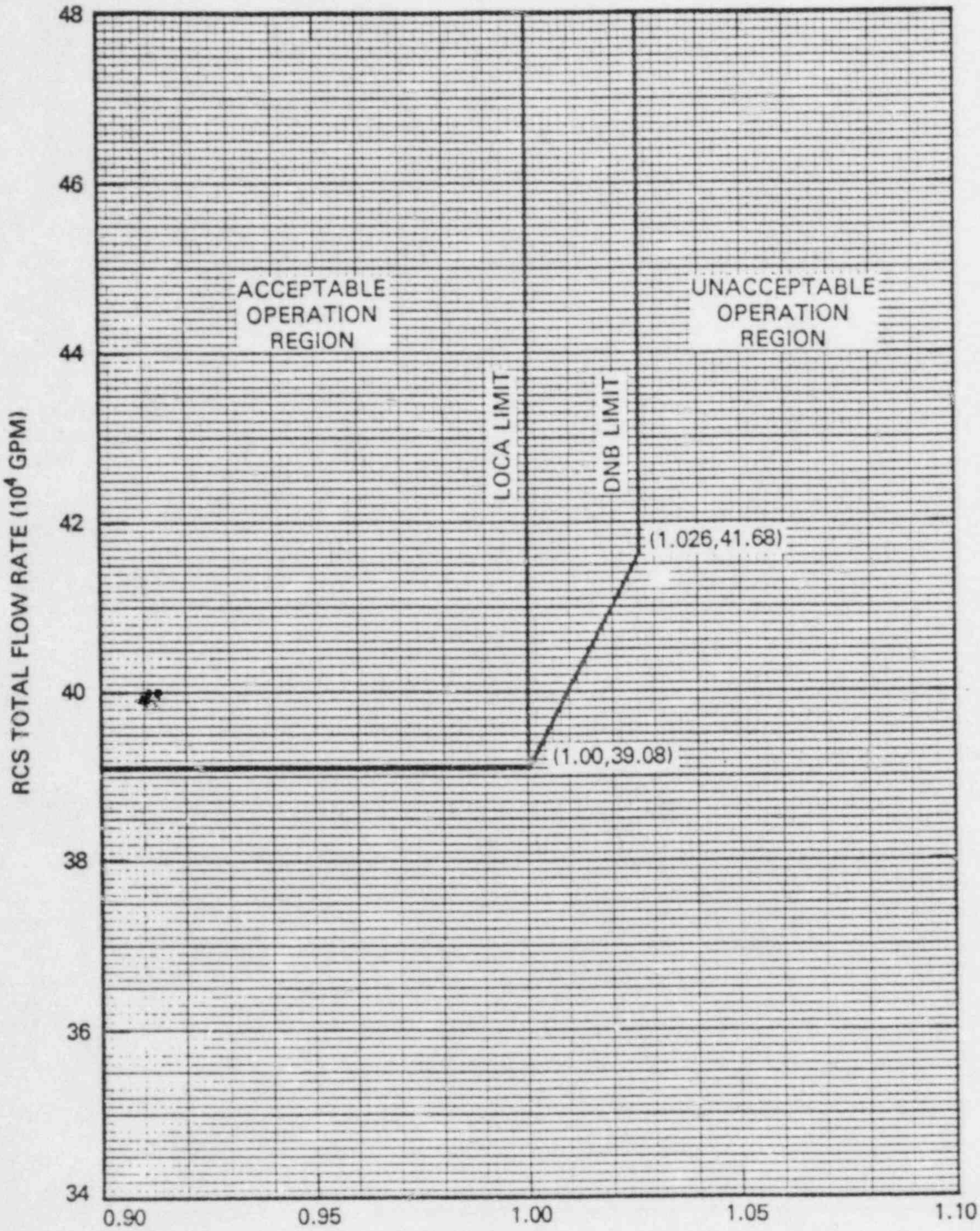

$R = F_{\Delta H}^{N}/1.49\left[1.0 + 0.2(1.0 - P)\right]$

FIGURE 3.2-3

RCS TOTAL FLOW RATE VERSUS R

FOUR LOOPS IN OPERATION

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3/4.2.4 QUADRANT POWER TILT RATIO

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 Specification 3.10.2.

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be in the closed position ~~and the~~ Control Rod Drive System is 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.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be closed and the Con- ~~trol 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 applicable.

(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.

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 (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Component Cooling Water, Turbine Trip, Auxiliary Feedwater-Motor-Driven Pump, Emergency Diesel Generator 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	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Feedwater Isolation & Turbine Trip					
a. Automatic Actuation Logic and Actuation Relay (SSPS)	2	1	2	1, 2	21
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.				
6. Auxiliary Feedwater					
a. Manual Initiation	3(1/pump)	1/pump	1/pump	1, 2, 3	24
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. Steam Generator 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*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
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*
9. Control Room Isolation					
a. Manual Initiation	2	1	2	All	18
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	All	14
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	2	1	2	All	14
d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation initiating functions and requirements.				
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
11. 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)

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 inoperable 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 - 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 - See Insert on following page

Action 25 - With the number of OPERABLE channels one less than the mininum 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.

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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 (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Turbine Trip, Component Cooling Water, Auxiliary Feedwater - Motor-Driven Pump, 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	3.6 ~~2.5~~	0.71	2.0 ~~1.5~~	≤ 3.5 psig	≤ 4.6 ~~4.0~~ psig
d. Pressurizer Pressure - Low	18.6 ~~13.0~~	14.41 ~~10.71~~	2.0 ~~1.5~~	≥ 1849 psig	≥ 1834 ~~1839~~ psig
e. Steam Line Pressure - Low	19.6 ~~14.2~~	14.81 ~~10.71~~	2.0 ~~1.5~~	≥ 615 ~~585~~ psig	≥ 571 ~~564~~ psig*
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.

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 (Continued)					
c. Containment Pressure-High-3	4.3	0.71	2.0	≤ 27.0 psig	≤ 28.3 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	0.71	2.0	≤ 27.0 psig	≤ 28.3 psig
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.

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)					
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
4) 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	0.71	2.0	$\leq$ 17.0 psig	$\leq$ 18.3 psig
d. Steam Line Pressure-Low	19.6	14.81	2.0	$\geq$ 615 psig	$\geq$ 571 psig*
e. Steam Line Pressure Negative Rate - High	3.0	0.5	0	$\leq$ -100 psi/s	$\leq$ -124 psi/s**
5. Feedwater Isolation & Turbine Trip					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A	N.A.	N.A.	N.A.

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
5. Feedwater Isolation (Continued)					
b. Steam Generator Water Level-High-High	5	2.18	2.0	$\leq$ 78% of narrow range instrument span	$\leq$ 79.8% of narrow range instrument span
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 Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level-Low-Low					
1) Start Motor-Driven Pumps	23.5	21.18	2.0	$\geq$ 23.5% of narrow range instrument span	$\geq$ 22.0% of narrow range instrument span
2) Start Turbine-Driven Pump	23.5	21.18	2.0	$\geq$ 23.5% of narrow range instrument span	$\geq$ 22.0% of narrow range instrument span

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)					
e. Safety Injection-Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
f. Loss-of-Offsite Power-Start Turbine-Driven Pump	N.A.	N.A.	N.A.	N.A.	N.A.
g. Trip of All Main Feedwater Pumps-Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
h. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to ESW)	N.A.	N.A.	N.A.	$\geq$23.4	$\geq$21.72
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	1.21	2.0	$\geq$36%	$\geq$35.2%
	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
8. Loss of Power					
a. 4 kV Undervoltage -Loss of Voltage	N.A.	N.A.	N.A.	83V (120V Bus) w/1s delay	83+0,-8.3V (120V Bus) w/1+0.2,-0.5s delay

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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
8. Loss of Power (Continued)					
b. 4 kV Undervoltage -Grid Degraded Voltage	N.A.	N.A.	N.A.	104.5V (120V Bus) w/119s delay	104.5+2.6,-0V (120V Bus) w/119 ± 11.6s delay
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. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1970 psig	≤ 1981 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
11. Solid State Load Sequencer	N.A.	N.A.	N.A.	N.A	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)	$\leq 29^{(1)}/12^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Start Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 24^{(3)}/12^{(4)}$
1) Reactor Trip (from SI)	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Start Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
b. Steam Line Isolation	≤ 7

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	$\leq 32^{(1)}/20^{(2)}$
b. Phase "B" Isolation	$\leq$ 31.5
6. Containment Pressure--High-2	
Steam Line Isolation	$\leq$ 7
7. Steam Line Pressure-Negative Rate-High	
Steam Line Isolation	N.A.
8. Steam Generator Water Level--High-High	
a. Feedwater Isolation	$\leq$ 7
b. Turbine Trip	$\leq$ 2.5
9. Steam Generator Water Level - Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
10. Loss-of-Offsite Power	
Start Turbine-Driven Auxiliary Feedwater Pump	N.A.
11. Trip of All Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	N.A.

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	≤ 14
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	≤ 144
15. Phase "A" Isolation	
a. Control Room Isolation	N.A.
b. Containment Purge Isolation	$\leq 2^{(5)}$

TABLE NOTATIONS

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

(2) Diesel generator starting delay <u>not</u> included. Offsite power available.

(3) Diesel generator starting and sequence loading delay included. RHR pumps <u>not</u> included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps <u>not</u> included.

(5) Does not include valve closure time.

(6) 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 SURVEILLANC IS REQUIRED
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Turbine Trip, Component Cooling Water, Auxiliary Feedwater-Motor-Driven Pump, Emergency Diesel Generator Operation, 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

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
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(2)(1)	N.A.	N.A.	1, 2, 3, 4
4) 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	R	M	N.A.	N.A.	N.A.	N.A.	3, 4
5. Feedwater Isolation & Turbine Trip								
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

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)								
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(2)(1)	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Loss-of-Offsite Power	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2
h. 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	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
Coincident With 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

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
8. Loss of Power (Continued)								
b. 4 kV Undervoltage-Grid Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
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
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(2)(1)	N.A.	N.A.	All
d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							
10. 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
11. Solid State Load Sequencer	N.A.	N.A.	N.A.	N.A.	M(1)(2)	N.A.	N.A.	1,2,3,4

TABLE NOTATIONS

(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.

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	###	26
b. Gaseous Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
c. 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-Gaseous Radioactivity-High (GG-RE-27 & 28)	1	2	**	##	30
b. Criticality-High Radiation Level (SD-RE-37 & 38)	1	2	*	$\leq$ 15 mR/h	28
3. Control Room					
Air Intake-Gaseous Radioactivity-High (GK-RE-04 & 05)	1	2	All	#	27

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

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

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

#Trip Setpoint concentration value ($\mu Ci/cm^3$) is to be established such that the actual submersion dose rate would not exceed 2 mR/h in the control room.

##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 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 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 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 requirement for Specification 3.4.6.1.

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 4.3-4

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 Air Filters	M	R	SA**
f. Free Field	M	R	SA**
3. Triaxial Response-Spectrum Recorder (Passive)			
Ctmt. Base Slab	N.A.	R	N.A.*
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 Bi-stable Trip Setpoint need not be determined during the performance of an ANALOG CHANNEL OPERATIONAL TEST.

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, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident 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 or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels for the reactor coolant radiation level monitor or the unit vent - high range noble gas monitor less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

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.

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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)
6202-Elec. Equipment Rm.	601			3/0
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	002			3/0
-South ESW Pumphouse	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. uc

(2) Line-type heat detector.

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-RE-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. Combined Cooling Tower Blowdown Line and Bypass Flow	1	34

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	P	R(2)	Q(1)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Discharge Line	D(3)	N.A.	R	N.A.
b. Steam Generator Blowdown Discharge Line	D(3)	N.A.	R	N.A.
c. Secondary Liquid Waste System Discharge Line	D(3)	N.A.	R	N.A.
d. Combined Cooling Tower Blowdown Line and Bypass Flow	D(3)	N.A.	R	N.A.

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur as appropriate if 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. Instrument controls not set in operate mode (alarm only).

(2) 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, measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, NBS traceable standard (gas, liquid, or solid) may be used; or a gas, liquid, or solid source that has been calibrated by relating it to equipment that was previously (within 30 days) calibrated by the same geometry and type of source standard traceable to NBS.

(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.

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 Monitors	1/recombiner	**	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 Sampler	1	*	43
c. Particulate Sampler	1	*	43
d. Flow Rate Monitor	1	*	39
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)	1	*	41
b. Iodine Sampler	1	*	43
c. Particulate Sampler	1	*	43
d. Flow Rate ~~Monitor~~	~~1~~ N.A.	*	~~39~~ 45
e. Sampler Flow Rate Monitor	1	*	39

TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLD UP 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 provided the flow rate is estimated based on fan status and operating curves or actual measurements 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 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 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.

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 sampling 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 requirement, suspend oxygen supply to the recombiner.

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

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(6)	M	**
2. Unit Vent System					
a. Noble Gas Activity Monitor Providing Alarm (GT-RE-21)	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R(7)	Q	*
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)	D	P	R(3)	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate 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	*

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 as appropriate if 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. Instrument 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, measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, NBS traceable standard (gas, liquid, or solid) may be used; or a gas, liquid, or solid source that has been calibrated by relating it to equipment that was previously (within 30 days) by the same geometry and type of source traceable to NBS.

(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.

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 maintained, calibrated, tested, and inspected in accordance with the Callaway Plant's Turbine Overspeed Protection Reliability Program. Adherence to this program shall demonstrate OPERABILITY of this system. The program and any revisions should be reviewed and approved in accordance with Specification 6.5.1.6~~0~~.o Revisions shall be made in accordance with the provisions of 10 CFR 50.59.

*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.

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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 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 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 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 ~~unless~~ the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS (Continued)

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 loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

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 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

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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.

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 $100/\bar{E}$ 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. 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:

a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

b. Results of: (1) the last isotopic analysis for radioiodine performed prior to exceeding the limit, (2) analysis while limit was exceeded, and (3) one analysis after the radioiodine was reduced to less than the limit, including for each isotopic analysis, the date and time of sampling and the radioiodine concentration;

c. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded;

d. History of degassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and

e. The time duration when the specific activity of the reactor coolant exceeded 1 microCurie per gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

MATERIAL PROPERTY BASIS

COPPER CONTENT : ~~CONSERVATIVELY~~ ASSUMED TO BE 0.10 WT%

RT_{NDT} INITIAL :50°F

RT_{NDT} AFTER 16 EFPY :1/4T, 110°F
3/4T, 87°F

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CURVE APPLICABLE FOR HEATUP RATES UP TO 60°F/HR AND 100°F/HR FOR THE SERVICE PERIOD UP TO 7 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

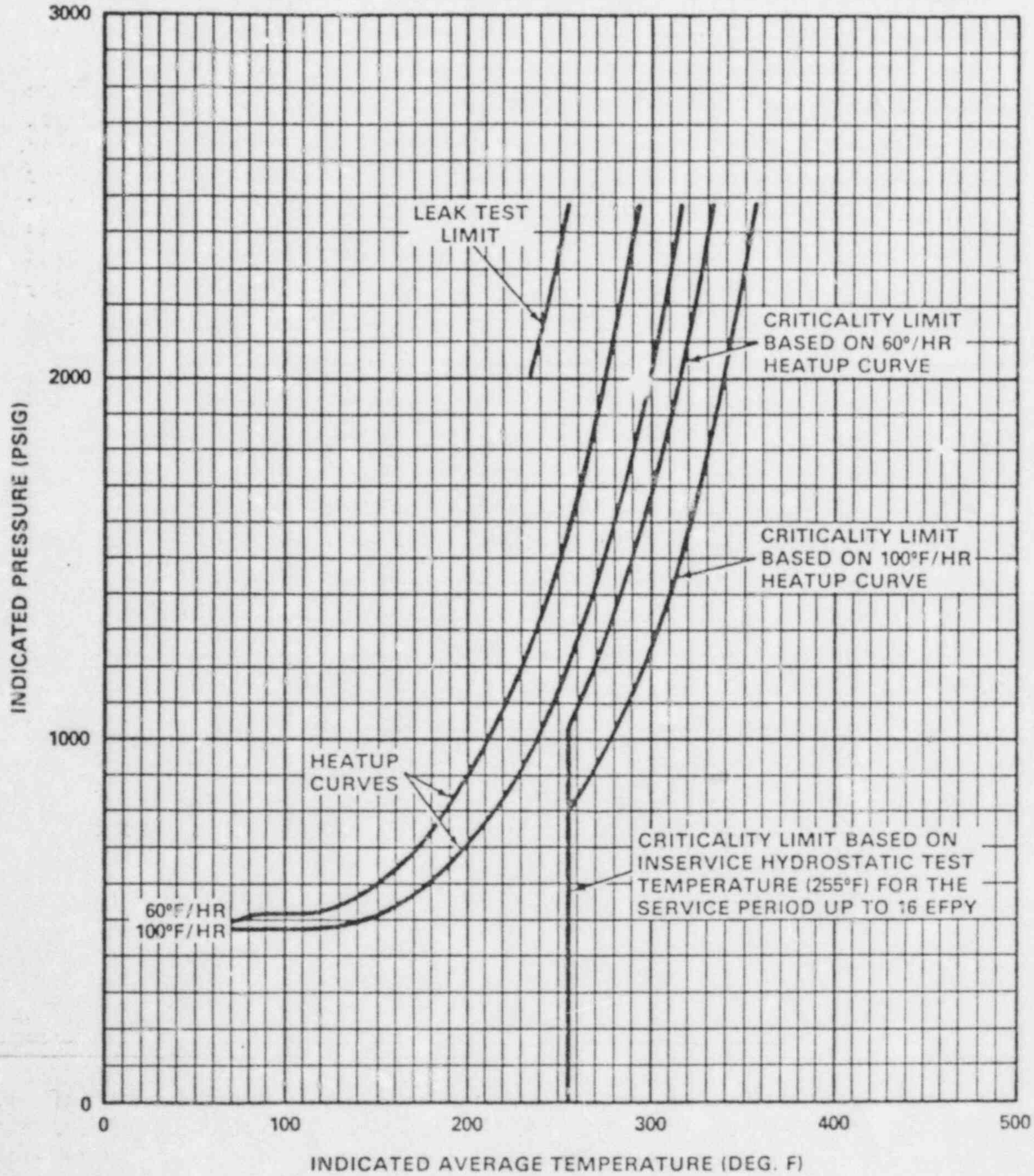

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS APPLICABLE UP TO 7 EFPY

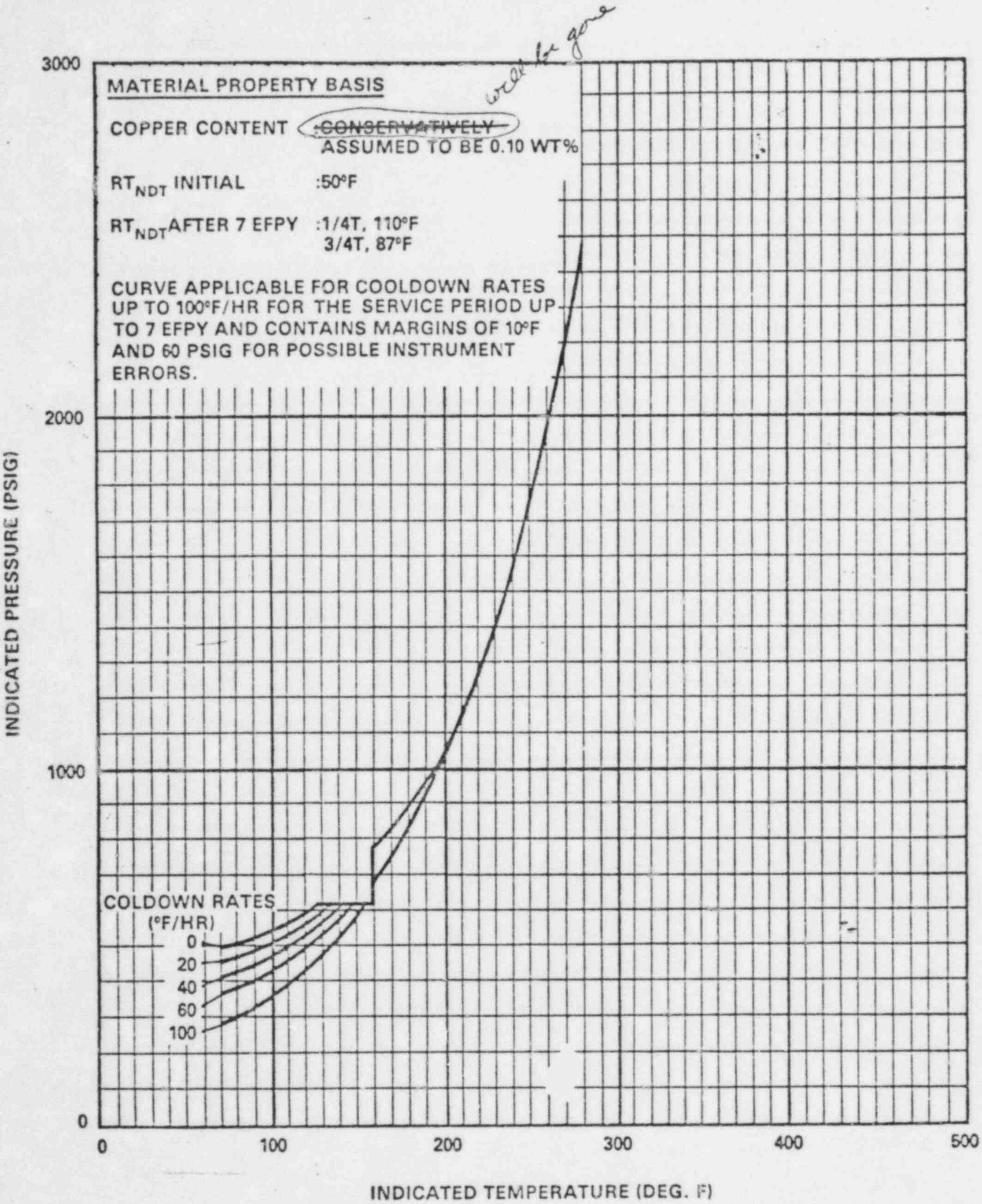

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS APPLICABLE UP TO 7 EFPY

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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.

SPACE

*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.

SURVEILLANCE REQUIREMENTS (Continued)

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 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 $\geq$ 2400 psid,

2) Safety Injection pump $\geq$ 1445 psid, and

3) RHR pump $\geq$ 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
[illegible]/108	EMV092	

SURVEILLANCE REQUIREMENTS (Continued)

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.

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.

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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 within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F 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.

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.

~~per spec 4.0.5~~

*An inoperable pump may be energized for testing 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.

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 or equal to 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;

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INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between +X and -X 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.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 36-inch containment shutdown purge supply and exhaust isolation valve shall be closed and blank flanged, and

b. The 18-inch containment mini-purge supply and exhaust isolation valve(s) may be open for up to 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 blank flanged, close and/or 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 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 4.6.1.7.2 and/or 4.6.1.7.4, 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.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment shutdown purge supply and exhaust isolation valve(s)* shall be verified blank flanged and closed at least once per 31 days.

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.

4.6.1.7.3 The cumulative time that all 18-inch containment mini-purge supply and exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.

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 0.05 L_a when pressurized to P_a.

*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.

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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-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 Supply 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 CTMT Iso	C	5
P-69	SJ HV-13	PZR Vapor Sample Outer ~~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.

*May be opened on an intermittent basis under administrative 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)
2. Phase "A" Isolation (passive)* - (Continued)				
P-16	EN HV-01	CTMT Recirc Sump to CTMT Spray Pump A Iso	A	N.A.
P-13	EN HV-07	CTMT Recirc Sump to CTMT Spray Pump B Iso	A	N.A.
P-45	EP HV-~~880~~ 8880	CTMT Nitrogen Supply Iso Valve	C	N.A.
P-65	GS HV-20	Hydrogen Purge Inner CTMT Iso	C	N.A.
P-65	GS HV-21	Hydrogen Purge Outer CTMT Iso	C	N.A.
P-67	KC HV-253	Fire Protection System Hdr Outer CTMT Iso	C	N.A.
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

*May be opened on an intermittent basis under administrative 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 Sucion	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 Disch arge 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.
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-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-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.
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.

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:

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.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment hydrogen analyzer shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 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.

CONTAINMENT SYSTEMS

HYDROGEN CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.2 A Hydrogen Control ~~System shall~~ be OPERABLE with two independent Hydrogen Recombiner Systems. deleted H_2 purge

APPLICABILITY: MODES 1 and 2

ACTION:

With one of the two independent Hydrogen Recombiner Systems inoperable, restore the inoperable Hydrogen Recombiner System to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 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.

H_2 purge

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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~~0~~ 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.

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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.

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 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, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks;

   3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge at less than or equal to a pressurization flow of 400 cfm relative to adjacent areas during system operation; and

   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.

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 1500 gpm, with their discharge aligned to the fire suppression header;

b. Two separate water supply tanks, each with a minimum level of 31 feet (260,000 gallons); and

c. An OPERABLE flow path capable of taking suction from both fire water storage tanks 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 of the two required pumps and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. 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 the water level in each fire water storage tank exceeds 31 feet (260,000 gallons),

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,

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PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

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	North Electric Cable Chase
Auxiliary	1988/2000	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	XNB01 *
ESF Transformer	Grade	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:

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;

*Areas contain redundant systems or components which could be damaged.

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PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.4 The fire hose stations given in Table 3.7-~~0~~4 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 given in Table 3.7-~~0~~4 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-~~0~~4 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-~~8~~4

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#
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

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TABLE 3.7-~~0~~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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3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature limit of each area given in Table 3.7-~~6~~5 shall not be exceeded for more 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~~5 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~~5 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~~5 shall be determined to be within its limit at least once per 12 hours.

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TABLE 3.7-5

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

LIMITING CONDITION FOR OPERATION

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 4.8.1.1.3a.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 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 in accordance with the OPERABILITY of the applicable Fire Detection Instrumentation (Specification 3.3.3.7) and the applicable Fire Suppression Systems (Specification 3.7.10) for the ESF transformers, XNB01 and XNB02.~~

~~4.8.1.1.2 The solid-state load sequencer logic shall be demonstrated OPERABLE by performing an ACTUATION LOGIC TEST and a MASTER RELAY TEST at least once per 31 days and a SLAVE RELAY TEST at least once per 92 days.~~

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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 514 rpm in less than or equal to 12 seconds.* The generator voltage and frequency shall be 4000 ± 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 pow r 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. 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:

*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.

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;

f. 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 1352 kW (ESW pump) while maintaining voltage at 4000 ± 320 volts and frequency at 60 ± 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 4000 ± 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 auto-connected emergency (accident) load through the LOCA sequencer. The generator voltage and frequency shall be 4000 ± 320 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 for greater than or equal to 5 minutes while its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4000 ± 320 volts and 60 ± 1.2 Hz during this test; and

   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 4000 ± 320 volts and 60 + 1.2,-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 during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2f.6)b)*;

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*If Specification 4.8.1.1.2f.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.

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,

   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. 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 514 rpm in less than or equal to 12 seconds; and

h. 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, 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.

4.8.1.1.4 Reports - All diesel generator failures, valid or nonvalid, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 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.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 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 purpose of this schedule, only valid tests conducted after the 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."

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 connection resistance of these items is less than 150×10^{-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 anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm, and

4) The battery charger will supply at least 300 amperes at 134 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 all of the actual emergency loads for 200 minutes when the battery is subject to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, 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 battery 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.

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	3600(50)/372 (51) & 840 (51)	0.1	Reactor Coolant Pump DPBB01B
P-252PA0205	3600 (50)/372(51) & 840 (51)	0.1	Reactor Coolant Pump DPBB01C
P-252PA0204	3600 (50/372 (51) & 840 (51)	0.1	Reactor Coolant Pump DPBB01D
480-V 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
480-V LOAD CENTER (Continued)			
P-52PG2202 Through 52PG2212	375 (Inst.)	0.025	Pressurizer Backup Heaters
B-350 A Fuse		N.A.	
P-52NG01TAF1	2000 (Inst.)	0.016	Containment Cooler
B-52NG0108	~~2400~~ 2700 (S.T.	0.19	DSGN01A
P-52NG03TAF	2000 (Inst.)	0.016	Containment Cooler
B-52NG0305	~~2400~~ 2700 (S.T.)	0.19	DSGN01C
P-52NG02TAF1	2000 (Inst.)	0.016	Containment Cooler
B-52NG0208	~~2400~~ 2700 (S.T.)	0.19	DSGN01B
P-52NG04TAF1	2000 (Inst.)	0.016	Containment Cooler
B-52NG0405	~~2400~~ 2700 (S.T.)	0.19	DSGN01D
480 V MOTOR CONTROL CENTER			
P-52NG01BDF3	75 (Inst.)	0.016	RHR Loop Inlet Iso Vlv EJHV8701B
B-40A Fuse		N.A.	
P-52NG02BHR2	10 (Inst.)	0.016	ESW from Ctmt Air Coolers Iso Vlv EFHV46
B-15A Fuse		N.A.	
P-52NG02BDF2	45 (Inst.)	.016	CCW to Ctmt Iso Vlv EGHV60
B-40A Fuse		N.A.	

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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
480 V MOTOR CONTROL CENTER (Continued)			
P-52NG01BGF3 B-60A Fuse	565 (Inst.)	0.016 N.A.	Accumulator Iso Vlv EPHV8808A
P-52NG01BGF2 B-60A Fuse	565 (Inst.)	0.016 N.A.	Accumulator Iso Vlv EPHV8808C
P-52NG01BFF2 B-15A Fuse	10 (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 (Inst.)	0.016 N.A.	RHR Loop Inlet Iso Vlv BBPV8702B
P-52NG02BCF2 B-40A Fuse	75 (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

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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
480 V MOTOR CONTROL CENTER (Continued)			
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 DCGN02A
P-52PG19NCF3 B-60A Fuse	260 (Inst.)	0.016 N.A.	Ctmt Atmospheric Control System Fan DCGR01A
P-52PG19NGF2 B-40 Fuse	675 (Inst.)	0.017 N.A.	RCP A Space Heater
P-52PG19NGF3 B-40 Fuse	675 (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
480 V MOTOR CONTROL CENTER (Continued)			
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.	Instrumnt Tunnel Sump Pump DPLF07A
P-52NG03CBF4 B-15A Fuse	29 (Inst.	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV15
P-52NG03CLF2 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	675 (Inst.)	0.017 N.A.	RCP C Space Heater
P-52PG20NCF1 B-40A Fuse	675 (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
480 V MOTOR CONTROL CENTER (Continued)			
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-52PG20NDR2 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 HKE13
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)

(50) - Protective Relay Instantaneous Unit
(51) - Protective Relay Inverse Time Unit
Inst. - Instantaneous Protection
S.T. - Short Time Protection

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 analysis of its burnup history, 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.

and independently verified

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.

(6)Samples shall be taken at the initiation of effluent flow and at least once per 24 hours thereafter while the release is occurring. To be representative of the liquid effluent, the sample volume shall be proportioned to the effluent stream discharge volume. The ratio of sample volume to effluent discharge volume shall be maintained constant for all samples taken for the composite sample.

LIQUID HOLDUP TANKS

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 liner being used to solidify radioactive waste.

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 and within 7 days following any addition of radioactive material to the tank.

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]	1x10^{-4}
2. Containment Purge or Vent[3]	P Each PURGE[3] Grab Sample	P Each PURGE[3]	Principal Gamma Emitters[2]	1x10^{-4}
		M	H-3 (oxide)	1x10^{-6}
3. Unit Vent	M[3],[4] Grab Sample	M[3]	Principal Gamma Emitters[2]	1x10^{-4}
		M[4]	H-3 (oxide)	1x10^{-6}
4. Spent Fuel Building Exhaust	M[5] Grab Sample	M	Principal Gamma Emitters[2]	1x10^{-4}
		M[5]	H-3 (oxide)	1x10^{-6}
5. Radwaste Building Vent	M Grab Sample	M	Principal Gamma Emitters[2]	1x10^{-4}
6. All Release Types as listed in 1., 2., 3., 4., and 5. above	Continuous[6]	W[7] Charcoal Sample	I-131	1x10^{-12}
			I-333	1x10^{-10}
	Continuous[6]	W[7] Particulate Sample	Principal Gamma Emitters[2]	1x10^{-11}
	Continuous[6]	M Composite Particulate Sample	Gross Alpha	1x10^{-11}
	Continuous[6]	Q Composite Particulate Sample	Sr-89, Sr-90	1x10^{-11}

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 an lyzed 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. 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

GAS STORAGE TANKS

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 2.5×10^5 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 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank.

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 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. One sample downstream.	Composite sample over 1-month period(7).	Gamma isotopic analysis(5) monthly. Composite for tritium analysis of composite sample (by location) quarterly.
b. Drinking	One sample of each of one to three of the nearest water supplies within 10 miles downstream 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.(8) Composite for gross beta and gamma isotopic analyses(5) monthly. Composite for tritium analysis quarterly.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(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.

(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.

(7) 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.

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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.

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.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

3. The control rod insertion limits of Specification 3.1.3.6 are maintained; and

4. 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 1. through 4. 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 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 Multipler of 0.86 vs. 0.88, and
5) Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

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.

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.

Unscheduled inservice inspections are performed on each steam generator following: (1) primary to secondary tube leaks; (2) a seismic occurrence greater than the Operating Basis Earthquake; and (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 Regulatory Guide 1.121, which unplugged steam generator tubes must be capable of withstanding.

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 48% of the tube nominal wall thickness. Steam generator

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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 accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2 and 3/4.5.3 ^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.

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 1.5 psig for initial positive containment pressure will limit the total pressure to 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 vessel 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.

The required Special Reports from any engineering evaluation of containment abnormalties shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances 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.

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 2000 hours during a calendar year. The total time the Containment Purge (vent) System isolation values 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, should be used to support 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 3.6.1.2b. 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 eductor 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 GDC 54 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 Loss-of-Coolant Accident," Revision 2, November 1978.

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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 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 identified 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 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/m^2.

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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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.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The 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.

g. Net free volume = 2.5×10^6 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.

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6.1 RESPONSIBILITY

6.1.1 The Manager, Callaway Plant, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor (or during his absence from the control room, a designated individual) 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;

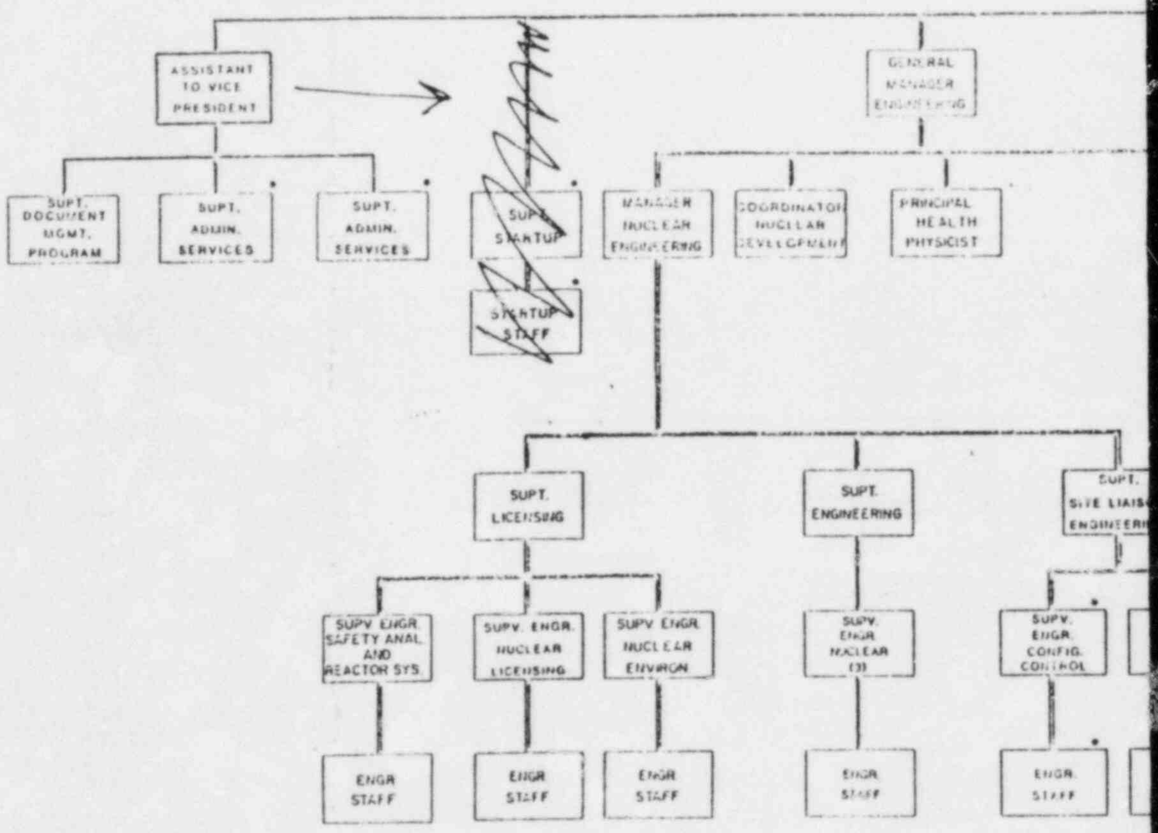
c. An individual from the Health Physics organization#, qualified in radiation protection procedures, shall be on site when fuel is in the reactor;

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 five members# shall be maintained onsite at all times. The Fire Brigade shall not include the Shift Supervisor, and the ~~(2)~~ other 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

two

#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.



* LOCATED ON SITE
----- COMMUNICATION PATH

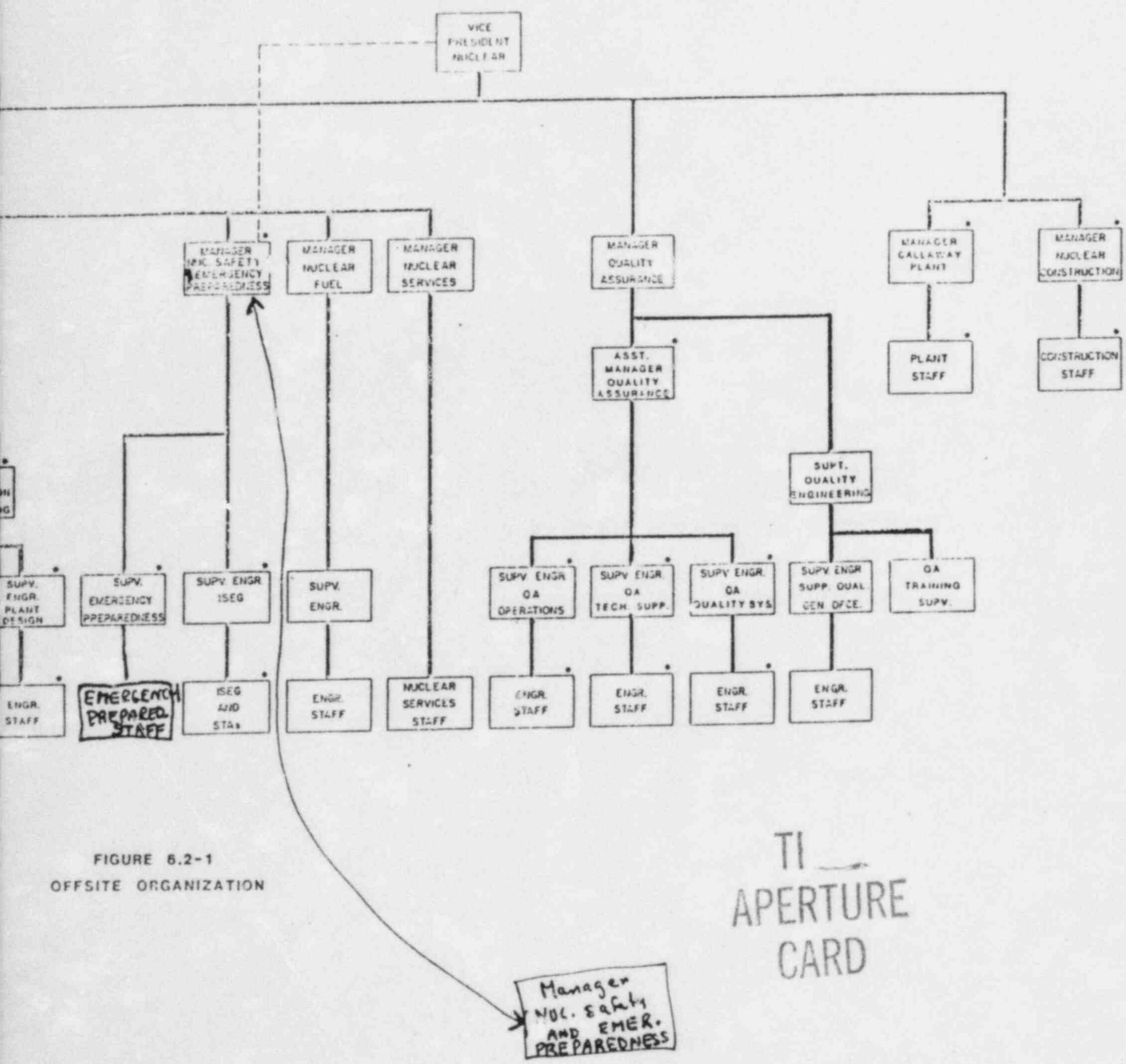


FIGURE 6.2-1
OFFSITE ORGANIZATION

TI
APERTURE
CARD

Also Available On
Aperture Card

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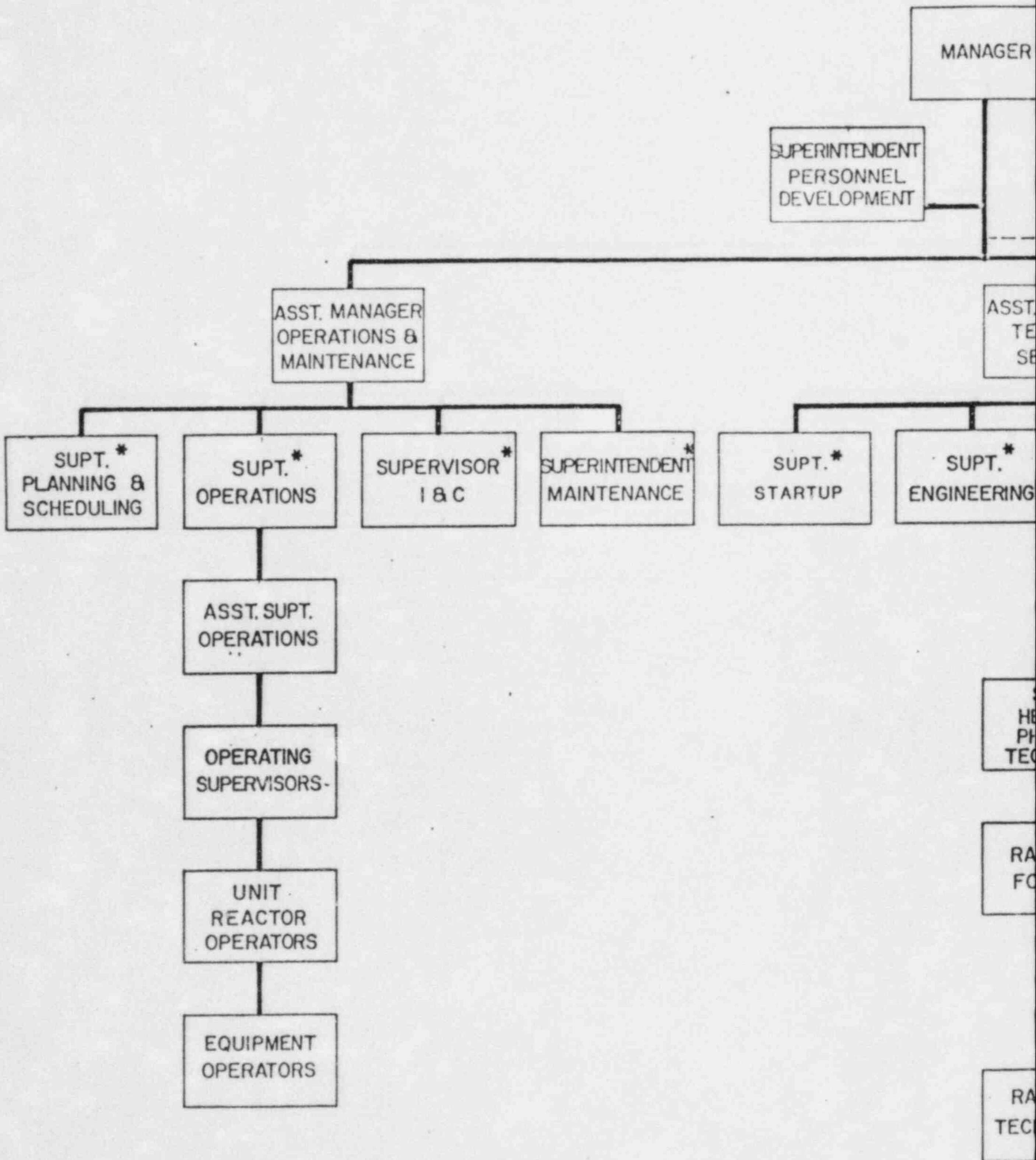

* DEPARTMENT HEAD

UNI

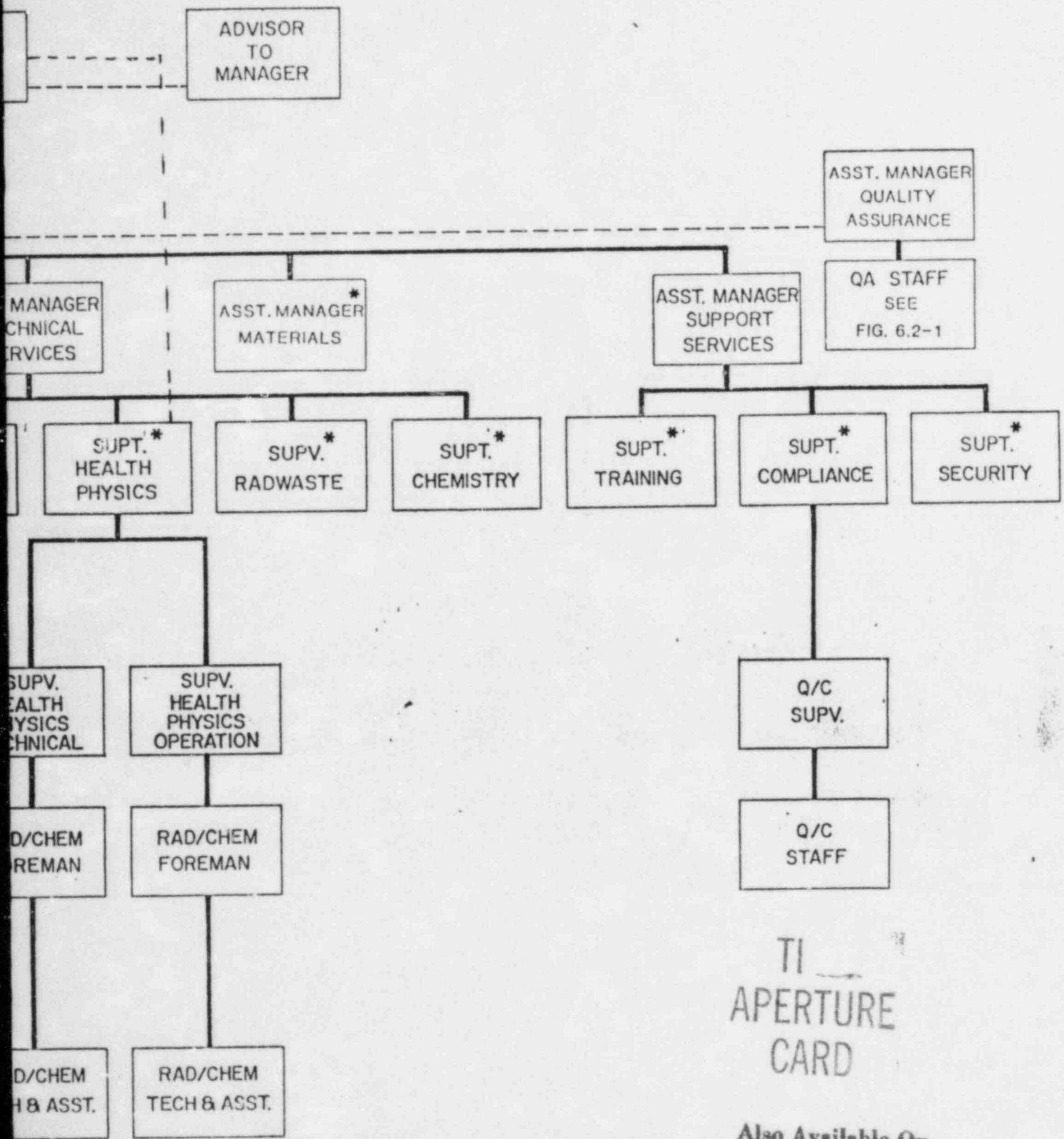

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FIGURE 6.2-2

T ORGANIZATION

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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 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 Manager, Nuclear Safety and Emergency Preparedness and the Manager, Callaway Plant.

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 the Manager, Nuclear Safety and Emergency Preparedness and the Manager, Callaway Plant.

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 Superintendent, Health Physics, 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.

*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.

RESPONSIBILITIES (Continued)

m. Review of Unit operations to detect potential hazards to nuclear safety;

n. Investigations or analysis of special subjects as requested by the Chairman of the NSRB; and

o. Review of Unit Turbine Overspeed Protection Reliability Program and revisions thereto.

6.5.1.7 The ORC shall:

a. Recommend in writing to the Manager, Callaway Plant approval or disapproval of items considered under Specifications 6.5.1.6a. through e., i., j., k., l., and o. above;

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6b. through e., and m., above, constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Vice President-Nuclear and the Nuclear Safety Review Board of disagreement between the ORC and the Manager, Callaway Plant; however, the Manager, Callaway Plant shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The ORC shall maintain written minutes of each ORC meeting that, at a minimum, document the results of all ORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President-Nuclear and the Nuclear Safety Review Board.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB 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,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NSRB shall report to and advise the Vice President-Nuclear on those areas of responsibility stated in Specifications 6.5.2.8 and 6.5.2.9.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by a qualified individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures shall be approved by the appropriate Department Head as designated in writing by the Manager, Callaway Plant. The Manager, Callaway Plant, shall approve Administrative Procedures, Security Plan implementing procedures and Radiological Emergency Response Plan implementing procedures. Temporary changes to procedures which do not change the intent of the approved procedures shall be approved for implementation by two members of the plant staff, at least one of whom holds a Senior Operator license, and documented. The temporary changes shall be approved by the original approval authority within 14 days of implementation. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change prior to implementation;

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Manager, Callaway Plant. Each such modification shall be reviewed by a qualified individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Proposed modifications to plant nuclear safety-related structures, systems and components shall be approved prior to implementation by the Manager, Callaway Plant;

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report or Technical Specifications shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by a qualified individual/group other than the individual/group which prepared the proposed test or experiment. Proposed test and experiments shall be approved before implementation by the Manager, Callaway Plant;

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB 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 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 No. 82-33;

c. Plant Security Plan implementation;

d. Radiological Emergency Response Plan implementation;

e. PROCESS CONTROL PROGRAM implementation,

f. OFFSITE DOSE CALCULATION MANUAL implementation,

g. Quality Assurance Program implementation for effluent and environmental monitoring, and

h. Turbine Overspeed Protection Reliability Program.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 The plant Administrative Procedures and changes thereto shall be reviewed in accordance with Specification 6.5.1.6 and approved in accordance with Specification 6.5.3.1. The associated implementing procedures and changes thereto shall be reviewed and approved in accordance with Specification 6.5.3.1.

6.8.4 The following programs shall be established, implemented, and maintained:

a. 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 recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

PROCEDURES AND PROGRAMS (Continued)

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.

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.

e. Turbine Overspeed Protection Reliability Program

(see following page)

e. Turbine Overspeed Protection Reliability Program

A program to increase the assurance that the turbine overspeed protection system functions, if challenged, and to assure structural integrity of turbine components which could result in missile generation in the event of an actual overspeed occurrence. The program shall include the following:

1) Periodic testing and inspection requirements,
2) Specification of test and inspection intervals, and
3) Administrative restrictions and procedural guidance for program implementation such as: record keeping; reporting, evaluation and disposition of discrepancies; review and approval of revisions to the program; and authorization(s) required to deviate from the program guidelines.

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

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 SOLIDICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall 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. All assumptions used in making these assessments, i.e., historical average atmospheric conditions, 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 OFFSITE DOSE CALCULATION MANUAL (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 with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power

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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.

ATTACHMENT 3

Specifications Which are Being Appealed

REACTOR COOLANT SYSTEM

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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 ± 1%, 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 one PORV and one RHR suction relief valve inoperable, either restore ~~the inoperable PORV~~ two PORV's 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 and 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, 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, 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.

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4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an 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.

4.4.9.3.~~2~~3 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.

4.4.9.3.2 The RHR suction relief valves shall be demonstrated OPERABLE by:

a. Verifying that RHR RCS suction isolation valves:

1) 8701B and 8702A are opened with power to the valve operator removed at least once per 31 days, and

2) 8701A and 8702B are opened at least once per 12 hours,

whenever the RHR suction relief valves are being used for cold overpressure protection.

b. Testing pursuant to specification 4.0.5.

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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 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

DRAFT

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 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.;

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 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 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 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 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

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SURVEILLANCE REQUIREMENTS (Continued)

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.

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3/4.6.3 CONTAINMENT ISOLATION VALVES

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 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.

ELECTRICAL POWER SYSTEMS

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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.

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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 charger 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:

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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HEATUP (Continued)

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.

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, 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 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 1 →

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 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 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 1

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 3.4.9.3.c, 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.

DRAFT

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DRAFT

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DRAFT

INDEX

BASES

SECTION PAGE

REACTOR COOLANT SYSTEM (Continued)

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

DRAFT

3/4.1.2 BORATION SYSTEMS

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, or 3.1.2.6a (as dictated by MODE) is OPERABLE; 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. or 3.1.2.6b (as dictated by MODE) is OPERABLE.

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.

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BORATED WATER SOURCE - MODES 5 & 6 ~~SHUTDOWN~~

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 2968 ~~2713~~ 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 55,416 ~~53,500~~ 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 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.

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BORATED WATER SOURCES - MODE 4 ~~OPERATING~~

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, one of the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.~~2~~1:

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 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: MODE~~S~~ ~~1, 2, 3, and~~ 4.

ACTION:

a. With no borated water source OPERABLE, ~~the Boric Acid Storage System inoperable and being used as one of the above required borated water sources,~~ restore the storage system to OPERABLE status within ~~72~~ 6 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~~ 24 hours.

b. ~~With the RWST inoperable, 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.~~

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required borated water source shall be demonstrated OPERABLE by the performance of each of the requirements of Specification 4.1.2.5.

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BORATED WATER SOURCES - ~~OPERATING~~ MODES 1,2 & 3

LIMITING CONDITION FOR OPERATION

3.1.2.~~6~~ 7 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

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 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 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% $\Delta k/k$ 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, 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.

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4.1.2.~~6~~ 7 Each 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.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

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 Specifications 3.10.2 and 3.10.3.

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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 to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

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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)

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POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod 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.

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POSITION INDICATION SYSTEM-SHUTDOWN

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 Specification 3.10.5.

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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.

REACTIVITY CONTROL SYSTEMS

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SHUTDOWN ROD INSERTION LIMIT

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 to be 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.

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CONTROL ROD INSERTION LIMITS

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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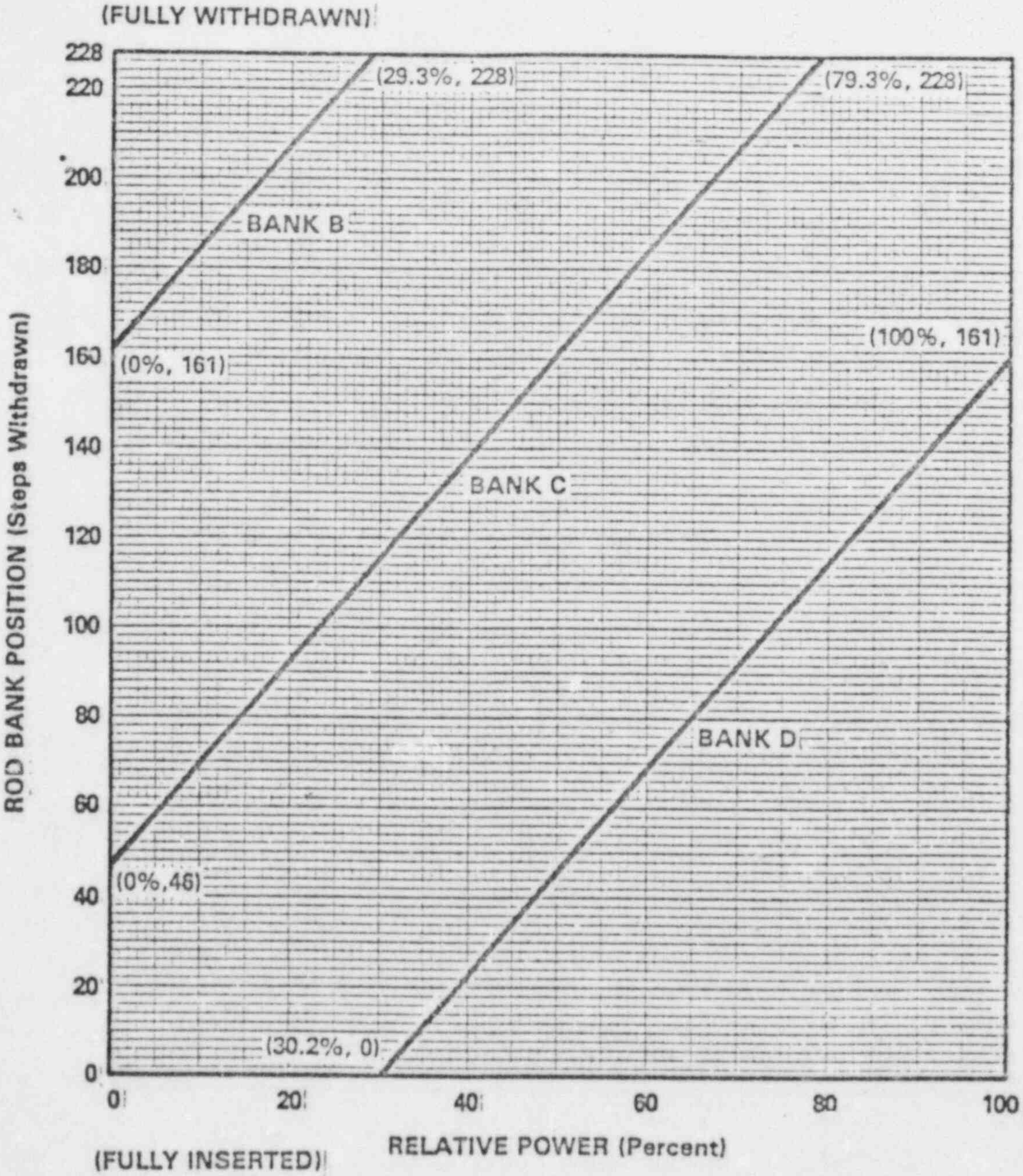

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS
THERMAL POWER - FOUR LOOP OPERATION

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

ATTACHMENT 4

Specifications Not Resolved

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LIMITING CONDITION FOR OPERATION

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 loop 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 measurement, the instrumentation used for determination of system pressure, feedwater pressure, feedwater temperature, and feedwater venturi Δp in the calorimetric calculation shall be calibrated.

4.2.3.6 The feedwater venturis shall ' inspected and cleaned if necessary at least once per 18 months.

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

4. Triaxial Seismic Switches	ACCELERATION LEVEL/DIRECTION	n	E	V	
a. OBE Ctmt. Base Slab	~~0.12 g~~	.09g	.09g	.13g	1
b. SSE Ctmt. Base Slab	~~0.20 g~~	.13g	.14g	.20g	1
c. OBE Ctmt. Oper. Fl.	~~0.13 g~~	.10g	.10g	.13g	1
d. SSE Ctmt. Oper. Fl.	~~0.24 g~~	.14g	.16g	.21g	1
e. System Trigger	~~0.01 g~~	.01g	.01g	.01g	1

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Normal Range	2	1
b. Extended Range	2	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
13. PORV Position Indicator*	1/Valve	1/Valve
14. PORV Block Valve Position Indicator**	1/Valve	1/Valve
15. Safety Valve Position Indicator	1/Valve	1/Valve
16. Containment Water Level	2	1
17. Containment Radiation Level (High Range)	2	1
18. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
~~19. Reactor Coolant Radiation Level#~~	~~1/loop~~	~~1/loop~~
20. Unit Vent - High Range Noble Gas Monitor	1	1
21. Reactor Vessel Water Level#	2	1
22. Steam Relief - Noble Gas Monitor#	4	4
23. Source Range - Neutron Flux#	2	1
24. Auxiliary Feedwater Pump Turbine Exhaust - Noble Gas Monitor#	2	1

TABLE NOTATIONS

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

*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.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	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
13. PORV Position Indicator*	M	N.A.
14. PORV Block Valve Position Indicator**	M	N.A.
15. Safety Valve Position Indicator	M	N.A.
16. Containment Water Level	M	R
17. Containment Radiation Level (High Range)	M	R***
18. Thermocouple/Core Cooling Detection System	M	R

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
~~19. Reactor Coolant Radiation Level#~~	~~M~~	~~R~~
20. Unit Vent - High Range Noble Gas Monitor	M	R
21. Reactor Vessel Water Level#	M	R
22. Steam Relief - Noble Gas Monitor#	M	R
23. Source Range - Neutron Flux#	M	R
24. Auxiliary Feedwater Pump Turbine Exhaust - Noble Gas Monitor#	M	R

TABLE NOTATIONS

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

*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 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

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 Automation Termination of Release (~~GT~~ GH-RE-10)	1	*	38, 40
b. Iodine Sampler	1	*	43
c. Particulate Sampler	1	*	43
d. Flow Rate	N.A.	*	45
e. Sampler Flow Rate Monitor	1	*	39

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 (~~GT~~ GH-RE-10)	D, P	M, P	R(3)	Q(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	*

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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, restor the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ 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~~5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.~~6~~5 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 solution temperature of 37°F, and

d. A maximum 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~~4 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.

SURVEILLANCE REQUIREMENTS (Continued)

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 test by verifying that the containment leakage rate calculated in accordance with ANSI N45.4-1972, Appendix C, is within 25% of the containment leakage rate measured prior to the introduction of the superimposed leak,

   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 gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

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.

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 Specifications 4.6.1.7.2 and 4.6.1.7.4, as applicable; and

g. The provisions of Specification 4.0.2 are not applicable.

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 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;

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.

Change to 0.005L is due to the reduced pressure — leakage proportional to $\sqrt{P}$

#The provisions of Specification 4.0.2 are not applicable.
*This represents an exemption to Appendix J of 10 CFR Part 50.

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3/4.7.11 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.11 All fire barrier penetrations (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, and cable, piping, and ventilation duct penetration seals) shall be OPERABLE. tray

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations 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 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 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 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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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 $-4.1 \times 10^{-4}\ \Delta k/k/^{\circ}F$. The MTC value of $-3.2 \times 10^{-4}\ \Delta k/k/^{\circ}F$ 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 $-4.1 \times 10^{-4}\ \Delta k/k/^{\circ}F$.

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 its 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 ensure 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~~ 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~~ a single 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 ~~12,117~~ gallons of 7000 ppm borated water from the boric acid storage tanks or 83,754 ~~72,096~~ gallons of 2000 ppm borated water from the RWST.

In MODE 4, a single flowpath is required to be OPERABLE as is necessary to assure that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve.

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~~ gallons of 7000 ppm borated water from the boric acid storage tanks or ~~12,117~~ gallons of 2000 ppm borated water from the RWST.
14,076 2968

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.

DRAFT

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

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 maneuvers 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% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, ~~a penalty of 0.1% for undetected fouling of~~ an inspection
of the feedwater venturi is ~~included in Figure 3.2-3. Any fouling which might~~ is performed
each ~~bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring~~
refueling ~~and trending various plant performance parameters~~. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measuremet or the venturi shall be cleaned to eliminate the fouling.

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.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging 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.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure, 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 ensure that a failure of one valve will not cause an intersystem LOCA. The Surveillance Requirement to vent the ECCS pump casings and accessible, i.e., 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.

~~3/4.5.4 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.~~5~~4 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