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AUG 8 1983

Docket No. 50-416

Mr. J.P. McGaughy, Jr.
Assistant Vice President - Nuclear Production
Mississippi Power & Light Company
P.O. Box 1640
Jackson, Mississippi 39205

Dear Mr. McGaughy:

Subject: Amendment No. 8 to Facility Operating License No. NPF-13 - Grand Gulf Nuclear Station, Unit 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 8 to Facility Operating License No. NPF-13 for the Grand Gulf Nuclear Station, Unit 1. This Amendment is in response to MP&L letters dated March 24, 1983, April 7, 1983, April 25, 1983, June 9, 1983, June 14, 1983, June 23, 1983, and June 29, 1983, which you submitted in partial response to the NRC Confirmation of Action (COA) letter of October 20, 1982. That COA letter called for MP&L to prepare and submit license amendment requests, where necessary, to correct administrative and technical deficiencies in your Technical Specifications during MP&L's review of the Grand Gulf Unit 1 surveillance procedures.

The bulk of the changes approved in Amendment No. 8 are administrative in nature and are necessary to correct editorial and nomenclature errors and to achieve consistency with the as-built condition of the plant. None of the changes involve a significant relaxation of the criteria used to establish safety limits or the bases for limiting safety system settings or limiting conditions for operation.

A copy of the related staff evaluation supporting Amendment No. 8 to Facility Operating License NPF-13 is enclosed. Also enclosed is a copy of a related notice which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by

the Office of ELD has NLO 8/5/83

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
1. Amendment No. 8 to NPF-13
2. Staff Evaluation
3. Federal Register Notice

cc w/ enclosures:
See next page

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SURNAME	DHouston:pt	EHylton	MWagner	ASchwencer	TNovak	
DATE	7/15/83	7/ /83	7/25/83	8/8/83	8/8/83	

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MISSISSIPPI POWER AND LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
DOCKET NO. 50-416
GRAND GULF NUCLEAR STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

License No. NPF-13
Amendment No. 8

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The applications for the amendment filed by the Mississippi Power and Light Company dated March 24, 1983, April 7, 1983, and April 25, 1983, June 9, 1983, June 14, 1983, June 23, 1983, and June 29, 1983, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended as follows:

   A. Page changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) to read as follows:

      (2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 8, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Date of Issuance: August 8 , 1983

*SEE PREVIOUS PAGE FOR CONCURRENCES

OFFICE	DL:LB#2/PM	DL:LB#2/LA	OELD	DL:LB#2/BC	DL:AD		
SURNAME	DHouston*:kw	EHylton	MWagner*	ASchwencer	TNovak		
DATE	7/15/83	8/ /83	7/25/83	8/8/83	8/8/83		

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Date of Issuance: July , 1983

OFFICE	DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD			
SURNAME	DHouston:pt	EHylton	ASchwencer	WgWagner MW			
DATE	7/15/83	7/ /83	7/ /83	7/25/83			

ATTACHMENT TO LICENSE AMENDMENT NO. 8
FACILITY OPERATING LICENSE NO. NPF-13
DOCKET NO. 50-416

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. This revised page is identified by Amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
2-4	2-4
3/4 1-5	3/4 1-5
3/4 2-5	3/4 2-5
3/4 3-11	3/4 3-11
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-28	3/4 3-28
3/4 3-29	3/4 3-29
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-41	3/4 3-41
3/4 3-56	3/4 3-56
3/4 3-57	3/4 3-57
3/4 3-95	3/4 3-95
3/4 4-18	3/4 4-18
3/4 5-9	3/4 5-9
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-7	3/4 6-7
3/4 6-15	3/4 6-15
3/4 6-16	3/4 6-16
3/4 6-21	3/4 6-21
3/4 6-22	3/4 6-22
3/4 6-45a	3/4 6-45a

REMOVE	INSERT
3/4 6-54	3/4 6-54
3/4 6-58	3/4 6-58
3/4 7-4	3/4 7-4
3/4 7-6	3/4 7-6
3/4 7-16	3/4 7-16
3/4 7-26	3/4 7-26
-	3/4 7-45
3/4 8-1	3/4 8-1
3/4 8-3	3/4 8-3
3/4 8-6	3/4 8-6
3/4 8-13	3/4 8-13
3/4 8-21	3/4 8-21
3/4 8-40	3/4 8-40
3/4 8-41	3/4 8-41
3/4 8-42	3/4 8-42
3/4 8-43	3/4 8-43
3/4 8-44	3/4 8-44
3/4 8-45	3/4 8-45
-	3/4 8-45a
3/4 11-1	3/4 11-1
3/4 11-5	3/4 11-5
3/4 11-12	3/4 11-12
3/4 11-13	3/4 11-13
6-11	6-11

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	≤ 0.66 W+48%, with a maximum of	≤ 0.66 W+51%, with a maximum of
2) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 11.4 inches above instrument zero*	≥ 10.8 inches above instrument zero*
5. Reactor Vessel Water Level-High, Level 8	≤ 53.5 inches above instrument zero*	≤ 54.1 inches above instrument zero*
6. Main Steam Line Isolation Valve - Closure	≤ 6% closed	≤ 7% closed
7. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
8. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
9. Scram Discharge Volume Water Level - High	≤ 60% of full scale	≤ 63% of full scale
10. Turbine Stop Valve - Closure	≥ 40 psig**	≥ 37 psig
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 44.3 psig**	≥ 42 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

   1. Close within 30 seconds after receipt of a signal for control rods to scram, and

   2. Open when the scram signal is reset.

b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation at least once per 31 days.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-high scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

Trip Setpoint	Allowable Value
$S \leq (0.66W + 48\%)T$	$S \leq (0.66W + 51\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER.

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 112.5 million lbs/hr.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD). T is always less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the allowable value-column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} to within the required limits* within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP AND MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-high scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	6[c][d][h]	2	1, 2, 3, and #	25
b. Drywell Pressure - High	6[c][d][h]	2	1, 2, 3	25
c. Fuel Handling Area Ventilation Exhaust Radiation - High High	N.A.[j]	2	1, 2, 3, and *	25
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	N.A.[j]	2	1, 2, 3, and *	25
e. Manual Initiation	6[f] 6[f]	1/group 1/group	1, 2, 3 *	26 25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	8	1	1, 2, 3	27
b. Δ Flow Timer	8	1	1, 2, 3	27
c. Equipment Area Temperature - High	8	1	1, 2, 3	27
d. Equipment Area Δ Temp. - High	8	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	8	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	8	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	8	1	1, 2, 3	27
h. SLCS Initiation	8[i]	NA	1, 2, 3	27
i. Manual Initiation	8	1/group	1, 2, 3	26

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

NOTES

* When handling irradiated fuel in the containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the isolation mode of operation.

(e) Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

(f) Also trips and isolates the mechanical vacuum pumps.

(g) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(h) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.6.2-1.

(i) Closes only RWCU system isolation valves G33-F001, G33-F004, and G33-F251.

(j) Actuates the Standby Gas Treatment System and isolates Auxiliary Building penetration of the ventilation systems within the Auxiliary Building.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches *	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
c. Containment and Drywell Ventilation Exhaust Radiation - High High	≤ 2.0 mr/hr**	≤ 4.0 mr/hr**
d. Manual Initiation	NA	NA
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 169 psid	≤ 176.5 psid
e. Condenser Vacuum - Low	≥ 9 inches Hg. Vacuum	≥ 8.7 inches Hg. Vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 185°F**	≤ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	≤ 101°F**	≤ 104°F**
h. Manual Initiation	NA	NA
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
c. Fuel Handling Area Ventilation Exhaust Radition - High High	≤ 2.0 mR/hr**	≤ 4.0 mR/hr**
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	≤ 18 mR/hr**	≤ 35 mR/hr**
e. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	$\leq$ 79 gpm	$\leq$ 89** gpm
b. Δ Flow Timer	$\leq$ 45 seconds	$\leq$ 57 seconds
c. Equipment Area Temperature - High		
1. RWCU Hx Room	$\leq$ 124°F	$\leq$ 130°F
2. RWCU Pump Rooms	$\leq$ 174°F	$\leq$ 180°F
3. RWCU Valve Nest Room	$\leq$ 139°F	$\leq$ 145°F
4. RWCU Demin. Rooms	$\leq$ 139°F	$\leq$ 145°F
5. RWCU Rec. Tank Room	$\leq$ 139°F	$\leq$ 145°F
6. RWCU Demin. Valve Room	$\leq$ 135°F	$\leq$ 141°F
d. Equipment Area Δ Temp. - High		
1. RWCU Hx Room	$\leq$ 65°F	$\leq$ 66°F
2. RWCU Pump Rooms	$\leq$ 115°F	$\leq$ 118°F
3. RWCU Valve Nest Room	$\leq$ 70°F	$\leq$ 73°F
4. RWCU Demin Rooms	$\leq$ 70°F	$\leq$ 73°F
5. RWCU Rec. Tank Room	$\leq$ 70°F	$\leq$ 73°F
6. RWCU Demin. Valve Room	$\leq$ 71°F	$\leq$74°F
e. Reactor Vessel Water Level - Low Low, Level 2	$\geq$ -41.6 inches*	$\geq$ -43.8 inches
f. Main Steam Line Tunnel Ambient Temperature - High	$\leq$ 185°F**	$\leq$ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	$\leq$ 101°F**	$\leq$ 104°F**
h. SLCS Initiation	NA	NA
i. Manual Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	$\leq$ 363" H_2O	$\leq$ 371" H_2O
b. RCIC Steam Supply Pressure - Low	$\geq$ 60 psig	$\geq$ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 20 psig

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
REACTOR CORE ISOLATION COOLING SYSTEM (Continued)		
d. RCIC Equipment Room Ambient Temperature - High	≤ 189°F**	≤ 195°F**
e. RCIC Equipment Room Δ Temp. - High	≤ 125°F**	≤ 128°F**
f. Main Steam Line Tunnel Ambient Temperature - High	≤ 185°F**	≤ 191°F**
g. Main Steam Line Tunnel Δ Temp. - High	≤ 101°F**	≤ 104°F**
h. Main Steam Line Tunnel Temperature Timer	≤ 30 minutes	≤ 30 minutes
i. RHR Equipment Room Ambient Temperature - High	≤ 169°F**	≤ 175°F**
j. RHR Equipment Room Δ Temperature - High	≤ 105°F**	≤ 108°F**
k. RHR/RCIC Steam Line Flow - High	≤ 145" H_2O	≤ 160" H_2O
l. Manual Initiation	NA	NA
6. RHR SYSTEM ISOLATION		
a. RHR Equipment Room Ambient Temperature - High	≤ 169°F**	≤ 175°F**
b. RHR Equipment Room Δ Temperature - High	≤ 105°F**	≤ 108°F**
c. Reactor Vessel Water Level - Low, Level 3	≥ 11.4 inches*	≥ 10.8 inches
d. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 135 psig	≤ 150 psig
e. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
f. Manual Initiation	NA	NA

*See Bases Figure B 3/4 3-1.

**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.89 psig	$\leq$ 1.94 psig
c. LPCI Pump A Start Time Delay Relay	$\leq$ 5 seconds	$\leq$ 5.25 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.89 psig	$\leq$ 1.94 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	$\geq$ 11.4 inches*	$\geq$ 10.8 inches
e. LPCS Pump Discharge Pressure-High	145 psig, increasing	125-165 psig, increasing
f. LPCI Pump A Discharge Pressure-High	125 psig, increasing	115-135 psig, increasing
g. Manual Initiation	NA	NA
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.89 psig	$\leq$ 1.94 psig
c. LPCI Pump B Start Time Delay Relay	$\leq$ 5 seconds	$\leq$ 5.25 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.89 psig	$\leq$ 1.94 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	$\geq$ 11.4 inches*	$\geq$ 10.8 inches
e. LPCI Pump B and C Discharge Pressure-High	125 psig, increasing	115 psig, increasing
f. Manual Initiation	NA	NA
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$-41.6 inches*	$\geq$-43.8 inches
b. Drywell Pressure - High	$\leq$ 1.89 psig	$\leq$ 1.94 psig
c. Reactor Vessel Water Level - High, Level 8	$\leq$ 53.5 inches*	$\leq$ 55.7 inches
d. Condensate Storage Tank Level - Low	$\geq$ 0 inches	$\geq$ -3 inches
e. Suppression Pool Water Level - High	$\leq$ 5.9 inches	$\leq$ 6.5 inches
f. Manual Initiation	NA	NA

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER		
1. Division 1 and 2		
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	1. 4.16 kV Basis 2912 volts	2912 +0, -291 volts
	2. 120 volt Basis 83.2 volts	83.2 +0, -8.3 volts
	3. Time Delay 0.5 seconds	0.5 +0.5, -0.1 seconds
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	1. 4.16 kV Basis 3328 volts	3328 +0, -167 volts
	2. 120 volt Basis 95.1 volts	95.1 +0, -4.8 volts
	3. Time delay 0.5 seconds	0.5 +0.5, -0.1 seconds
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	1. 4.16 kV Basis 3744 volts	3744 +93.6, -0 volts
	2. 120 volt Basis 107 volts	107 +2.7, -0 volts
	3. Time Delay 9.0 seconds	9.0 ± 0.5 seconds
2. Division 3		
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	1. 4.16 kV Basis 3045 volts	3045 ± 61 volts
	2. 120 volt Basis 87 volts	87 ± 1.7 volts
	3. Time Delay 2.3 seconds	2.3 + 0.2, -0.3 seconds

*See Bases Figure B 3/4 3-1.

#These are inverse time delay voltage relays or instantaneous voltage relays with a time delay. The voltages shown are the maximum that will not result in a trip. Lower voltage conditions will result in decreased trip times.

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES (SECONDS)

1.	LOW PRESSURE CORE SPRAY SYSTEM	$\leq$ 40
2.	LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM	
	a. Pumps A and B	$\leq$ 45
	b. Pump C	$\leq$ 40
3.	AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4.	HIGH PRESSURE CORE SPRAY SYSTEM	$\leq$ 27
5.	LOSS OF POWER	NA

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R(b)(c)	Q(d)	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3
e. LPCS Pump Discharge Pressure-High	S	M	R	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	S	M	R(a)	1, 2, 3
g. Manual Initiation	NA	R(b)	NA	1, 2, 3
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R(b)(c)	Q(d)	1, 2, 3, 4*, 5*

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM (Continued)				
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	S	M	R(a)	1, 2, 3
f. Manual Initiation	NA	R(b)	NA	1, 2, 3
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. Reactor Vessel Water Level-High, Level 8	S	M	R(a)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R(a)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(a)	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R(b)	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Division 1 and 2				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	M(e)	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	NA	M(e)	R	1, 2, 3, 4**, 5**
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	M(e)	R	1, 2, 3, 4**, 5**
2. Division 3				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

NOTATION

- \# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.
- * Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
- ** Required when ESF equipment is required to be OPERABLE.
- (a) Calibrate trip unit at least once per 31 days.
- (b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.
- (c) Manual initiation test shall include verification of the OPERABILITY of the LPCS and LPCI injection valve interlocks. (See Note 1)
- (d) This calibration shall consist of the CHANNEL CALIBRATION of the LPCS and LPCI injection valve interlocks with the interlock setpoint verified to be $\leq$ 150 psig. (See Note 1)
- (e) Functional Testing of Time Delay Not Required

Note 1: Until restart after the first refueling outage, the requirements of (c) and (d) above do not apply.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve - Closure	≥ 40 psig*	≥ 37 psig
2. Turbine Control Valve - Fast Closure	≥ 44.3 psig*	≥ 42 psig

*Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Component Cooling Water Radiation Monitor	1	At all times	$\le 1 \times 10^5$ cpm/NA	10 to 10^6 cpm	70
2. Standby Service Water System Radiation Monitor	1/heat exchanger train	1, 2, 3, and*	$\le 1 \times 10^5$ cpm/NA	10 to 10^6 cpm	70
3. Offgas Pre-treatment Radiation Monitor	1	1, 2	$\le 5 \times 10^3$ mR/hr/NA	1 to 10^6 mR/hr	70
4. Offgas Post-treatment Radiation Monitor	2[a]	1, 2	$\le 1 \times 10^5$ cpm (Hi), $\le 1.0 \times 10^6$ cpm (Hi Hi Hi)	10 to 10^6 cpm	71
5. Carbon Bed Vault Radiation Monitor	1	1, 2	$\le$ 2 x full power background/NA	1 to 10^6 mR/hr	72
6. Control Room Ventilation Radiation Monitor	2	1,2,3,5 and**	$\le$4 mR/hr/ $\le$5 mR/hr#	10^{-2} to 10^2 mR/hr	73
7. Containment and Drywell Ventilation Exhaust Radiation Monitor	3[h]	At all times	$\le$2.0 mR/hr/ $\le$4 mR/hr[b]#	10^{-2} to 10^2 mR/hr	74
8. Fuel Handling Area Ventilation Exhaust Radiation Monitor	3[h]	1,2,3,5 and**	$\le$ 2mR/hr/ $\le$4 mR/hr[d]#	10^{-2} to 10^2 mR/hr	75
9. Fuel Handling Area Pool Sweep Exhaust Radiation Monitor	3[h]	(c)	$\le$ 18 mR/hr/ $\le$35 mR/hr[d]#	10^{-2} to 10^2 mR/hr	75

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
10. Area Monitors					
a. Fuel Handling Area Monitors					
1) New Fuel Storage Vault	1	(e)	$\leq$2.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72
2) Spent Fuel Storage Pool	1	(f)	$\leq$2.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72
3) Dryer Storage Area		(g)	$\leq$2.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72
b. Control Room Radiation Monitor	1	At all times	$\leq$0.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72

* With RHR heat exchangers in operation.

** When irradiated fuel is being handled in the primary or secondary containment.

\# Initial setpoint. Final Setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to Commission within 90 days after test completion.

(a) Trips system with 2 channels upscale-Hi Hi Hi, or one channel upscale Hi Hi Hi and one channel downscale or 2 channels downscale.

(b) Isolates containment/drywell purge penetrations.

(c) With irradiated fuel in spent fuel storage pool.

(d) Also isolates the Auxiliary Building and Fuel Handling Area Ventilation Systems.

(e) With fuel in the new fuel storage vault.

(f) With fuel in the spent fuel storage pool.

(g) With fuel in the dryer storage area.

(h) Two upscale Hi Hi, one upscale Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated isolation values.

TABLE 4.3.7.12-2 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

*** During operation of the main condenser air ejector.

#SOURCE CHECK may be deferred to the next shutdown of greater than 8 hours duration if unable to be performed at the monthly interval due to inaccessibility because of being in a high radiation area.

##The sensor will be calibrated for mr/hr or cpm from the calibration standard. The conversion to release rate will be performed during subsequent unit operation, but within one week.

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

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

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

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

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

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figure 3.4.6.1-1. The adjusted reference temperature resulting from neutron irradiation shall be calculated based on the greater of the following:

a. Actual shift in the RT_{NDT} for materials in the capsules as defined by the CVN impact test.

b. Predicted shift in RT_{NDT} for plate C2594-2 and weld 627260/B322A27AE (heat/lot) as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials".

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 70°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

   2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool water level instrumentation division inoperable, restore the inoperable division to OPERABLE status within 7 days or verify the suppression pool water level to be greater than or equal to 18'4-3/4" or 12'8", as applicable, at least once per 12 hou's by an alternate indicator.

d. With both suppression pool water level instrumentation divisions inoperable, restore at least one inoperable division to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and verify the suppression pool water level to be greater than or equal to 18'4-3/4" or 12'8", as applicable, at least once per 12 hours by at least one alternate i.ndicator.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool shall be determined OPERABLE by verifying:

a. The water level to be greater than or equal to, as applicable:

1. 18'4-3/4" at least once per 24 hours.

2. 12'5" at least once per 12 hours.

b. Two suppression pool water level instrumentation divisions, with 1 channel per division, OPERABLE with the low water level alarm setpoint $\geq$ 18'5½" or 12'8", as applicable, by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months.

4.5.3.2 With the suppression pool level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 2 scf per hour at P_a, 11.5 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one containment air lock door inflatable seal system seal pressure instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the associated inflatable seal pressure to be $\geq$ 60 psig at least once per 12 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours after each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 2 scf per hour when the gap between the door seals is pressurized to Pa, 11.5 psig.

b. By conducting an overall air lock leakage test at P_a, 11.5 psig, and verifying that the overall air lock leakage rate is within its limit:

1. At least once per 6 months#, and

2. Prior to establishing PRIMARY 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.

d. By verifying each airlock door inflatable seal system OPERABLE by:

1. Demonstrating each of the two inflatable seal pressure instrumentation channels per airlock door OPERABLE by performance of a:

a) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b) CHANNEL CALIBRATION at least once per 18 months,

with a low pressure setpoint of $\geq$ 60 psig.

2. At least once per 7 days, verifying seal air flask pressure to be greater than or equal to 60 psig.

3. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 90 psig within 48 hours.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

CONTAINMENT SYSTEMS

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSIV leakage control system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1. Blower OPERABILITY by starting the blowers from the control room and operating the blowers for at least 15 minutes.

   2. Heater OPERABILITY by demonstrating electrical continuity of the heating element circuitry.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor operated valve through at least one complete cycle of full travel.

c. At least once per 18 months by:

   1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position, the blowers start and the heater draws 7.8 to 9.5 amperes per phase.

   2. Verifying that the blower developed at least the below required vacuum at the rated capacity.

      a) Inboard valves, 15" $\pm$1" H_2O at 100 scfm.

      b) Outboard valves, 50" $\pm$ 2" H_2O at 200 scfm.

d. By verifying the inboard flow, inboard and outboard pressure, and inboard temperature instrumentation to be OPERABLE by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,
   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
   3. CHANNEL CALIBRATION at least once per 18 months.

CONTAINMENT SYSTEMS

DRYWELL AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.2.3 Each drywell air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the drywell, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 2 scf per hour at P_a, 11.5 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With one drywell air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

   2. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

   3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   4. The provisions of Specification 3.0.4 are not applicable.

b. With the drywell air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one drywell air lock door inflatable seal system seal pressure instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 7 days or verify the associated inflatable seal pressure to be $\geq$ 60 psig at least once per 12 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each drywell air lock shall be demonstrated OPERABLE:

a. Within 8 hours after each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 2 scf per hour when the gap between the door seals is pressurized to P_a, 11.5 psig.

b. At least once per 6 months by conducting an overall air lock leakage test at P_a, 11.5 psig and by verifying that the overall air lock leakage rate is within its limit.#

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

d. By verifying each airlock door inflatable seal system OPERABLE by:

1. Demonstrating each of the two inflatable seal pressure instrumentation channels per airlock door OPERABLE by performance of a:

a) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b) CHANNEL CALIBRATION at least once per 18 months,

with a low pressure setpoint of $\geq$ 60 psig.

2. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 60 psig.

3. At least once per 18 months, conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 2 psig from 90 psig within 48 hours.

#The provisions of Specification 4.0.2 are not applicable.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one suppression pool water level instrumentation division inoperable and/or with one suppression pool water temperature instrumentation channel in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore the inoperable channel(s) to OPERABLE status within 7 days or verify suppression pool water level and/or temperature to be within the limits at least once per 12 hours.

d. With both suppression pool water level instrumentation divisions inoperable and/or with both suppression pool water temperature instrumentation channels in any pair(s) of temperature instrumentation channels in the same sector inoperable, restore at least one inoperable water level division and at least one inoperable water temperature instrumentation channel in each pair of temperature instrumentation channels in the same sector to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The suppression pool shall be demonstrated OPERABLE:

a. By verifying the suppression pool water volume to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression pool average water temperature to be less than or equal to 95°F, except:

   1. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature less than or equal to 105°F.

   2. At least once per hour when suppression pool average water temperature is greater than or equal to 95°F, by verifying suppression pool average water temperature to be less than or equal to 110°F and THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

   3. At least once per 30 minutes following a scram with suppression pool average water temperature greater than or equal to 95°F, by verifying suppression pool average water temperature less than or equal to 120°F.

SURVEILLANCE REQUIREMENTS (Continued

c. By verifying two suppression pool water level instrumentation divisions, with 1 channel per division, and at least twelve suppression pool water temperature instrumentation channels, at least two channels in each suppression pool sector shown below in Table 4.6.3.1-1, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the water level and temperature alarm setpoint for:

1. High water level $\leq$ 18'9",

2. Low water level $\geq$ 18'5-1/2", and

3. High water temperature $\leq$ 90°F.

SURVEILLANCE REQUIREMENTS (Continued)

3. By verifying the OPERABILITY of the vacuum breaker isolation valve differential pressure actuation instrumentation with the opening setpoint of -1.0 to 0.0 psid (Drywell minus Containment) by performance of a:

   a) CHANNEL CHECK at least once per 24 hours,

   b) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   c) CHANNEL CALIBRATION at least once per 18 months.

Note 1: Until restart after the first refueling outage, the following requirements shall apply:

3.6.5

c. With the position indicator of an OPERABLE drywell post-LOCA isolation valve for a vacuum breaker inoperable, verify the isolation valve to be closed at least once per 24 hours by local indication. Otherwise declare the isolation valve inoperable.

4.6.5.b.1

b. Verifying the position indicator for the vacuum breaker isolation valve OPERABLE by observing expected valve movement during the cycling test.

4.6.5.b.2

At least once per 18 months by:

a) Verifying the pressure differential required to open the vacuum breaker, from the closed position, to be less than or equal to 1.0 psid, and

b) Verifying the position indicator for the vacuum breaker isolation valve OPERABLE by performance of a CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   1. Verifying that the subsystem satisfies the in-place testing acceptance criteria and uses the test procedures of 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 4000 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

   3. Verifying a subsystem flow rate of 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

   1. Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence for the:

      a) LOCA, and

      b) Fuel handling accident.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.2 inches Water Gauge while operating the filter train at a flow rate of 4000 cfm ± 10%.

   3. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      a. Drywell pressure - high,

      b. Reactor vessel water level - low low, level 2,

      c. Fuel handling area ventilation exhaust radiation - high, and

      d. Fuel handling area pool sweep exhaust radiation - high.

      e. Manual initiation from the Control Room.

   4. Verifying that the fan can be manually started.

   5. Verifying that the heaters dissipate 50 ± 5.0 kW when tested in accordance with ANSI N510-1975.

CONTAINMENT SYSTEMS

DRYWELL PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.7.3 Two independent drywell purge system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell purge system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS Continued

4.6.7.3 Each drywell purge system subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days by:

   1. Starting the subsystem from the control room, and

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by:

   1. Verifying a subsystem flow rate of at least 1000 cfm during subsystem operation for at least 15 minutes.

   2. Verifying the pressure differential required to open the vacuum breakers on the drywell purge compressor discharge lines, from the closed position, to be less than or equal to 1.0 psid.

c. Verifying the OPERABILITY of the drywell purge compressor discharge line vacuum breaker isolation valve differential pressure actuation instrumentation with an opening setpoint of 0.0 to 1.0 psid (Drywell minus Containment) by performance of a:

   1. CHANNEL CHECK at least once per 24 hours,

   2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   3. CHANNEL CALIBRATION at least once per 18 months.

PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 At least the following independent SSW cooling tower basins, each with:

a. A minimum basin water level at or above elevation 130'3" Mean Sea Level, USGS datum, equivalent to an indicated level of $\geq$ 87".

b. Two OPERABLE cooling tower fans,#

shall be OPERABLE:

a. In OPERATIONAL Condition 1, 2 and 3, two basins,

b. In OPERATIONAL Condition 4, 5 and *, the basins associated with systems and components required OPERABLE by Specifications 3.7.1.1 and 3.7.1.2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, 5 and *.

ACTION:

a. In OPERATIONAL CONDITION 1, 2, 3, 4, 5 and * with one SSW cooling tower basin inoperable, declare the associated SSW subsystem inoperable and, if applicable, declare the HPCS service water system inoperable, and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2, as applicable.

b. In OPERATIONAL CONDITION 1, 2, 3, 4 or 5 with both SSW cooling tower basins inoperable, declare the SSW system and the HPCS service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.

c. In Operational Condition * with both SSW cooling tower basins inoperable, declare the SSW system inoperable and take the ACTION required by Specification 3.7.1.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 At least the above required SSW cooling tower basins shall be determined OPERABLE at least once per:

a. 24 hours by verifying basin water level to be greater than or equal to 87".

b. 31 days by starting each SSW cooling tower fan from the control room and operating the fan for at least 15 minutes.

c. 18 months by verifying that each SSW cooling tower fan starts automatically when the associated SSW subsystem is started.

*When handling irradiated fuel in the Auxiliary Building or Enclosure Building.

#The basin cooling tower fans are not required to be OPERABLE for HPCS service water system OPERABILITY.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the subsystem satisfies the in-place testing acceptance criteria and uses the test procedures of 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 4000 cfm ± 10%.

3. 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.

4. Verifying a subsystem flow rate of 4000 cfm ± 10% during subsystem operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Positon 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.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.2 inches Water Gauge while operating the subsystem at a flow rate of 4000 cfm ± 10%.

2. Verifying that on each of the below isolation mode actuation test signals, the subsystem automatically switches to the isolation mode of operation and the isolation valves close within 4 seconds:

   a) High radiation in the outside air intake duct,
   b) High chlorine concentration in the outside air intake duct,
   c) High drywell pressure, and
   d) Low reactor water level.
   e) Manual initiation from the Control Room.

3. Verifying that the heaters dissipate 20.7 ± 2.1 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 4000 cfm ± 10%.

TABLE 3.7.4-2

SAFETY RELATED MECHANICAL SNUBBERS*

a. RECIRCULATION SYSTEM

SNUBBER NO.	AREA	ELEVATION
Q1B33G023R01	11	117
Q1B33G023R01	11	117
Q1B33G024R01	11	102
Q1B33G024R02	11	102
Q1B33G024R02	11	102
Q1B33G024R05	11	101
Q1B33G105C01	11	101
Q1B33G105R01	11	101
Q1B33G105R02	11	101
Q1B33G105R02	11	101
Q1B33G108C01	11	101
Q1B33G108R01	11	101
Q1B33G108R01	11	101
Q1B33G108R01	11	101
Q1B33G108R02	11	101
Q1B33G108R02	11	101

RECIRCULATION SYSTEM (Continued)

SNUBBER NO.	AREA	ELEVATION
Q1B33G112R02	11	101
Q1B33G124R01	11	122
Q1B33G128C01	11	121
Q1B33G128C01	11	121
Q1B33G129C01	11	121
Q1B33G262R02	11	103
Q1B33G265C01	11	102
Q1B33G265R04	11	107
Q1B33G265R05	11	112
Q1B33G322R01	11	112
Q1B33G322R01	11	112
Q1B33G331R02	11	111
Q1B33G337R02	11	109
Q1B33G339R01	11	111
Q1B33G346R01	11	105
Q1B33G355R01	11	100
Q1B33G318R01	11	100

*Snubbers may be added to safety related systems without prior License Amendment to Table 3.7.4-2 provided that a revision to Table 3.7.4-2 is included with the next License Amendment request.

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 10 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.5.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.5.2 Test Frequencies - Each category of sealed sources, excluding startup sources and fission detectors previously subjected to core flux, shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days, excluding Hydrogen 3, and

   2. In any form other than gas.

3/4.7.10 EMBANKMENT STABILITY

LIMITING CONDITION FOR OPERATION

3.7.10 The downstream access road slope at Culvert No. 1 and the drainage basin slopes shall remain stable.

APPLICABILITY: At all times.

ACTION: If Culvert No. 1 has blockage exceeding 15% of its cross-sectional area, the Culvert shall be cleaned and the slope embankments verified to be stable.

SURVEILLANCE REQUIREMENTS

4.7.10 The downstream access road slope at Culvert No. 1 and the drainage basin slopes shall be confirmed to be stable by:

a. At least once per year, performing a visual inspection of the embankments and Culvert No. 1.

b. At least once per five years, performing a five-year survey to confirm no significant degradation to the base-line data.

c. Following the occurrence of earthquakes, hurricanes, tornados, or intense local rainfalls, a visual inspection of the embankments and Culvert No. 1 will be made. If this special inspection reveals evidence of change, a survey will be performed to confirm no significant degradation to the base-line data.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

   1. Separate day fue. tanks containing a minimum of 220 gallons of fuel.

   2. A separate fuel storage system containing a minimum of:

      a) 48,000 gallons of fuel each for diesel generators 11 and 12, and

      b) 39,000 gallons of fuel for diesel generator 13.

   3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With either one offsite circuit or diesel generator 11 or 12 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a within one hour and 4.8.1.1.2.a.4, for one diesel generator at a time, within three hours and at least once per 8 hours therafter; restore at least two offsite circuits and diesel generators 11 and 12 to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one offsite circuit and diesel generator 11 or 12 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a within one hour and 4.8.1.1.2.a.4, for one diesel generator at at time, within two hours and at least once per 8 hours thereafter; restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators 11 and 12 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 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 and indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each of the above required diesel generators shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8.1.1.2-1 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 441 rpm for diesel generators 11 and 12 and 882 rpm for diesel generator 13 in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual.

b) Simulated loss of offsite power by itself.

c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.

d) An ESF actuation test signal by itself.

5. Verifying the diesel generator is synchronized, loaded to greater than or equal to 3500 kW for diesel generators 11 and 12 and 1650 kW for diesel generator 13 in less than or equal to 60 seconds, and operates with these loads for at least 60 minutes.

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to:

a) 160 psig for diesel generator 11 and 12, and

b) 175 psig for diesel generator 13.

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 fuel tanks.

SURVEILLANCE REQUIREMENTS (Continued)

9. 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 7700 kW for diesel generators 11 and 12 and 3630 kW for diesel generator 13 and during the remaining 22 hours of this test, the diesel generator shall be loaded to 7000 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.d.7.a).2) and b).2)*.

10. Verifying that the auto-connected loads to each diesel generator do not exceed the continuous rating of 7000 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13.

11. 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.

12. Verifying that with the diesel generator operating in a test mode and connected to its bus that a simulated ECCS actuation signal:

    a) For Divisions 1 and 2, overrides the test mode by returning the diesel generator to standby operation.

    b) For Division 3, overrides the test mode by bypassing the diesel generator automatic trips per Surveillance Requirement 4.8.1.1.2.d.8.b).

13. Verifying that with all diesel generator air start receivers pressurized to less than or equal to 256 psig and the compressors secured, the diesel generator starts at least 5 times from ambient conditions and accelerates to at least 441 rpm for diesel generators 11 and 12 and 882 rpm for diesel generator 13 in less than or equal to 10 seconds.

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*If Surveillance Requirement 4.8.1.1.2.d.4.a)2) or b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at rated load for one hour or until operating temperatures have stabilized.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(b)	> 2.07 volts
Specific Gravity(a)	≥ 1.195	≥ 1.190	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells ≥ 1.190

(a) Corrected for electrolyte temperature and level.
(b) May be corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

TABLE 3.8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	RESPONSE TIME (Cycles)	SYSTEM/ COMPONENT AFFECTED
a. 6.9 kV Circuit Breakers			
252-1103-B	7200/45/± 10%#	60	Reactor Recir. Pump
252-1103-C	7200/45/± 10%#	60	Pump B33C001A
252-1205-B	7200/45/± 10%#	60	Reactor Recir. Pump
252-1205-C	7200/45/± 10%#	60	Pump B33C001B

b. 480 VAC Molded Case Circuit Breakers

1. Stored Energy Type SS3G3

BREAKER NUMBER	TRIP SETPOINT (Amperes)	RESPONSE TIME (Seconds)	SYSTEM/COMPONENT AFFECTED
52-12202	1200	0.05	CONTAINMENT COOLING FILTER TRAIN HEATERS (N1M41D002B-N)
52-12209	2000	0.05	CNTMT POLAR CRANE (Q1F13E001-N)
51-11502	1200	0.05	CNTMT CLG. FILTER TRAIN HEATER (N1M41D002A-N)
52-15105	2000	0.05	DRYWELL PURGE COMPRESS. (Q1E61C001A-A)
52-16204	2000	0.05	DRYWELL PURGE COMPRESS. (Q1E61C001B-B)
52-16404	1200	0.05	HYDROGEN RECOMBINER (Q1E61C003B-B)

#Primary current/setpoint.

TABLE 3.8.4.2-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E51F010-A	Continuous	RCIC System
Q1E51F013-A	Continuous	RCIC System
Q1E51F019-A	Continuous	RCIC System
Q1E51F022-A	Continuous	RCIC System
Q1E51F031-A	Continuous	RCIC System
Q1E51F045-A	Continuous	RCIC System
Q1E51F046-A	Continuous	RCIC System
Q1E51F059-A	Continuous	RCIC System
Q1E51F068-A	Continuous	RCIC System
Valve on Turbine Q1E51C002	Continuous	RCIC System
Q1B21F065A-A	No	Reactor Coolant System
Q1B21F065B-A	No	Reactor Coolant System
Q1B21F098A-B	No	Reactor Coolant System
Q1B21F098B-B	No	Reactor Coolant System
Q1B21F098C-B	No	Reactor Coolant System
Q1B21F098D-B	No	Reactor Coolant System
Q1B21F019	Continuous	Reactor Coolant System
Q1B21F067A	Continuous	Reactor Coolant System
Q1B21F067B	Continuous	Reactor Coolant System
Q1B21F067C	Continuous	Reactor Coolant System
Q1B21F067D	Continuous	Reactor Coolant System
Q1B21F016	Continuous	Reactor Coolant System
Q1B21F147A	Continuous	MSL Drain Post LOCA Leakage Control
Q1B21F147B	Continuous	MSL Drain Post LOCA Leakage Control
Q1B33F019	Continuous	Recirculation System
Q1B33F020	Continuous	Recirculation System
Q1B33F125	Continuous	Recirculation System
Q1B33F126	Continuous	Recirculation System
Q1B33F127	Continuous	Recirculation System
Q1B33F128	Continuous	Recirculation System
Q1D23F591B	*	Drywell Monitoring System
Q1D23F592A	*	Drywell Monitoring System
Q1D23F593B	*	Drywell Monitoring System
Q1D23F594A	*	Drywell Monitoring System
Q1E12F040	Continuous	RHR System
Q1E12F023	Continuous	RHR System
Q1E12F006A	Continuous	RHR System
Q1E12F052A	Continuous	RHR System
Q1E12F008	Continuous	RHR System

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E12F074A	Continuous	RHR System
Q1E12F026A	Continuous	RHR System
Q1E12F082A	No	RHR System
Q1E12F082B	No	RHR System
Q1E12F290A	Continuous	RHR System
Q1E12F047A	Continuous	RHR System
Q1E12F027A	Continuous	RHR System
Q1E12F073A	Continuous	RHR System
Q1E12F346	Continuous	RHR System
Q1E12F024A	Continuous	RHR System
Q1E12F087A	Continuous	RHR System
Q1E12F048A	Continuous	RHR System
Q1E12F042A	Continuous	RHR System
Q1E12F004A	Continuous	RHR System
Q1E12F003A	Continuous	RHR System
Q1E12F011A	Continuous	RHR System
Q1E12F053A	Continuous	RHR System
Q1E12F037A	Continuous	RHR System
Q1E12F028A	Continuous	RHR System
Q1E12F064A	Continuous	RHR System
Q1E12F290B	Continuous	RHR System
Q1E12F004C	Continuous	RHR System
Q1E12F021	Continuous	RHR System
Q1E12F064C	Continuous	RHR System
Q1E12F042C	Continuous	RHR System
Q1E12F048B	Continuous	RHR System
Q1E12F049	Continuous	RHR System
Q1E12F037B	Continuous	RHR System
Q1E12F053B	Continuous	RHR System
Q1E12F074B	Continuous	RHR System
Q1E12F042B	Continuous	RHR System
Q1E12F064B	Continuous	RHR System
Q1E12F096	Continuous	RHR System
Q1E12F094	Continuous	RHR System
Q1E12F006B	Continuous	RHR System
Q1E12F011B	Continuous	RHR System
Q1E12F052B	Continuous	RHR System
Q1E12F047B	Continuous	RHR System
Q1E12F027B	Continuous	RHR System
Q1E12F004B	Continuous	RHR System
Q1E12F087B	Continuous	RHR System
Q1E12F003B	Continuous	RHR System
Q1E12F026B	Continuous	RHR System
Q1E12F024B	Continuous	RHR System
Q1E12F028B	Continuous	RHR System
Q1E12F009	Continuous	RHR System
Q1E12F073B	Continuous	RHR System

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1C11F083	No	CRD Hydraulic System
Q1C11F322	Continuous	CRD Hydraulic System
Q1C41F001A	Continuous	Standby Liquid Control
Q1C41F001B	Continuous	Standby Liquid Control
Q1E21F001	Continuous	LPCS System
Q1E21F011	Continuous	LPCS System
Q1E21F012	Continuous	LPCS System
Q1E21F005	Continuous	LPCS System
Q1E30F002A	Continuous	Suppression Pool Makeup System
Q1E30F591A	*	Suppression Pool Makeup System
Q1E30F592A	*	Suppression Pool Makeup System
Q1E30F593A	*	Suppression Pool Makeup System
Q1E30F594A	*	Suppression Pool Makeup System
Q1E30F001A	Continuous	Suppression Pool Makeup System
Q1E30F001B	Continuous	Suppression Pool Makeup System
Q1E30F002B	Continuous	Suppression Pool Makeup System
Q1E30F591B	*	Suppression Pool Makeup System
Q1E30F592B	*	Suppression Pool Makeup System
Q1E30F593B	*	Suppression Pool Makeup System
Q1E30F594B	*	Suppression Pool Makeup System
Q1E31F100A	Continuous	Fuel Pool Cooling and Cleanup System
Q1E31F100B	Continuous	Fuel Pool Cooling and Cleanup System
Q1E32F001A	Continuous	MSIV - LCS
Q1E32F001E	Continuous	MSIV - LCS
Q1E32F003A	Continuous	MSIV - LCS
Q1E32F003E	Continuous	MSIV - LCS
Q1E32F003J	Continuous	MSIV - LCS
Q1E32F003N	Continuous	MSIV - LCS
Q1E32F001J	Continuous	MSIV - LCS
Q1E32F001N	Continuous	MSIV - LCS
Q1E32F002A	Continuous	MSIV - LCS
Q1E32F002E	Continuous	MSIV - LCS
Q1E32F002J	Continuous	MSIV - LCS
Q1E32F002N	Continuous	MSIV - LCS
Q1E32F006	Continuous	MSIV - LCS
Q1E32F007	Continuous	MSIV - LCS
Q1E32F008	Continuous	MSIV - LCS
Q1E32F009	Continuous	MSIV - LCS
Q1E38F001A	Continuous	Feedwater LCS
Q1E38F001B	Continuous	Feedwater LCS

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E51F064	Continuous	RCIC System
Q1E51F063	Continuous	RCIC System
Q1E51F076	Continuous	RCIC System
Q1E51F077	Continuous	RCIC System
Q1E51F078	Continuous	RCIC System
Q1E22F001	Continuous	HPCS System
Q1E22F004	Continuous	HPCS System
Q1E22F010	Continuous	HPCS System
Q1E22F011	Continuous	HPCS System
Q1E22F012	Continuous	HPCS System
Q1E22F015	Continuous	HPCS System
Q1E22F023	Continuous	HPCS System
Q1E61F595A	*	Combustible Gas Control System
Q1E61F596A	*	Combustible Gas Control System
Q1E61F597A	*	Combustible Gas Control System
Q1E61F598A	*	Combustible Gas Control System
Q1E61F595C	*	Combustible Gas Control System
Q1E61F596C	*	Combustible Gas Control System
Q1E61F597C	*	Combustible Gas Control System
Q1E61F598C	*	Combustible Gas Control System
Q1E61F595B	*	Combustible Gas Control System
Q1E61F596B	*	Combustible Gas Control System
Q1E61F597B	*	Combustible Gas Control System
Q1E61F598B	*	Combustible Gas Control System
Q1E61F595D	*	Combustible Gas Control System
Q1E61F596D	*	Combustible Gas Control System
Q1E61F597D	*	Combustible Gas Control System
Q1E61F598D	*	Combustible Gas Control System
Q1E61F003A	Continuous	Combustible Gas Control System
Q1E61F005A	Continuous	Combustible Gas Control System
Q1E61F003B	Continuous	Combustible Gas Control System
Q1E61F005B	Continuous	Combustible Gas Control System
Q1G33F251	Continuous	RWCU System
QIG33F253	Continuous	RWCU System
Q1G33F004	Continuous	RWCU System
Q1G33F039	Continuous	RWCU System
Q1G33F034	Continuous	RWCU System
Q1G33F054	Continuous	RWCU System
Q1G33F028	Continuous	RWCU System
Q1G33F053	Continuous	RWCU System
Q1G33F040	Continuous	RWCU System
Q1G33F001	Continuous	RWCU System
Q1G33F250	Continuous	RWCU System
Q1G33F252	Continuous	RWCU System

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1G41F028	Continuous	Spent Fuel Pool Cooling and Cleanup System
Q1G41F029	Continuous	Spent Fuel Pool Cooling and Cleanup System
Q1G41F044	Continuous	Spent Fuel Pool Cooling and Cleanup System
Q1G41F021	No	Spent Fuel Pool Cooling and Cleanup System
Q1G41F043	No	Spent Fuel Pool Cooling and Cleanup System
Q1M71F591A	*	Containment/Drywell I&C
Q1M71F593A	*	Containment/Drywell I&C
Q1M71F592B	*	Containment/Drywell I&C
Q1M71F595	*	Containment/Drywell I&C
Q1M71F591B	*	Containment/Drywell I&C
Q1M71F592A	*	Containment/Drywell I&C
Q1M71F594	*	Containment/Drywell I&C
Q1P21F017	Continuous	Makeup Water Treatment System
Q1P21F018	Continuous	Makeup Water Treatment System
Q1P41F237	Continuous	SSW System
Q1P41F018	Continuous	SSW System
Q1P41F241	Continuous	SSW System
Q1P41F238	Continuous	SSW System
QSP41F081A	Continuous	SSW System
QSP41F064A	Continuous	SSW System
Q1P41F068A	Continuous	SSW System
Q1P41F014A	Continuous	SSW System
Q1P41F159A	Continuous	SSW System
Q1P41F160A	Continuous	SSW System
Q1P41F113	Continuous	SSW System
Q1P41F168A	Continuous	SSW System
Q1P41F001A	Continuous	SSW System
Q1P41F016A	Continuous	SSW System
Q1P41F015A	Continuous	SSW System
Q1P41F006A	Continuous	SSW System
Q1P41F005A	Continuous	SSW System
Q1P41F007A	Continuous	SSW System
QSP41F074A	Continuous	SSW System
QSP41F066A	Continuous	SSW System
QSP41F125	Continuous	SSW System
Q1P41F018B	Continuous	SSW System
Q1P41F160B	Continuous	SSW System
Q1P41F159B	Continuous	SSW System
Q1P41F168B	Continuous	SSW System
QSP41F154	Accident Conditions	SSW System

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
QSP41F155A	Accident Conditions	SSW System
Q1P41F068B	Continuous	SSW System
QSP41F155B	Accident Conditions	SSW System
Q1P41F014B	Continuous	SSW System
QSP41F064B	Continuous	SSW System
QSP41F081B	Continuous	SSW System
Q1P41F006B	Continuous	SSW System
Q1P41F007B	Continuous	SSW System
Q1P41F001B	Continuous	SSW System
Q1P41F016B	Continuous	SSW System
Q1P41F005B	Continuous	SSW System
Q1P41F015B	Continuous	SSW System
QSP41F066B	Continuous	SSW System
QSP41F074B	Continuous	SSW System
QSP41F189	Continuous	SSW System
Q1P41F011	Continuous	SSW System
Q1P41F119A	No	SSW System
Q1P41F119B	No	SSW System
Q1P41F121A	No	SSW System
Q1P41F121B	No	SSW System
Q1P41F122A	No	SSW System
Q1P41F122B	No	SSW System
QSZ51F007	Continuous	Control Room HVAC
QSZ51F008	Continuous	Control Room HVAC
QSZ51F014	Continuous	Control Room HVAC
QSZ51F016	Continuous	Control Room HVAC
Q1P42F067	Continuous	CCW System
Q1P42F116	Continuous	CCW System
Q1P42F028A	Continuous	CCW System
Q1P42F032A	Continuous	CCW System
Q1P42F201A	Continuous	CCW System
Q1P42F204	Continuous	CCW System
Q1P42F205	Continuous	CCW System
Q1P42F105	Continuous	CCW System
Q1P42F200A	Continuous	CCW System
Q1P42F203	Continuous	CCW System
Q1P42F117	Continuous	CCW System
Q1P42F114	Continuous	CCW System
Q1P42F068	Continuous	CCW System
Q1P42F200B	Continuous	CCW System
Q1P42F028B	Continuous	CCW System
Q1P42F201B	Continuous	CCW System
Q1P42F032B	Continuous	CCW System
Q1P42F066	Continuous	CCW System

*Manual bypass of thermal overload protection of manually controlled valve.

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (MO)	SYSTEM(S) AFFECTED
Q1P44F053	Continuous	Plant SW System
Q1P44F069	Continuous	Plant SW System
Q1P44F076	Continuous	Plant SW System
Q1P44F070	Continuous	Plant SW System
Q1P44F074	Continuous	Plant SW System
Q1P44F077	Continuous	Plant SW System
Q1P44F042	Continuous	Plant SW System
Q1P44F054	Continuous	Plant SW System
Q1P44F067	Continuous	Plant SW System
Q1P45F096	Continuous	Floor & Eqmt. Drain System
Q1P45F097	Continuous	Floor & Eqmt. Drain System
Q1P52F195	Continuous	Service Air System
Q1P53F003	Continuous	Instrument Air System
Q1P53F007	Continuous	Instrument Air System
Q1T48F005	Continuous	SGTS
Q1T48F006	Continuous	SGTS
Q1T48F024	Continuous	SGTS
Q1T48F026	Continuous	SGTS
Q1T48F023	Continuous	SGTS
Q1T48F025	Continuous	SGTS
Q1P45F273	Continuous	Floor & Eqmt. Drain System
Q1P45F274	Continuous	Floor & Eqmt. Drain System

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site to unrestricted areas (see Figure 5.1.3-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11.1.1.1-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11.1.1.1-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.1.2.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from the radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions from gaseous effluents for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from the radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions from gaseous effluents for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM, AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ, and

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, or radioactive materials in paticulate form, with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to ensure that future releases will be in compliance with Specification 3.11.2.3.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations. Cumulative dose contributions from tritium, radioiodines, and radioactive materials in particulate form wth half-lives greater than 8 days for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the SRC. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Appendix A Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 12 months.

f. The Security Plan and implementing procedures at least once per 12 months.

g. Any other area of unit operation considered appropriate by the SRC or the Senior Vice President - Nuclear.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, February 1979, at least once per 12 months.

STAFF EVALUATION
AMENDMENT NO. 8 TO NPF-13
GRAND GULF NUCLEAR STATION, UNIT 1
DOCKET NO. 50-416

Introduction

Mississippi Power & Light Company, Middle South Energy Inc., and South Mississippi Electric Power Association (the licensees) are the holders of Facility Operating License No. NPF-13, which authorizes the operation of the Grand Gulf Nuclear Station, Unit 1, (the facility) at steady-state reactor power levels not in excess of 191 megawatts thermal. The facility consists of a boiling water reactor (BWR) located in Claiborne County, Mississippi.

It has been discovered that there were editorial and nomenclature errors in the Technical Specifications for the subject facility. The Technical Specifications did not in all cases agree with the actual as-built condition of the facility as actually described, analyzed in the Final Safety Analysis Report and approved in the NRC staff's Safety Evaluation Reports as supplemented. In addition, typographical errors were contained in the Technical Specifications. These matters were in part addressed in a confirmatory letter of October 20, 1982 from the NRC staff to the licensees.

Mississippi Power & Light Company responded by letters dated March 24, 1983, April 7, 1983, April 25, 1983, June 9, 1983, June 14, 1983, June 23, 1983, and June 29, 1983. In these submittals, MP&L has identified and committed to implement changes to the Technical Specifications. The need for these changes resulted from MP&L's review of the facility's surveillance test procedures.

Evaluation

The bulk of the changes to the Technical Specifications are administrative in nature and are necessary to correct editorial and nomenclature errors and to achieve consistency throughout the Technical Specifications and with the as-built condition of the plant. None of the changes involve a significant relaxation of the criteria used to establish safety limits or the bases for limiting safety system settings or limiting conditions for operation.

In the following tables, the changes are grouped together in common categories with cross-reference to the MP&L letters.

Table 1
Editorial or Nomenclature
Corrections to Technical Specifications

Letter Reference	Item	Technical Specification Section
3/24/83	6	Table 3.3.2-1
4/7/83	3	4.4.6.1.3
4/7/83	13	4.8.1.1.2.d.9
4/7/83	22	6.5.2.8
4/25/83	1	3.7.5
6/9/83	3	Table 3.3.2-1
6/9/83	4	Table 3.3.3-3
6/9/83	6	Table 3.3.7.1-1
6/9/83	7	Table 4.3.7.12-2
6/14/83	5	Table 3.3.7.1-1
6/14/83	10	4.6.7.3
6/14/83	15	3.11.1.1, 3.11.1.2, 3.11.2.2, 3.11.2.3

Table 1 (continued)

Letter Reference	Item	Technical Specification Section
6/29/83	2	3.7.10, 4.7.10

Table 2
Changes to Maintain Consistency
Within Technical Specifications

Letter Reference	Item	Technical Specification Section
3/24/83	28	Table 4.3.3.1-1
6/9/83	12	3.7.1.3, 4.7.1.3
6/14/83	1	3.2.2
6/23/83	17	3.8.1.1
6/29/83	1	Table 4.3.3.1-1
6/29/83	9	4.5.3.1, 4.6.3.1

The changes listed in Tables 1 and 2 above are purely administrative changes.

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The remaining changes to the Technical Specifications are necessary to properly account for as-built plant conditions and for clarification. The as-built conditions conform to the system described and analyzed in the Final Safety Analysis Report (FSAR). The staff reviewed and approved these as-built conditions in their Operating License review.

Table 3
Technical Specification Changes
to Conform to As-built Plant

Letter Reference	Item	Technical Specification Section/Discussion of Change Bases
3/24/83	14	Table 3.8.4.1-1/Revised to reflect locked rotor current rise from residual voltage, equivalent protection of equipment.
3/24/83	15	Table 4.8.2.1-1/Different Battery Type, new limits reflect manufacturer's specifications.
3/24/83	23	Table 3.8.4.2-1/Addition of more valves to surveillance table
6/9/83	1	Table 2.2.1-1/More conservative setpoints per NSSS specification
6/9/83	5	Table 4.3.3.1-1/Solid state digital systems allows only testing of overall delay, not individual inputs.
6/9/83	10	3.6.1.3, 4.6.1.3, 3.6.2.3, 4.6.2.3/Air lock door has inflatable seal rather than air flask
6/9/83	11	4.6.6.3.d.3/System allows for additional testing by manual initiation
6/9/83	13	4.7.2.d.2/System allows for additional testing by manual initiation
6/9/83	14	Table 3.7.4-2/Addition of more snubbers to surveillance table

Table 3 (continued)

Letter Reference	Item	Technical Specification Section/Discussion of Change Bases
6/14/83	2	Table 3.3.2-2/Revision of setpoints per NSSS specification, within bounds of previous analysis
6/14/83	3	Table 3.3.3-3/Revision of setpoints per NSSS specification, within bounds of previous analysis
6/14/83	8	4.6.1.4/Original values from purchase specification, revised values from functional test
6/14/83	9	4.6.6.3.d.2/Reflects a more conservative pressure drop for filter bank
6/14/83	12	4.8.1.1.1/Automatic transfer to another offsite source not incorporated or required
6/23/83	1	Table 3.3.3-2/Revised values more conservative than previous analysis
6/23/83	4	Tables 2.2.1-1, 3.3.4.2-2/Revised values more conservative setpoints than current values
6/29/83	4	Table 3.3.2-2/Reflects actual conditions rather than nominal conditions
6/29/83	5	Table 3.3.3-2/Revised timer delay to incorporate tolerance, still within bounds of analysis
6/29/83	10	4.1.3.1.4/deletes running of a test individually that cannot be run separately, test still retained as part of another test

Table 4
Technical Specification Changes for Clarification

Letter Reference	Item	Technical Specificaton Section
6/23/83	3	4.6.5.b.3, 4.6.7.3

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These changes to the Technical Specifications are administrative in nature and are being made as editorial or nomenclature corrections of errors, to assure consistency within the Technical Specifications themselves, and to make the Technical Specifications consistent with the as-built condition of the plant which was described and analyzed in the FSAR and approved by the staff in its operating license review. These changes are necessary to correct inadvertent errors in the Technical Specifications when the license was issued rather than to change any physical features of the plant.

In view of the foregoing, the NRC staff concludes that these changes to the Technical Specifications are both appropriate and necessary and should be incorporated into the Technical Specifications at this time.

Environmental Consideration

The Commission has determined that the issuance of this amendment will not result in any environmental impacts other than those evaluated in the Final Environmental Statement since the activity authorized by the amendment is encompassed by the overall action evaluated in the Final Environmental Statement dated September 1981.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) this amendment results as part of the review for the full power operating license (43 FR 32903), (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 8, 1983

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-416

MISSISSIPPI POWER AND LIGHT COMPANY

MIDDLE SOUTH ENERGY, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

NOTICE OF ISSUANCE OF AMENDMENT OF FACILITY

OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 8 to Facility Operating License No. NPF-13, issued to Mississippi Power and Light Company, Middle South Energy, Inc., and South Mississippi Electric Power Association (the licensees), for Grand Gulf Nuclear Station, Unit No. 1 (the facility) located in Claiborne County, Mississippi. This amendment grants changes to the Technical Specifications which are administrative in nature and are necessary to correct editorial and nomenclature errors and to achieve consistency with the as-built condition of the plant. None of the changes involve a significant relaxation of the criteria used to establish safety limits or the bases for limiting safety system settings or limiting conditions for operation.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment. The changes to the Technical Specifications approved in this amendment are to correct deficiencies and inadvertent errors in the Technical Specifications which were identified during the low power testing period at Grand

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Gulf Unit 1. These corrective measures result as part of the review for the full power operating license and are encompassed by the prior public notice of the overall action involving the proposed issuance of an operating license published in the FEDERAL REGISTER on July 28, 1978 (43 FR 32903).

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact other than those evaluated in the Final Environmental Statement since the activity authorized by this amendment is encompassed by the overall action evaluated in the Final Environmental Statement dated September 1981.

For further details with respect to this action, see (1) the applications for the amendment dated March 24, 1983, April 7, 1983, April 25, 1983, June 9, 1983, June 14, 1983, June 23, 1983, and June 29, 1983; (2) Amendment No. 8 to License NPF-13 dated August 8, 1983; (3) the Commission's evaluation dated August 8, 1983; (4) Final Safety Analysis Report (FSAR) and amendments thereto; (5) Final Environmental Statement dated September 1981; (6) the Commission's Safety Evaluation Report dated September 1981 (NUREG-0831) and supplements thereto; and (7) the Commission's Confirmation of Action letter dated October 20, 1982. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Hinds Jr. College, George M. McLendon Library, Raymond, Mississippi 39154. A copy of items (1), (2), (3) and (7) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing. Copies of items (5) and (6) may be purchased at current rates from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161, and through the NRC GPO sales program

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by writing to the U. S. Nuclear Regulatory Commission, Attention: Sales Manager, Washington, D. C. 20555. GPO deposit account holders may call 301-492-9530.

Dated at Bethesda, Maryland, this 8th day of August 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

R. Auluck, Acting Chief
Licensing Branch No. 2
Division of Licensing

*SEE PREVIOUS CONCURRENCES

OFFICE	DL:LB#2/PM	DL:LB#2/LA	OELD	DL:LB#2/ABC			
SURNAME	DHouston*:kw	EHylton	MWagner*	RAuluck			
DATE	7/15/83	8/ /83	7/25/83	8/8/83			

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Dated at Bethesda, Maryland, this day of July 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

OFFICE	DL:LB#2/PM	DL:LB#2/LA	DL:LB#2/BC	OELD			
SURNAME	DHouston:pt	EHylton	ASchwencer	MGWagner MGW			
DATE	7/15/83	7/ /83	7/ /83	7/25/83			