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MAY 05 1982

Docket No. 50-364

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Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated October 28 and November 16, 1981 and March 18, 1982.

The amendment contains changes necessary to assure identical requirements and editorial improvements as were recently reviewed and approved for Unit 1 in Amendment No. 26 dated March 1, 1982. The plant design and operation are the same for Unit 1 and Unit 2. This amendment minimizes the potential for confusion and Technical Specification violations and allows a consistent set of operating, maintenance and surveillance procedures for the essentially identical units.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED

Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 13 to NPF-8
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

8205170145

*Previous concurrence see next page

no legal objection to amdt & notice

OFFICE	ORB#1:DL*	ORB#1:DL*	ORB#1:DL	AD/OR:DL	OELD		
SURNAME	CParrish	EReeves:ds	SVarga	TNovak			
DATE	04/ /82	04/29/82	04/29/82	04/29/82	05/ 3/82		

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Docket No. 50-364

Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated October 28 and November 16, 1981 and March 18, 1982.

The amendment is performa in nature resulting in identical requirements and improvements as recently reviewed and approved for Unit No. 1 in License Amendment No. 26.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. to NPF-8
2. Safety Evaluation
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cc w/encls:
See next page

4/20/82

OFFICE	DL:ORB#1	DL:ORB#1	DL:ORB#1	DL:OR	OELD		
SURNAME	CParrish	EAReeves:ms	SAVarga	TMNovak			
DATE	4/16/82	4/19/82	4/ /82	4/ /82	4/ /82		

Mr. F. L. Clayton
Alabama Power Company

cc: Mr. W. O. Whitt
Executive Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Ruble A. Thomas, Vice President
Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202

George F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, D. C. 20036

Chairman
Houston County Commission
Dothan, Alabama 36301

Robert A. Buettner, Esquire
Balch, Bingham, Baker, Hawthorne,
Williams and Ward
Post Office Box 306
Birmingham, Alabama 35201

George S. Houston Memorial Library
212 W. Burdeshaw Street
Dothan, Alabama 36303

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 24-Route 2
Columbia, Alabama 36319

State Department of Public Health
ATTN: State Health Officer
State Office Building
Montgomery, Alabama 36104

Regional Radiation Representatives
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

D. Biard MacGuineas, Esquire
Volpe, Boskey and Lyons
918 16th Street, N.W.
Washington, D.C. 20006

Charles R. Lowman
Alabama Electric Corporation
P.O. Box 550
Andalusia, Alabama 36420

Mr. R. P. McDonald
Vice President - Nuclear Generation
Alabama Power Company
P.O. Box 2641
Birmingham, Alabama 35291

James P. O'Reilly
Regional Administrator - Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
License No. NPF-8

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

   A. The applications for amendment by Alabama Power Company (the licensee) dated October 28 and November 16, 1981 and March 18, 1982, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and 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 rules and regulations of the Commission;

   C. There is reasonable assurance (1) 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.

8205170148

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 13 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 5, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-9	1-9
2-2	2-2
3/4 0-2	3/4 0-2
3/4 2-4	3/4 2-4
3/4 3-2	3/4 3-2
3/4 3-3	3/4 3-3
3/4 3-6	3/4 3-6
3/4 3-12	3/4 3-12
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-36	3/4 3-36
3/4 3-37	3/4 3-37
3/4 3-39	3/4 3-39
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-57	3/4 3-57
3/4 3-58	3/4 3-58
3/4 3-59	3/4 3-59
3/4 4-4a	3/4 4-4a
3/4 4-8	3/4 4-8
3/4 4-17	3/4 4-17
3/4 7-9	3/4 7-9
3/4 7-13	3/4 7-13
3/4 7-21	3/4 7-21
3/4 7-22	3/4 7-22
3/4 7-23	3/4 7-23
3/4 8-1	3/4 8-1
3/4 8-2	3/4 8-2
3/4 8-3	3/4 8-3
3/4 8-4	3/4 8-4
3/4 8-5	3/4 8-5
3/4 8-6	3/4 8-6
3/4 8-7	3/4 8-7
3/4 8-8	3/4 8-8
3/4 8-12	3/4 8-12
3/4 8-15	3/4 8-15
3/4 8-16	3/4 8-16
3/4 9-16	3/4 9-16
3/4 10-5	3/4 10-5
3/4 11-2	3/4 11-2
3/4 11-5	3/4 11-5
3/4 11-12	3/4 11-12
3/4 11-13	3/4 11-13

Remove Pages	Insert Pages
B 3/4 2-1	B 3/4 2-1
B 3/4 4-2	B 3/4 4-2
B 3/4 7-5	B 3/4 7-5
6-15	6-15
6-24	6-24
7-1	--

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This specification is not applicable in MODES 5 or 6.

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

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

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

b. The combined time interval for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

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

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

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

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

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

*Upon determination that the surveillance had been inadvertantly omitted, the Surveillance Requirement shall be successfully performed within the Limiting Condition of Operation (LCO) period which would begin upon discovery.

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

LIMITING CONDITION FOR OPERATION

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

$$F_Q(Z) \leq \left[\frac{2.31}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [(4.62)] [K(Z)] \text{ for } P \leq 0.5$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

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

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similiarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower delta T Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2, and *	12
2. Power Range, Neutron Flux					
A. High	4	2	3	1, 2	2#
B. Low	4	2	3	2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2, and *	3####
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and *	4
B. Shutdown	2	0	1###	3, 4 and 5	5
7. Overtemperature ΔT					
Three Loop Operation	3	2	2	1, 2	7#
Two Loop Operation	3	1**	2	1, 2	9
8. Overpower ΔT					
Three Loop Operation	3	2	2	1, 2	7#
Two Loop Operation	3	1**	2	1, 2	9
9. Pressurizer Pressure-Low	3	2	2	1	7#
10. Pressurizer Pressure--High	3	2	2	1, 2	7#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	7#
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
13. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	7#
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch in same loop or 2/loop-level and 1/loop-flow mismatch in same loop	1, 2	7#
15. Undervoltage-Reactor Coolant Pumps	3-2/bus	2	2	1	7#
16. Underfrequency-Reactor Coolant Pumps	3-2/bus	2	2	1	7#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal, and fuel in the reactor vessel.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above P-6.

###Indication only.

####The provisions of Specification 3.0.3 are not applicable if THERMAL POWER level ≥ 10% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

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

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

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

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

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors, is verified consistent with the quadrant power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Range, Neutron Flux				
A. High	S	D(2), M(3) and Q(6)	M	1, 2
B. Low	S	D(2), M(3) and Q(6)	S/U(10)	2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S(7)	R(6)	M and S/U(1)	2, 3, 4, 5, and *
7. Overtemperature ΔT	S	R	M	1, 2
8. Overpower ΔT	S	R	M	1, 2
9. Pressurizer Pressure--Low	S	R	M	1
10. Pressurizer Pressure--High	S	R	M	1, 2
11. Pressurizer Water Level--High	S	R	M	1
12. A. Loss of Flow - Single Loop	S	R	M	1
B. Loss of Flow - Two Loops	S	R	N.A.	1

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual	1/steam line	1/steam line	1/operating steam line	1, 2**, 3**	22
b. Automatic Actuation Logic	2	1	2	1, 2**, 3**	21
c. Containment Pressure--High-High	3	2	3	1, 2**, 3**	14*
d. Steam Flow in Two Steam Lines--High				1, 2**, 3**	
Three Loops Operating	2/steam line	1/steam line any 2 steam lines	1/steam line		14*
Two Loops Operating	2/operating steam line	1### /any operating steam line	1/operating steam line		15
COINCIDENT WITH T_{avg}--Low-Low				1, 2**, 3**	
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		14*
Two loops Operating	1 T_{avg}/operating loop	1### T_{avg} in any operating loop	1 T_{avg} in any operating loop		15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
				1, 2**, 3##,**	
e. Steam Line Pressure-Low					
Three Loops Operating	1 pressure/loop	1 pressure any 2 loops	1 pressure any 2 loops		14*
Two Loops Operating	1 pressure/operating loop	1### pressure in any operating loop	1 pressure any operating loop		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	14*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below the P-11. (Pressurizer Pressure Block of Safety Injection) setpoint.

##Trip function may be bypassed in this MODE below P-12. (T_{avg} Block of Safety Injection) setpoint.

###The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped mode.

*The provisions of Specification 3.0.4 are not applicable.

**Not applicable if main steam isolation valves are closed.

ACTION STATEMENTS

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 14 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 15 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

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

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

TABLE 3.3-3 (Continued)

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

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

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

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

ACTION 20 - With the interlock inoperable to the extent that a safeguards function which should not be blocked in the current MODE is blocked, declare the safeguard function(s) inoperable and follow the appropriate ACTION statement(s) of Table 3.3-3 for the affected function(s).

Interlock	Affected Channels on Table 3.3-3
1. P-4	a. Pressurizer Pressure - Low
2. P-12	a. Steam Line Pressure - Low b. Steam Flow in Two Steam Lines High Coincident With T_{avg}-Low-Low

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

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

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. AUXILIARY FEEDWATER				
a. Automatic Actuation Logic	N.A.	N.A.	M(2)(6)	1, 2, 3
b. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
c. Undervoltage - RCP	N.A.	R	M	1
d. S.I.	See 1 above (all SI surveillance requirements)			
e. Trip of Main Feedwater Pumps	N.A.	N.A.	S/U(5)	1
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R(3)	M(4)	1, 2, 3, 4
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R(3)	M(4)	1, 2, 3, 4
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS	N.A.	N.A.	R	N.A.

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

(2) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(3) Channel calibration shall exclude actuation of the final trip actuation relay.*

(4) Functional testing shall consist of verification of relay operation upon removal of input voltage and operation of 2-out-of-3 logic excluding the final trip actuation relay.*

(5) If not performed in the previous 92 days.

(6) Excluding automatic actuation logic for trip of main feedwater pumps.

*Actuation of the final trip actuation relay shall be included in response time testing.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area (R-5)	1	(a)	$\leq$ 15 mR/hr	10^{-1}-10^{4} mR/hr	23
b. Containment Area (R-27A&B)	2	1,2,3,4	N/A	1 - 10^{7} R/hr	27a
2. PROCESS MONITORS					
a. Fuel Storage Pool Area Gaseous Activity- Ventilation System Isolation (R-25A&B)	1	(b)	$\leq 8.73 \times 10^{-3}$ µCi/cc(c)	10-10^{6} cpm	25
b. Containment					
i. Gaseous Activity-					
a) Purge & Exhaust Isolation (R-24A&B)	1	1,2,3 (d)	$\leq 2.27 \times 10^{-2}$ µCi/cc(c)	10-10^{6} cpm	26
		4,5,6 (d)	$\leq 2.27 \times 10^{-2}$ µCi/cc(c)		26
		1,2,3,4,5,6 (e)	$\leq 4.54 \times 10^{-3}$ µCi/cc(c)		26
		1,2,3,4,5,6 (f)	$\leq 2.27 \times 10^{-3}$ µCi/cc(c)		26
b) RCS Leakage Detection (R-12)	1	1,2,3 & 4	N/A	10-10^{6} cpm	24
ii. Particulate Activity RCS Leakage Detection (R-11)	1	1,2,3 & 4	N/A	10-10^{6} cpm	24
c. Control Room Isolation (R35A&B)	1	1,2,3,4 and during movement of irradiated fuel or movement of loads over irradiated fuel	$\leq$ 800 cpm	10-10^{6} cpm	27

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.7.1.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12 and 3.9.13.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 27 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 27a - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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a. With fuel in storage pool.
b. With irradiated fuel in the storage pool.
c. Above background with no flow.
d. With mini-purge in operation.
e. With slow speed main purge in operation.
f. With fast speed main purge in operation.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area (R-5)	S	R	M	*
b. Containment Area (R-27A&B)	S	R	N/A	1,2,3,4
2. PROCESS MONITORS				
a. Fuel Storage Pool Area				
Gaseous Activity - Ventilation System Isolation (R-25 A&B)	S	R	M	**
b. Containment				
i. Gaseous Activity				
a) Purge & Exhaust Isolation (R-24 A&B)	S	R	M	All MODES
b) RCS Leakage Detection (R-12)	S	R	M	1,2,3 & 4
ii. Particulate Activity				
RCS Leakage Detection (R-11)	S	R	M	1,2,3 & 4
c. Control Room Isolation	S	R	M	1,2,3,4 and during movement of irradiated fuel or movement of loads over irradiated fuel

*With fuel in the storage pool or building.

**With irradiated fuel in the storage pool.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant Outlet Temperature-T_{Hot}-Wide Range	2	1
2. Reactor Coolant Inlet Temperature-T_{Cold}-Wide Range	2	1
3. Reactor Coolant Pressure-Wide Range	2	1
4. Steam Generator Water Level-Wide Range or Narrow Range	2/steam generator	1/steam generator
5. Refueling Water Storage Tank Water Level	2	1
6. Containment Pressure	2	1
7. Pressurizer Water Level	2	1
8. Steam Line Pressure	2/steam generator	1/steam generator
9. Auxiliary Feedwater Flow Rate	2	1
10. Reactor Coolant System Subcooling Margin Monitor	2	1
*11. PORV Position Indicator	1/valve	1/valve
**12. PORV Block Valve Position Indicator	1/valve	1/valve
13. Safety Valve Position Indication (One channel is position indicator and one channel is discharge temperature)	2/valve	1/valve
14. Containment Water Level - Narrow Range	2	1
15. Containment Water Level - Wide Range	2	1
16. Incore Thermocouples	4/core quadrant	2/core quadrant

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

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant Outlet Temperature-T_{Hot}-Wide Range	M	R
2. Reactor Coolant Temperature-T_{Cold}-Wide Range	M	R
3. Reactor Coolant Pressure-Wide Range	M	R
4. Steam Generator Water Level- Wide Range or Narrow Range	M	R
5. Refueling Water Storage Tank Water Level	M	R
6. Containment Pressure	M	R
7. Pressurizer Water Level	M	R
8. Steam Line Pressure	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
*11. PORV Position Indicator	M	R
**12. PORV Block Valve Position Indicator	M	R
13. Safety Valve Position Indication	M	R
14. Containment Water Level - Narrow Range	M	R
15. Containment Water Level - Wide Range	M	R
16. Incore Thermocouples	M	R

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

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-12 shall be OPERABLE.

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

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-12:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days, or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a function test which includes subjecting the detector to test aerosol. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of this functional test during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

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

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5.##

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

*The normal or emergency power source may be inoperable in MODE 5.

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

#Three filled Reactor Coolant loops and at least two steam generators having levels greater than or equal to 10% of wide range indication may be substituted for one RHR loop.

##A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 310°F unless (1) the pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication) or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 All power relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION and operating the valve through one cycle of full travel.

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

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 31 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 1 GPM leakage from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater that the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by;

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment air cooler condensate level system or containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 6 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided the isolation valve is restored to OPERABLE status or closed within 4 hours after entering MODE 2; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

3/4.7.5 RIVER WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 At least two independent river water loops shall be OPERABLE with at least two river water pumps per loop.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one river water loop OPERABLE, restore at least two loops to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 Each river water loop shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by:

1. Verifying that each automatic valve servicing safety related equipment actuates to its correct position on a low pond level signal.

2. Verifying that the buried piping is leak tight by a visual inspection of the ground area.

SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.9.d or 4.7.9.e, as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension and compression mode directions. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample of 88 snubbers shall be functionally tested either in place or in a bench test. If more than 3 snubbers do not meet the functional test acceptance criteria of Specification 4.7.9.d or 4.7.9.e, an additional sample selected according to the expression 22(a-3) shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the initial representative sample.

Functional testing shall continue according to the expression (22)b where b is the number of snubbers found inoperable in the previous re-sample, until no additional inoperable snubbers are found within a sample or until all snubbers in Table 3.7-4a and 3.7-4b have been functionally tested.

Snubbers greater than 50,000 lb. capacity may not be excluded from functional testing requirements.*

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the initial representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

*This portion of the specification is not effective until the second refueling outage or when a commercial in-place testing device is available, whichever is later.

SURVEILLANCE REQUIREMENTS (Continued)

2. Snubbers within five feet of heavy equipment (valve, pump, turbine, motor, etc.)

3. Snubbers within ten feet of the discharge from a safety relief valve

Snubbers identified in Tables 3.7-4a and 3.7-4b as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.* Tables 3.7-4a and 3.7-4b may be used jointly or separately as the basis for the sampling plan.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the attached component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria*

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

*This portion of the specification is effective prior to startup following the second refueling outage.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits from the offsite transmission network to the switchyard and two physically independent circuits from the switchyard to the onsite Class 1E distribution system, and

b. Two separate and independent diesel generator sets (Set A: DG 1-2A and DG-1C, Set B: DG-2B and DG-2C) each with:

   1. Separate day tanks containing a minimum volume of 900 gallons of fuel for the 4075 kw diesel generators and 700 gallons of fuel for the 2850 kw diesel generators.

   2. A separate fuel transfer pump for each diesel.

c. A fuel storage system consisting of four, independent storage tanks each containing a minimum of 25,000 gallons of fuel.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With an offsite circuit inoperable, demonstrate the OPERABILITY of the remaining offsite A.C. source by performing Surveillance Requirement 4.8.1.1.1.a within 8 hours and at least once per 24 hours thereafter, and performing Surveillance Requirement 4.8.1.1.2.a, items 1, 2, 3, 4, and 6 on diesel generators 1-2A and 2B within 8 hours unless such surveillance has been performed within the previous 7 days. Restore at least two offsite circuits to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

b. With one diesel generator set inoperable for reasons other than the yearly scheduled maintenance,** demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 8 hours and at least once per 72 hours thereafter, and performing Surveillance Requirement 4.8.1.1.2.a, items 1, 2, 3, 4, and

*One operable fuel storage tank must be available for each required diesel generator.

**If this scheduled maintenance exceeds 10 days, the diesel generator set must be declared inoperable.

ACTION (Continued)

6 on two* diesel generators within 12 hours. Restore the diesel generator set to OPERABLE status within 18 days or be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable if only one of the four diesel generator units is inoperable.

c. With one offsite circuit and one diesel generator set of the above required A.C. electrical power sources inoperable for reasons other than the yearly scheduled maintenance,** demonstrate the OPERABILITY of the remaining offsite A.C. source by performing Surveillance Requirement 4.8.1.1.1.a within 2 hours and performing Surveillance Requirement 4.8.1.1.2.a, items 1, 2, 3, 4, and 6 on two* diesel generators within 12 hours. Restore at least one of the inoperable sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other AC power source (offsite circuit or diesel generator set) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statements a or b, as appropriate.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of both diesel generator sets by performing Surveillance Requirement 4.8.1.1.2.a within 2 hours; unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore both offsite circuits to OPERABLE status within 7 days from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With both of the above required diesel generator sets inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 2 hours and performing Surveillance Requirement 4.8.1.1.2.a, items 1, 2, 3, 4, and 6 on one diesel generator in a diesel set on the other Unit within 8 hours; restore at least one of the inoperable diesel generator sets to OPERABLE status:

*The two diesel generators chosen to be tested shall verify that at least one train of LOCA/shutdown loads is capable of being powered at each Unit.

**If this scheduled maintenance exceeds 10 days, the diesel generator set must be declared inoperable.

ACTION (Continued)

1. Within 24 hours or be in at least HOT STANDBY within the next 6 hours if (DG 1-2A and DG-2C) or(DG-2B and DG-1C) or (DG-1C and DG-2C) are inoperable; or

2. Within 8 hours or be in at least HOT STANDBY within the next 6 hours if DG 1-2A and DG-2B are inoperable; or

3. Within 2 hours or be in at least HOT STANDBY within the next 6 hours if three or more diesel generators are inoperable.

Restore both diesel generator sets to OPERABLE status within 18 days from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

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

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank.

2. Verifying the fuel level in the fuel storage tanks.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

4. Verifying the diesel starts and accelerates to at least 900 rpm for the 2850 kw generator and 514 rpm for the 4075 kw generators in less than or equal to 12 seconds. The generator voltage and frequency shall be $\geq$ 3952 volts and $\geq$ 57 Hz within 12 seconds after the start signal and operates for 5 minutes.

5. Verifying the generator is synchronized, loaded to 2700-2850 kw for the 2850 kw generator and 3875-4075 kw for the 4075 kw generator and operates for greater than or equal to 60 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

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

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D270-65 is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment,

c. At least once per 18 months by:

1. Subjecting the diesel to an inspection and maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations,

2. Simulating a loss of offsite power by itself, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12* seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of all loads, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

3. Verifying that on a Safety Injection test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be ≥ 3952 volts and ≥ 57 Hz within 12 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained between 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

4. Simulating a loss of offsite power in conjunction with a Safety Injection test signal, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12* seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and

* Energization of the Unit 2 emergency bus for diesel generator 2C is achieved within 26 seconds.

SURVEILLANCE REQUIREMENTS (Continued)

operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential and low lube oil pressure, are automatically bypassed upon loss of voltage on the emergency bus and/or a safety injection test signal.

5. Verify that the diesel generators operate for 24 hours while loaded to 4353 kw for the 4075 kw diesels and 3100 for the 2850 kw diesels (2000 hour rating). After completing this 24-hour test, manually trip the diesel generator from the 2000-hour load and demonstrate hot restart capability by performing Surveillance Requirement 4.8.1.1.2.a.4 within 10 minutes.

6. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 4353 kw for the 4075 kw generator and 3100 kw for the 2850 kw generator.

7. 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.

8. Verifying that with the diesel generators operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by returning the diesel generator to standby operation.

9. Verifying that the automatic load sequence timer is OPERABLE with each load sequence time within ± 10% of its required value or 0.5 seconds whichever is greater.

10. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a) Oil Temperature High (OTH)

SURVEILLANCE REQUIREMENTS (Continued)

b) Coolant Temperature High (CTH)

c) Coolant Pressure Low (CPL)

d) Crankcase Pressure High (CCPH)

11. Verifying the capability to reject a load of greater than or equal to the largest single load associated with that diesel generator (approximately 1000 kw); while maintaining voltage between 3740 and 4580 volts and speed less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint.

d. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, and verifying that the diesel generators accelerate to at least 900 rpm, for the 2850 kw generator and 514 rpm for the 4075 kw generator, in less than or equal to 12 seconds.

e. At least once per 5 years, on a staggered basis, by verifying that the diesel generator can reject a load of 1200-2400 kw without tripping. The diesel generator output breaker(s) must remain closed such that the diesel generator is connected to at least one emergency bus. Verify that all fuses and breakers on the energized emergency bus(es) are not tripped. The generator voltage shall remain within 3330 and 4990 volts during and following the load rejection.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per diesel basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE (PER DIESEL)

Number of Failures In Last 100 Valid Tests*	Test Frequency
≤2	At least once per 14 days
≥3	At least once per 7 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per diesel basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in the computation of the "last 100 valid tests."

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One circuit from the offsite transmission network to the switchyard and from the switchyard to the onsite Class 1E distribution system, and

b. Diesel generator 1-2A, 1C or 2B each with:

1. A day tank containing a minimum volume of 900 gallons of fuel for the 4075 kw diesel generator and 700 gallons of fuel for the 2850 kw diesel generator.

2. A fuel storage tank containing a minimum volume of 25,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.5.

SURVEILLANCE REQUIREMENTS (Continued)

2. The voltage of each connected cell is greater than or equal to 2.02 volts under float charge and has not decreased more than 0.1 volts from the value observed during the original acceptance test.

3. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is greater than or equal to 1.190 and has not decreased more than 0.08 from the value observed during the previous test, and

4. The total battery terminal voltage is greater than or equal to 121.2 volts.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material, and

3. The battery charger will supply at least 536 amperes at $\geq$ 125 volts for at least 4 hours.

d. At least once per 18 months during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 2 hours when the battery is subjected to a battery service test or the individual cell voltage does not decrease below 1.75 volts when the battery is subjected to the following equivalent load profile.

Order In Which Loads Are Applied	Current (amps)	Duration (min.)
1	920	1
2	430	58
3	920	1
4	430	59
5	920	1

SERVICE WATER BUILDING D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The following D.C. distribution systems shall be energized and OPERABLE:

TRAIN "A" consisting of 125-volt D.C. Distribution Cabinet 1M, 125-volt battery bank No. 1 and a full capacity charger.

TRAIN "B" consisting of 125-volt D.C. Distribution Cabinet 1N, 125-volt battery bank No. 2, and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one 125-volt D.C. distribution train inoperable, restore the inoperable distribution system to OPERABLE and energized status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Each D.C. train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.5.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of the pilot cell is between the minimum and maximum level indication marks.

2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is greater than or equal to 1.190,

3. The pilot cell voltage is greater than or equal to 2.02 volts, and

4. The total battery terminal voltage is greater than or equal to 121.2 volts.

b. At least once per 92 days by verifying that:

1. The electrolyte level of each cell is between the minimum and maximum level indication marks,

SURVEILLANCE REQUIREMENTS (Continued)

2. The voltage of each connected cell is greater than or equal to 2.02 volts under float charge and has not decreased more than 0.1 volts from the value observed during the original acceptance test, and

3. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is greater than or equal to 1.190 and has not decreased more than 0.08 from the value observed during the previous test.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material, and

3. The battery charger will supply at least 3 amperes at ≥ 125 volts for at least 4 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 2 hours when the battery is subjected to a battery service test or the individual cell voltage does not decrease below 1.75 volts when the battery is subjected to the following equivalent load profile:

Order in Which Loads are Applied	Current (amps)	Duration
1	25	0 - 0.1 sec
2	1	0.1 sec - 2 hours

3/4.9.14 CONTAINMENT PURGE EXHAUST FILTER

LIMITING CONDITION FOR OPERATION

3.9.14 The containment purge exhaust filter shall be OPERABLE and valve N1P13V293 closed.

APPLICABILITY: During CORE ALTERATIONS and Fuel Movement inside containment with any containment purge isolation valve open.

ACTION: With the containment purge exhaust filter inoperable either:

1. Immediately close the 48 inch containment purge isolation valves (CBV-HV-3196, 3197, 3198A and 3198D) and the 18 inch containment mini-purge isolation valves (CBV-HV-2866A, 2866B, 2867A and 2867B), or

2. Cease all CORE ALTERATIONS and fuel movement.

SURVEILLANCE REQUIREMENTS

4.9.14 The above required containment purge exhaust filter shall be demonstrated OPERABLE:

a. 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 that could have contaminated the charcoal adsorbers or HEPA filter in any ventilation zone communicating with the system by:

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Minimum Detectable Concentration (MDC) (μCi/ml)[a, g]
A. Batch Waste Release Tanks[c]	P Each Batch	P Each Batch	Principal Gamma Emitters[e]	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	$1x10^{-5}$
	P Each Batch	M Composite[b]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases[d, f]	D Grab Sample	W Composite[b]	Principal Gamma Emitters[e]	$5x10^{-7}$
1. Steam Generator Blowdown			I-131	$1x10^{-6}$
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	D Grab Sample	M Composite[b]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	D Grab Sample	Q Composite[b]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
2. Turbine Building Sump[f]	P Grab Sample	W Composite[b]	Principle Gamma Emitters[e] H-3	$5x10^{-7}$ $1x10^{-5}$

DOSE

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 5.1-4) 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 reduce the releases of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year, so that the cumulative dose or dose commitment to an individual from these releases is within 3 mrem to the total body and 10 mrem to any organ.

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.

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) 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 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 reduce the releases of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year, so that the cumulative dose is within 10 mrad for gamma radiation and 20 mrad for beta radiation.

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 for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from radioiodines and radioactive materials in particulate form, and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1-3) 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 radioiodines, radioactive materials in particulate form, or radionuclides (other than noble gases) 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 reduce the releases of radioiodines and radioactive materials in particulate form, and radionuclides (other than nobles gases) with half-lives greater than 8 days in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year, so that the cumulative dose or dose commitment to an individual from these releases is within 15 mrem to any organ.

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 for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods and measurement uncertainty.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.31 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant Pressure and establish natural circulation.

3/4.4.5 RELIEF VALVES (PORV's)

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORV's minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the attached component.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample according to the expression $35(1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits, where c is the allowable number of snubbers not meeting the acceptance criteria. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in certain plant areas where personnel may be present under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analyses equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and the control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, including monitoring the condenser hotwells for evidence of condenser in-leakage.

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for off-control-point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the training of personnel, the procedures for sampling and analysis and the provisions for maintenance of sampling and analysis equipment.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physics Supervisor.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved Radiation Work Permit which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are following plant radiation protection procedures for entry into high radiation areas.

**Measurement made at 18" from source of radioactivity.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-364

Introduction

By letters dated October 28 and November 16, 1981 and March 18, 1982, Alabama Power Company (APCo) proposed changes to the Technical Specifications. These changes are based on the recent changes reviewed and approved for Farley Unit No. 1 by License Amendment No. 26 dated March 1, 1982. The most significant changes proposed were changes to F_Q to allow plugging of steam generators first row tubes and the improvements in diesel generator operation and surveillance. These changes were proposed previously by letters dated October 28 and November 16, 1981 and were included in Amendment No. 26 for Unit No. 1. Our evaluation is as follows.

Discussion and Evaluation

We have reviewed APCo's proposed changes to the Unit No. 2 Technical Specifications. We find the changes to be identical to those previously evaluated in our Safety Evaluation, Enclosure 2 to Joseph M. Farley Nuclear Plant, Unit No. 1, Licensing Amendment No. 26, dated March 1, 1982. We incorporate that SER by reference.

These changes for Unit No. 2 Technical Specifications are identical and consistent with those which we approved for Unit No. 1. Complete consistency and uniformity between the Unit No. 1 and Unit No. 2 Technical Specifications are now completed except for certain Unit No. 2 requirements (Near Term Operating License items) which we did not require for Unit No. 1.

This action will minimize the potential for confusion and Technical Specification violations and allow the consistent basis for operating, maintenance, and surveillance for both units.

Thus, we conclude that the proposed Technical Specification changes are acceptable.

8205170150

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (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.

Date: May 13, 1982

The principal contributor to this evaluation is E. A. Reeves, Project Manager.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-364

ALABAMA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. NPF-8 issued to Alabama Power Company (the licensee), which revised Technical Specifications for operation of the Joseph M. Farley Nuclear Plant, Unit No. 2 (the facility) located in Houston County, Alabama. The amendment is effective as of the date of issuance.

The amendment contains changes necessary to assure identical requirements and editorial improvements as were recently reviewed and approved for Unit 1 in Amendment No. 26 dated March 1, 1982. The plant design and operation are the same for Unit 1 and Unit 2. This amendment minimizes the potential for confusion and Technical Specifications violations and allows a consistent set for operating, maintenance and surveillance procedures for the essentially identical units.

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 rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since this amendment does not involve a significant hazards consideration.

8205170153

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendments dated October 28 and November 16, 1981 and March 18, 1982, (2) Amendment No. 13 to License No. NPF-8, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. 20555 and at the George S. Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 5th day of May, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing