Wayne H. Jens
Vice President
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Detroit Edison

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June 6, 1985
NE-85-0710

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

Reference: (1) Fermi 2
NRC Docket No. 50-341

(2) Appendix A to License No. NPF-33
Technical Specifications, Fermi 2

Subject: Technical Specification Change Request

During recent weeks, the need for a number of corrections and clarifications to the Fermi 2 Technical Specifications (Reference 2) has become apparent. Attachment 1 consists of the proposed changes shown as marked up Technical Specification pages. Attachment 2 provides the individual justifications in the same sequence as the marked up pages in Attachment 1.

Detroit Edison requests that these proposed changes be incorporated into the Technical Specifications with the Fermi 2 full power license.

I hereby certify that these proposed changes reflect the plant, Final Safety Analysis Report and the staff's Safety Evaluation Reports in all material respects.

Please direct any questions to Mr. O. K. Earle at (313) 586-4211.

Sincerely,

Wayne H. Jens

cc: Mr. P. M. Byron
Mr. M. D. Lynch
USNRC, Document Control Desk
Washington, D.C. 20555

A001
1/40

I, WAYNE H. JENS, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Wayne H. Jens

WAYNE H. JENS
Vice President
Nuclear Operations

On this 6th day of June, 1985, before me personally appeared Wayne H. Jens, being first duly sworn and says that he executed the foregoing as his free act and deed.

Marcia Buck

Notary Public

MARCIA BUCK
Notary Public, Washtenaw County, MI
My Commission Expires Dec. 28, 1987
acting in Monroe County, Mi.

Attachment 1

Proposed Changes to Fermi 2 Technical Specifications

Pages Attached:

1-10
2-4
B2-9
3/4 1-5
3/4 3-5
3/4 3-16
3/4 3-17
3/4 3-44
3/4 3-54
3/4 3-55
3/4 3-68
3/4 3-69
3/4 3-73
3/4 4-1
3/4 4-7
3/4 5-4
3/4 6-13
3/4 6-15
3/4 6-17
3/4 6-21
3/4 7-8
3/4 7-13
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3/4 8-31
3/4 8-32
3/4 9-3
3/4 9-4
3/4 9-7
3/4 9-12
3/4 9-13
3/4 10-1
3/4 10-3
3/4 10-5
3/4 11-17
3/4 11-20
B3/4 10-1
6-7
6-8
6-18

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown#,***	> 200°F
4. COLD SHUTDOWN	Shutdown#,##,***	≤ 200°F
5. REFUELING*	Shutdown or Refuel**,#	≤ 140°F

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

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

**See Special Test Exceptions 3.10.1 and 3.10.3.

***The reactor mode switch may be placed in the Refuel position while a single control rod is being recoupled or withdrawn provided that the one-rod-out interlock is OPERABLE.

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-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.66 W+51%, with a maximum of	≤ 0.66 W+54%, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1068 psig	≤ 1088 psig
4. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Float Switch	≤ ~~125 gallons**~~ 594'8"	≤ ~~160 gallons**~~ 596'0"
b. Level Transmitter	≤ ~~100 gallons**~~ 592'6"	≤ ~~160 gallons**~~ 596'0"
9. Turbine Stop Valve - Closure	≤ 5% closed	≤ 7% closed
10. Turbine Control Valve Fast Closure	Initiation of fast closure	N.A.
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.
13. Backup Manual Scram	N.A.	N.A.

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

~~**Volume is from closed drain valve C11-F011.~~

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. ~~The trip setpoint for the combined scram discharge volume is equivalent to a contained volume of 125 gallons of water.~~

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of 7% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient.

10. Turbine Control Valve Fast Closure

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection with or without coincident failure of the turbine bypass valves. The turbine control valve (TCV) fast closure signal is generated independently in each valve control logic and connected directly to the Reactor Protection System. The signal to the Reactor Protection System is generated simultaneously with the deenergizing of the solenoid dump valves which produces control valve fast closure. Therefore, when TCV fast closure occurs, a scram trip signal is initiated.

11. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

13. Backup Manual Scram

The Backup Manual Scram is a diverse method for manual scram and provides a second means for manual reactor trip capability.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6, and 4.1.3.7.

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 float response by verification of proper float switch actuation within 72 hours after each scram from a pressurized condition greater than or equal to 900 psig.

~~*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 2 provided the surveillance is performed within 12 hours after achieving less than or equal to 50% ROD DENSITY.~~

{Unless adequate shutdown margin has been demonstrated,

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) 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 OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* ~~and shutdown margin demonstrations are being performed per Specification 3.10.3~~.

(d) When the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and per Specification 3.9.2, 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is < 154 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
2. REACTOR WATER CLEANUP SYSTEM ISOLATION		
a. Δ Flow - High	≤ 55.1 gpm	≤ 63.4 gpm
b. Heat Exchanger/Pump/High Energy Piping Area Temperature - High	≤ 175°F**	≤ 183°F**
c. Heat Exchanger/Pump Area Ventilation Δ Temperature - High	≤ 50°F**	≤ 53°F**
d. SLCS Initiation	NA	NA
e. Reactor Vessel Low Water Level - Level 2	≥ 110.8 inches*	≥ 103.8 inches
f. NRHX Outlet Temperature - High	≤ 140°F	≤ 150°F
g. Manual Initiation	NA	NA
3. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High		
1. Differential Pressure	≤ 87.0 inches H_2O/90,875 lbm/hr**	≤ 95.0 inches H_2O/94,865 lbm/hr
2. Time Delay	3 seconds	3±2 seconds
b. RCIC Steam Supply Pressure - Low	≥ 62 psig	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure - High	≤ 10 psig	≤ 20 psig
d. RCIC Equipment Room Temperature - High	≤ 150°F**	≤ 160°F**
e. Manual Initiation	NA	NA

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION		
a. HPCI Steam Line Flow - High		
1. Differential Pressure	$\leq$ 395.0 inches H_2O/536,625** lbm/hr	$\leq$ 410.0 inches H_2O/546,165 lbm/hr
2. Time Delay	3 seconds	3±2 seconds
b. HPCI Steam Supply Pressure - Low	$\geq$ 100 psig	$\geq$ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 20 psig
d. HPCI Equipment Room Temperature - High	$\leq$ 150°F**	$\leq$ 162°F**
e. Manual Initiation	NA	NA
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION		
a. Reactor Vessel Low Water Level - Level 3	$\geq$ 173.4 inches*	$\geq$ 171.9 inches
b. Reactor Vessel (Shutdown Cooling Cut-in Permissive Interlock) Pressure - High	$\leq$ 89.5 psig***	$\leq$ 95.5 psig***
c. Manual Initiation	NA	NA

*Above TAF. 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.

***Represents steam dome pressure; actual trip setpoint is corrected for cold water head with reactor vessel flooded.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + 40%*	≤ 0.66 W + 43%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - High	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Setdown	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 2 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ ~~70 gallons**~~ 589'11½"	≤ ~~90 gallons**~~ 591'0"
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125% of rated flow	≤ 111/125% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

*The rod block function is varied as a function of recirculation loop drive flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

~~**Volume is measured from closed drain valve C11-F011.~~

INSTRUMENTATION

METEOROLOGICAL MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.3 The meteorological monitoring instrumentation channels shown in Table 3.3.7.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

less than the required channels operable in Table 3.3.7.3-1

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

The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.3 Each of the above required meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.3-1.

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS ~~INSTRUMENTS~~ OPERABLE
a. Wind Speed	
1. Elev. 10 meters	1
2. Elev. 60 meters	1
b. Wind Direction	
1. Elev. 10 meters	1
2. Elev. 60 meters	1
c. Air Temperature Difference	
1. Elev. 10/60 meters	1

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	FIRE DETECTION ZONE	TOTAL NUMBER OF INSTRUMENTS* IONIZATION (x/y)	PHOTOELECTRIC (x/y)	FIXED THERMAL (x/y)	INFRARED (x/y)
a. Reactor Building#					
1. Torus Area	1	8/0			
2. NW Corner Rooms, RHR Pump	2	4/0			
3. SW Corner Rooms, RHR Pump	3	4/0			
4. SE Corner Rooms, CRD HPCI	4	~~8~~ 9/0			
5. NE Corner Rooms, RCIC	5	5/0			
6. First Floor	7	20/0		8/0	
7. EECW System Area Second Floor	10	21/0			
8. Third Floor	15	15/0			
9. Fourth Floor	17	8/0		2/0	
10. Refueling Area, Fifth Floor	17				~~8~~ 10/0
b. Auxiliary Building					
1. Basement, N Control Air Equipment	4	6/0			
2. Corridors, 562', 563'	5	2/0	2/0		
3. First Floor Mezzanine, Cable Trays, 583', 603'	6	17/0			
4. Switchgear Room, Corridor Area Second Floor	9	9/0			
5. Cable Tunnel	9	10/0			
6. Cable Tray Area Second Floor Mezzanine	9A	0/22			
7. DC/MCC Room, Third Floor	14	0/10			
8. Switchgear, Battery and M-G Rooms, Third Floor	14	14/0			
9. Fourth Floor	16	6/0			
10. Fifth Floor	16	~~21~~ 25/0			

3.

TABLE 3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	FIRE DETECTION ZONE	TOTAL NUMBER OF INSTRUMENTS* IONIZATION (x/y)	PHOTOELECTRIC (x/y)	FIXED THERMAL (x/y)	INFRARED (x/y)
c. Control Center					
1. Relay Room	8	0/27			
2. Cable Spreading Room	11	0/28			
3. Control Room	12	50/0	4/0	2/0	
4. Computer Room	13	0/13			
5. Computer Room above Drop ceiling	13	5/0	2/0		
d. RHR Complex					
1. Division I Pump Room	50	8 ~~4~~/0			
2. Division II Pump Room	51	8 ~~4~~/0			
3. EDG 11 Room Suppression				0/8	
4. EDG 12 Room Suppression				0/8	
5. EDG 13 Room Suppression				0/8	
6. EDG 14 Room Suppression				0/8	
7. EDG 11 Switchgear Room	52	6/0			
8. EDG 12 Switchgear Room	53	6/0			
9. EDG 13 Switchgear Room	54	6/0			
10. EDG 14 Switchgear Room	55	6/0			
e. General Service Water Pump House					
1. First Floor	31	2/0		3/0	

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

y is number of Function B (actuation of fire suppression systems and early warning and notification.) instruments.

#The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

TABLE 3.3.7.11-1 (Continued)

TABLE NOTATION

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

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

b. At least two technically qualified individuals independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 111 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that~~, at least once per 8 hours, grab samples are collected~~ and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml, for Cs-137 within the following 4 hours.

ACTION 112 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

grab samples are collected at least once per 8 hours

Otherwise, suspend release of radioactive effluents via this pathway.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation. ~~with~~ When

~~a.~~ Total core flow is less than ~~greater than or equal to~~ 45% of rated core flow, then ~~or~~

~~b.~~ THERMAL POWER must be less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

   1. Monitor the APRM and LPRM** noise levels (Surveillance 4.4.1.1.3):

      a) Within 8 hours of entry into this condition and at least once per 24 hours thereafter while in this condition and,

      b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER in an hour by control rod movement.

   2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

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*See Special Test Exception 3.10.4.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored when operating with a nonsymmetric control rod pattern. Only the center of the core LPRM string detectors A and C and two other LPRM string detectors A and C need be monitored for operations with a symmetric control rod pattern.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

5 safety/relief valves @ 1110 psig $\pm$1%
5 safety/relief valves @ 1120 psig $\pm$1%
5 safety/relief valves @ 1130 psig $\pm$1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of less than 11 of the above ~~one or more of the above required~~ safety/relief valves OPERABLE ~~inoperable~~, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 95°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 95°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve position indicators inoperable, restore the inoperable indicator(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The valve position indicator for each safety/relief valve shall be demonstrated OPERABLE with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a

~~a. CHANNEL FUNCTIONAL TEST** at least once per 31 days,~~

~~b.~~ CHANNEL CALIBRATION at least once per 18 months.~~***~~

4.4.2.1.2 At least 1/2 of the safety relief valves shall be set pressure tested at least once per 18 months, such that all 15 safety relief valves are set pressure tested at least once per 40 months.

---

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

**Any portion of this surveillance requirement which requires entry into the primary containment and whose surveillance interval expires when the primary containment is inerted may be rescheduled to the next time the primary containment is not inerted.

***The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure of greater than or equal to 270 psig, corresponding to a reactor vessel pressure of ≥ 100 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure of ≥ 230 psig, corresponding to a reactor vessel to primary containment differential pressure of ≥ 20 psig.

Insert below

3. The HPCI pump develops a flow of at least 5000 gpm ~~against a test line pressure of ≥ 1100 psig~~ when steam is being supplied to the turbine at 1000 +20 , -80 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

2. For the HPCI system, verifying that:

Insert below

a) The system develops a flow of at least 5000 gpm ~~against a test line pressure of 265 psig, corresponding to a reactor vessel pressure of ≥ 165 psig,~~ when steam is being supplied to the turbine at 165 + 50, - 0 psig.*

b) The suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.

3. Performing a CHANNEL CALIBRATION of the CSS and the LPCI system discharge line "keep filled" alarm instrumentation.

4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 1.0 psid.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

Insert in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit 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

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth (At least one at each elevation)
a.	590'0" ~~574'1"~~	90°, 135°, 270° or 316° ~~160°, 200°, 300° or 330°~~
b.	597'0"	35°, 75°, 93°, 135°, 175°, 200°, 246°, 272°, 306° or 345°
c.	621'8"	0°, 90°, 180° or 270°
d.	648'6"	45°, 135°, 225° or 315°
e.	662'0"	0°, 90°, 180° or 285° ~~270°~~
f.	665'6"	0° or 180°

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION SYSTEMS

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:

1. Volume between 121,080 ft3 and 124,220 ft3, equivalent to a level between 14'4" (-2 inches indication) and 14'8" (+2 inches indication) and a

2. Maximum average temperature of 95°F during OPERATIONAL CONDITION 1 or 2, except that the maximum average temperature may be permitted to increase to:

a) 105°F during testing which adds heat to the suppression chamber.

See attached insert

b) 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

~~c) 120°F with the main steam line isolation valves closed following a scram.~~

b. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1-inch diameter orifice at a differential pressure of approximately 1 psid.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the suppression chamber water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~In OPERATIONAL CONDITION 1 or 2~~ With the suppression chamber average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except as permitted above:

1. With the suppression chamber average water temperature greater than 105°F during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression chamber average water temperature greater than:

a) 95°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

3.6.2.1.a.3 INSERT

3. Maximum average temperature of 95°F during OPERATIONAL CONDITION 3, except that the maximum average temperature may be permitted to increase to 120°F with the main steam line isololation valves closed following a scram.

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

2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:

   a) Suppression chamber average water temperature to be less than or equal to 110°F, and

   b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.

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

d. ~~c.~~ By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 160°F and reactor coolant system pressure greater than 200 psig.

e. ~~d.~~ At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

f. ~~e.~~ By verifying eight suppression pool water temperature instrumentation channels 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 high temperature alarm setpoint for ≤ 105°F.

g. ~~f.~~ By verifying both narrow range suppression chamber water level instrumentation channels 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 alarm setpoint for:

1. High water level ≤ 14'8"
2. Low water level ≥ 14'4" (TWMS Narrow Range)

h. ~~g.~~ At least once per 18 months by conducting a drywell-to-suppression chamber bypass leak test at an initial differential pressure of 1 psi and verifying that the differential pressure does not decrease by more than 0.20 inch of water per minute for a period of 10 minutes. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.*

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

* Except for TIP shear valves which are demonstrated OPERABLE per Specification 4.6.3.5.

PLANT SYSTEMS

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The control room emergency filtration system shall be OPERABLE with the system composed of:

a. The passive components of the emergency makeup air filter train.

b. The passive components of the emergency recirculation air filter train.

c. Two emergency makeup inlet air heaters.

d. Two recirculation fans.

e. Two return and supply fans.

f. A flowpath capable of:

1. Recirculating control room air.

2. Supplying emergency makeup air to the control room.**

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

ACTION:

a. With the control room air temperature greater than 95°F but less than 105°F, restore the control room air temperature to less than or equal to 95°F within 12 hours or go to a 4 hour operating shift. With the control room air temperature greater than or equal to 105°F, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION 1, 2, or 3:

1. With one of the above required filter trains or flow paths inoperable, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With a recirculation, return or supply fan; emergency makeup inlet air heater; damper; or other required redundant component inoperable, restore the inoperable component to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. In OPERATIONAL CONDITION 4, 5 or *:

1. With a recirculation, return or supply fan; emergency makeup inlet air heater; damper; or other required redundant component inoperable, restore the inoperable component to OPERABLE status within 7 days or initiate and maintain operation of the system in the recirculation mode of operation.

2. With the control room emergency filtration system inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

d. The provisions of Specification 3.0.3 are not applicable in Operational Condition *.

*When irradiated fuel is being handled in the secondary containment.
**Not applicable in the chlorine mode of operation.
~~***Not applicable prior to achieving criticality in OPERATIONAL CONDITION 2 after initial fuel load.~~

TABLE 3.7.3-1 (Continued)

SURVEY POINTS FOR SHORE BARRIER*

SURVEY POINT	LOCATION** NORTH-SOUTH	EAST-WEST	DECEMBER 1984 CONTROL ELEVATION
9A	N7529	E5948	583.04
9B	N7531	E5961	582.10
9C	N7531	E5965	579.91
9D	N7526	E5973	575.13
10A	N7612	E5937	583.85
10B	N7610	E5950	582.21
10C	N7618	E5961	582.56
10D	N7616	E5972	576.58
11A	N7721	E5940	583.15
11B	N7721	E5956	582.08
11C	N7718	E5963	579.82
11D	N7722	E5971	576.43
12A	N7814	E5949	~~581.66~~ 581.36
12B	N7809	E5955	581.11
12C	N7814	E5965	578.88
12D	N7815	E5975	577.81

*Measuring reference points are anchored into the capstones using center notched self-drilling bolts.

**See Figure B3/4.7.3-1 for location sketch.

PLANT SYSTEMS

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days, otherwise be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that the pump flow controller is in the correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

including injection line losses

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 ~~±15~~ +50,-0 psig.* Insert below

3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

Insert with a system head corresponding to reactor vessel operating pressure including injection line losses

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.5f., an additional ~~10%~~ 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.5-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.5f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.5-1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the points falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.3 The following low pressure CO_2 systems shall be OPERABLE:

a. Emergency diesel generators, RHR complex.

b. Standby gas treatment system charcoal filters, Auxiliary Building, elevation 677'6".

c. Cable tray area, Auxiliary Building, elevation 631'.

d. Outside Division II switchgear room, Auxiliary Building, elevation 643'6".

APPLICABILITY: Whenever equipment protected by the CO_2 systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.7.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4.7.7.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than 50% full for systems a and b above and greater than 40% full for systems c and d above, and pressure to be greater than 330 250 psig but less than ~~315~~ psig for all of the systems.

b. At least once per 18 months by verifying:

1. The system, including associated ventilation system fire dampers and fire door release mechanisms, actuates, manually and/or automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

PLANT SYSTEMS

3/4.7.8 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.8 All fire rated assemblies, including walls, floors/ceilings, cable tray enclosures and other fire barriers, separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area, and all sealing devices in fire rated assembly penetrations including fire doors, fire dampers, cable, piping and ventilation duct penetration seals and ventilation seals, shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour establish a continuous fire watch on at least one side of the affected assembly(s) and/or sealing device(s) or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly(s) and sealing device(s) and establish an hourly fire watch patrol.

b. If the fire door supervision system is inoperable, the affected doors shall be verified per Specification 4.7.8.2.a and b until the supervision system is OPERABLE.

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

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each of the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE at least once per 18 months by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly.

b. Each fire damper and associated hardware.

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

~~a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.~~

a ~~b.~~ The position of each locked-closed fire door at least once per 7 days.

b ~~c.~~ That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

4.7.8.3 The fire door supervision system shall be verified OPERABLE for each electrically supervised fire door by performing a CHANNEL FUNCTIONAL TEST at least once per 31 days.

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

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 fuel 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 fuel tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds.* The generator voltage and frequency shall be 4160 ± 420 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 ~~2850~~ kW in less than or equal to 150 seconds,* and operates with this load for at least 60 minutes.

an indicated
2500 - 2600

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 ~~225~~ 215 psig.

*All diesel generator starts for the purpose of this Surveillance Requirement may be preceded by an engine prelube period. The diesel generator start (10 sec)/load (150 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

7. Verifying that all automatic diesel generator trips, except overspeed, generator differential, low lube oil pressure, crankcase overpressure, and failure to start are automatically bypassed for an emergency start signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 22 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~2850~~ kW and during the remaining 2 hours of this test, the diesel generator shall be loaded to ~~3135~~ kW. The generator voltage and frequency shall be 4160 ± 420 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.e.4.b).*

an indicated 2500-2600

an indicated 2800-2900

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW.

10. 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.

11. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

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

    a) 4160-volt ESF bus lockout.

    b) Differential trip.

    c) Shutdown relay trip.

---

*If Surveillance Requirement 4.8.1.1.2.e.4.b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at ~~2850~~ kW for 1 hour or until operating temperature has stabilized.

an indicated 2500-2600

SURVEILLANCE REQUIREMENTS (Continued)

f. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all four diesel generators simultaneously, during shutdown, and verifying that all four diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

g. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section 11 Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests ~~on a per nuclear unit basis~~ is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

of any diesel generator

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last ~~100~~ 20 Valid Tests*	Number of failures in Last 100 Valid Tests	Test Frequency
≤ 1	≤ 4	At least once per 31 days
≥ 2	≥ 5	At least once per ~~14~~ 7 days**
~~3~~		~~At least once per 7 days~~
~~≥ 4~~		~~At least once per 3 days~~

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the ~~last 100 tests are~~ number of valid tests and failures is determined on a per ~~nuclear unit~~ diesel generator basis. For the purposes of this test schedule, only valid tests conducted after the OL issuance date shall be included in ~~the~~ this computation. ~~of the "last 100 valid tests."~~ Entry into this test schedule shall be made at the 31 day test frequency.

**Maintain this frequency until seven (7) consecutive valid tests have been performed and the number of failures in the last 20 valid tests has been reduced to one (1) or less.

SURVEILLANCE REQUIREMENTS (Continued)

2. By functionally testing the 480 volt circuit breakers. Testing of these circuit breakers shall consist of injecting a current in excess of 120% of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation of the affected equipment.

~~3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a destructive test to demonstrate that the fuses meet the manufacturer's time versus current design criteria. Fuses removed for the functional testing shall be replaced with fuses* from a batch as noted below prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.~~

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

~~*Replacement fuses will be supplied from a batch in which each fuse of a ten percent sample of that type fuse meets the manufacturer's time versus current design criteria by destructive testing.~~

TABLE 3.8.4.2-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP SETPOINT (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
1. 4.16-kV Circuit Breaker					
B-31-P003A Recirc Pump A Generator Field Breaker	GE	AC-50 (K9A, K22A)	1440A	NA	B31-C001A Recirc Pump A Motor
2. 4.16-kV Circuit Breaker					
B31-P003B Recirc Pump B Generator Field Breaker	GE	AC-50 (K9B, K22B)	1440A	NA	B31-C001B Recirc Pump B Motor
3. 480-V A.C.			TRIP OR FUSE RATING (A)		
~~30 A fuse disconnect (MCC 72E-3A)~~	~~Bussmann (FRS)~~	~~72E-3A-1A(R)~~	~~15 A~~	~~N.A.~~	~~G1101-C001B Drywell floor drain sump 72 pump~~
~~72E-3A-1A(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-3A-1A(R)~~	~~15 A~~	~~N.A.~~	
~~30A fuse disconnect (MCC 72E-3A)~~	~~Bussmann (FRS)~~	~~72E-3A-1B(R)~~	~~15 A~~	~~N.A.~~	~~G1101-C006B Drywell equipment drain sump 71 pump~~
~~72E-3A-1B(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-3A-1B(R)~~	~~15 A~~	~~N.A.~~	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
~~30 A fuse disconnect (MCC 72E-3A)~~	~~Bussmann (FRS)~~	~~72E-3A-2B(R)~~	~~15 A~~	~~N.A.~~	~~B3352-F102 common recirculation line valve~~
~~72E-3A-2B(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-3A-2B(R)~~	~~15 A~~	~~N.A.~~	
~~30 A fuse disconnect (MCC 72E-3A)~~	~~Bussmann (FRS)~~	~~72E-3A-2C(R)~~	~~15 A~~	~~N.A.~~	~~B3352-F100 recirculation line B valve~~
~~72E-3A-2C(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-3A-2C(R)~~	~~15 A~~	~~N.A.~~	
15 A Circuit breaker (MCC 72E-3A)	ITE (HE3B015)	72E-3A-1C(R)	15 A	N.A.	B3101-C001B recirculation pump B motor heater
~~72E-3A-1C(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-3A-1C(R)~~	~~15 A~~	~~N.A.~~	
~~30A fuse disconnect (MCC 72E-3A)~~	~~Bussman (FRS)~~	~~72E-3A-2A(R)~~	~~15 A~~	~~N.A.~~	~~B3351-F100 recirculation line A valve~~
~~72E-3A-2A(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-3A-2A(R)~~	~~15 A~~	~~N.A.~~	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
~~30 A fuse disconnect (MCC 72B-4A)~~	~~Bussmann (FRS)~~	~~72B-4A-2D(R)~~	~~20 A~~	~~N.A.~~	~~B3101-F023B recirculation pump B suction valve~~
~~72B-4A-2D(R) (fuse box R1600S002E)~~	~~Bussmann (FRS)~~	~~72B-4A-2D(R)~~	~~20 A~~	~~N.A.~~	
~~60 A fuse disconnect (MCC 72E-5B)~~	~~Bussmann (FRS)~~	~~72E-5B-1B~~	~~60 A~~	~~N.A.~~	~~T4700-C011 drywell cooling fan 11~~
~~72E-5B-1B (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-5B-1B~~	~~60 A~~	~~N.A.~~	
~~60 A fuse disconnect (MCC 72E-5B)~~	~~Bussmann (FRS)~~	~~72E-5B-1A(R)~~	~~60 A~~	~~N.A.~~	~~T4700-C012 drywell cooling fan 12~~
~~72E-5B-1A(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-5B-1A(R)~~	~~60 A~~	~~N.A.~~	
~~60 A fuse disconnect (MCC 72E-5B)~~	~~Bussmann (FRS)~~	~~72E-5B-1A~~	~~60 A~~	~~N.A.~~	~~T4700-C010 drywell cooling fan 10~~
~~72E-5B-1A (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-5B-1A~~	~~60 A~~	~~N.A.~~	
~~60 A fuse disconnect (MCC 72E-5B)~~	~~Bussmann (FRS)~~	~~72E-5B-2A(R)~~	~~60 A~~	~~N.A.~~	~~T4700-C014 drywell cooling fan 14~~
~~72E-5B-2A(R) (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-5B-2A(R)~~	~~60 A~~	~~N.A.~~	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-4D(F)	15 A	N.A.	E5150-F007 RCIC steam line inboard isolation valve
72F-4A-4D(F) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-4D(F)	15 A	N.A.	
30 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-5B(F)	15 A	N.A.	P4400-F608 RBCCW to drywell equipment sump HX inlet valve
72F-4A-5B(F) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-5B(F)	15 A	N.A.	
30 A fuse disconnect (MCC 72F-4A)	Bussmann	72F-4A-3B(R)	15 A	N.A.	G1154-F600 drywell floor drain valve motor
72F-4A-3B(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-3B(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-2B(R)	15 A	N.A.	G1154-F018 drywell equipment drain sump discharge valve motor
30 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-2A	15 A	N.A.	P4400-F615 EECW Return from Drywell Isolation Valve
72F-4A-2B(R) (fuse box R1600S005G)	Bussman (FRS)	72F-4A-2B(R)	15 A	N.A.	
72F-4A-2A (fuse box R1600S005G)	Bussman (FRS)	72F-4A-2A	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
100 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-2A(R)	80 A	N.A.	T4700-C004 drywell cooling fan 4
72F-4A-2A(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-2A(R)	80 A	N.A.	
30 A fuse disconnect (MCC 72C-F)	Bussmann (FRS)	72C-F-5A	20 A	N.A.	B3101-F031B recirculation pump B discharge valve
72C-F-5A (fuse box R1600S003J)	Bussmann (FRS)	72C-F-5A	20 A	N.A.	
30 A fuse disconnect (MCC 72C-F)	Bussmann (FRS)	72C-F-1B	20 A	N.A.	B3101-F031A recirculation pump A discharge valve
72C-F-1B (fuse box R1600S003J)	Bussmann (FRS)	72C-F-1B	20 A	N.A.	
100 A fuse disconnect (MCC 72E-5A)	Bussmann (FRS)	72E-5A-1A	80 A	N.A.	T4700-C003 drywell cooling fan 3
72E-5A-1A (fuse box R1600S004E)	Bussmann (FRS)	72E-5A-1A	80 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
60 A fuse disconnect (MCC 72E-5B)	Bussmann (FRS)	72E-5B-1B(R)	60 A	N.A.	T4700-C013 drywell cooling fan 13
72E-5B-1B(R) (fuse box R1600S004E)	Bussmann (FRS)	72E-5B-1B(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72F-4A)	Bussmann (FRS)	72F-4A-4D(R)	50 A	N.A.	E1150-F608 reactor recirculation extractor isolation to RHR valve
72F-4A-4D(R) (fuse box R1600S005G)	Bussmann (FRS)	72F-4A-4D(R)	50 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2A	15 A	N.A.	G1101-C001A drywell floor drain sump 72 pump
72B-4A-2A (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2A	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2B	15 A	N.A.	G1101-C006A drywell equipment drain sump 71 pump
72B-4A-2B (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2B	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72B-2A)	Bussmann (FRS)	72B-2A-4B	15 A	N.A.	P4400-F616 EECW return drywell
72B-2A-4B (fuse box R1600S002D)	Bussmann (FRS)	72B-2A-4B	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-1D(R)	15 A	N.A.	G3352-F101 vessel drain line recirculation valve
72B-4A-1D(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1D(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-1A(R)	15 A	N.A.	G1154-F015 drywell recirculation equipment drains sump valve
72B-4A-1A(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1A(R)	15 A	N.A.	
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-2C(R)	15 A	N.A.	P5000-F604 drywell station air inboard isolation valve

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
72B-4A-2C(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-2C(R)	15 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-2B	60 A	N.A.	T4700-C007 drywell cooling fan 7
72B-4C-2B (fuse box R1600S002F)	Bussmann (FRS)	72B-4C-2B	60 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-1A(R)	60 A	N.A.	T4700-C008 drywell cooling fan 8
72B-4C-1A(R) (fuse box R1600S002F)	Bussmann (FRS)	72B-4C-1A(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-1D	60 A	N.A.	T4700-C005 drywell cooling fan 5
72B-4C-1D (fuse box R1600S002F)	Bussmann (FRS)	72B-4C-1D	60 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-2A	60 A	N.A.	T4700-C006 drywell cooling fan 6
72B-4C-2A (fuse box R1600S002F)	Bussmann (FRS)	72B-4C-2A	60 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-4A	20 A	N.A.	E4150-F002 HPCI steam supply inboard isolation valve 1
72C-3A-4A (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-4A	20 A	N.A.	
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-3D	15 A	N.A.	E1150-F022 RHR head spray inboard isolation valve
72C-3A-3D (fuse box R16[illegible])	Bussmann (FRS)	72C-3A-3D	15 A	N.A.	
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-4B	15 A	N.A.	B2103-F016 main steam line drains inboard isolation valve
72C-3A-4B (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-4B	15 A	N.A.	
30 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5B	15 A	N.A.	G3352-F001 cleanup supply inboard isolation valve

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
72B-3A-5B (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-5B	15 A	N.A.	
100 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-1A(R)	80 A	N.A.	T4700-C001 drywell
72B-3A-1A(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-1A(R)	80 A	N.A.	
60 A fuse disconnect (MCC 72B-4C)	Bussmann (FRS)	72B-4C-1B(R)	60 A	N.A.	T4700-C009 drywell cooling fan 9
72B-4C-1B(R) (fuse box R1600S002F)	Bussmann (FRS)	72B-4C-1B(R)	60 A	N.A.	
60 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-3C	40 A	N.A.	E1150-F009 RHR suction cooling inboard isolation
72C-3A-3C (fuse box R16005003H)	Bussmann (FRS)	72C-3A-3C	40 A	N.A.	E1150-F009 RHR suction cooling inboard isolation valve
30 A fuse disconnect (MCC 72B-4A)	Bussmann (FRS)	72B-4A-1C(R)	20 A	N.A.	B3101-F023A recirculation Pump A suction valve

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
72B-4A-1C(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-4A-1C(R)	20 A	N.A.	
30 A fuse disconnect (MCC 72C-3A)	Bussmann (FRS)	72C-3A-1A	15 A	N.A.	P4400-F614 drywell penetration cooling jacket inlet valve
72C-3A-1A (fuse box R1600S003H)	Bussmann (FRS)	72C-3A-1A	15 A	N.A.	
30 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5B(R)	20 A	N.A.	T4803-F601 nitrogen supply drywell inboard isolation valve
72B-3A-5B(R) (fuse box R1600S002E)	Bussmann (FRS)	72B-3A-5B(R)	20 A	N.A.	
30 A fuse disconnect (MCC 72B-3A)	Bussmann (FRS)	72B-3A-5C(R)	15 A	N.A.	T4803-F602 drywell inboard isolation valve
72B-3A-5C(R) (fusa box R1600S002E)	Bussmann (FRS)	72B-3A-5C(R)	15 A	N.A.	

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
3. 480-A.C. (Continued)					
~~100 A fuse disconnect (MCC 72C-3A)~~	~~Bussmann (FRS)~~	~~72C-3A-9C~~	~~80 A~~	~~N.A.~~	~~T4700-C002 drywell cooling fan 2~~
~~72C-3A-9C (fuse box R1600S003H)~~	~~Bussmann (FRS)~~	~~72C-3A-9C~~	~~80 A~~	~~N.A.~~	
15 A circuit breaker (MCC 72B-4A)	ITE (HE3B015)	72B-4A-1A	15 A	N.A.	B3101-C001A recirculation Pump A motor heater
~~72B-4A-1A (fuse box R1600S002E)~~	~~Bussmann (FRS)~~	~~72B-4A-1A~~	~~15 A~~	~~N.A.~~	
~~30 A fuse disconnect (MCC 72B-3A)~~	~~Bussmann (FRS)~~	~~72B-3A-5A(R)~~	~~15 A~~	~~N.A.~~	~~T4901-F601 nitrogen supply inboard isolation valve~~
~~72B-3A-5A(R) (fuse box R1600S002E)~~	~~Bussmann (FRS)~~	~~72B-3A-5A(R)~~	~~15 A~~	~~N.A.~~	
~~72E-5A-5E (fuse box R1600S004E)~~	~~Bussmann (FRS)~~	~~72E-5A-5E~~	~~15 A~~	~~N.A.~~	
~~30 A fuse disconnect (MCC 72E-5A)~~	~~Bussmann (FRS)~~	~~72E-5A-5E~~	~~15 A~~	~~N.A.~~	~~T4901-F602 inboard isolation valve~~

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
4. 120-V ac					
30 A fuse block (single pole)	Bussmann (FNQ)	72B-3A-1A(R)	15 A	N.A.	T4700-C001 drywell cooling fan 1 motor winding heater
72B-3A-1A(R) (fuse box R1600S002E)	Bussmann (FRN)	72B-3A-1A(R)	15 A	N.A.	T4700-C002 drywell cooling fan 1 motor winding heater
30 A fuse block (single pole)	Bussmann (FNQ)	72C-3A-9C	15 A	N.A.	T4700-C002 drywell cooling fan 2 motor starter control circuit
72C-3A-9C (fuse box R1600S003H)	Bussmann (FRN)	72C-3A-9C	15 A	N.A.	T4700-C002 drywell cooling fan 2 motor starter control circuit
30 A fuse block (single pole)	Bussmann (FNQ)	72C-3A-9C	15 A	N.A.	T4700-C002 drywell cooling fan 2 motor winding heater
72C-3A-9C (fuse box R1600S003H)	Bussmann (FRN)	72C-3A-9C	15 A	N.A.	T4700-C002 drywell cooling fan 2 motor winding heater
30 A fuse block (single pole)	Bussmann (FNQ)	72E-5A-1A	15 A	N.A.	T4700-C003 drywell cooling fan 3 motor winding heater
72E-5A-1A (fuse box R1600S004E)	Bussmann (FRN)	72E-5A-1A	15 A	N.A.	T4700-C003 drywell cooling fan 3 motor winding heater
30 A fuse block (single pole)	Bussmann (FNQ)	72F-4A-2A(R)	15 A	N.A.	T4700-C004 drywell cooling fan 4 motor winding heater

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TYPE	SOURCE	TRIP OR FUSE RATING (A)	RESPONSE TIME ms/cycle	SYSTEMS/COMPONENTS POWERED
72F-4A-2A(R) (fuse box R1600S005G)	Bussmann (FRN)	72F-4A-2A(R)	15 A	N.A.	T4700-C004 drywell cooling fan 4 motor winding heater
30A fuse block (single pole) (H21-P328A)	Bussman (FRN)	Dist. Pnl. H21-P553 Ckt. #13	15 A	N.A.	T4700-C001 and -C002 cooling fans 1 and 2 vibr. sw. reset coils
30A fuse block (single pole)	Bussman (FRN)	Dist. Pnl. H21-P553 Ckt. #13	5 A	N.A.	T4700-C001 and -C002 cooling fans 1 and 2 vibr. sw. reset coils
30A fuse block (single pole) (H21-P328B)	Bussman (FRN)	Dist. Pnl. H21-P555 Ckt. #16	15 A	N.A.	T4700-C003 and -C004 cooling fans 3 and 4 vibr. sw. reset coils
30A fuse block (single pole)	Bussmann (FRN)	Dist. Pnl. H21-P555 Ckt. #16	5 A	N.A.	T4700-C003 and -C004 cooling fans 3 and 4 vibr. sw. reset coils
5. 130V dc					
4.16 kV switchgear Bus 65G, Position G3	Bussmann (FRN)	130V dc at swgr. Bus 65G	15 A (3 fuses)	N.A.	B3100-S001A MG Set A drive motor control circuit
4.16kV switchgear Bus 65G, Position G5	Bussmann (FRN)	130v dc at swgr. Bus 65G	15 A (3 fuses)	N.A.	B3100-S001B MG Set B drive motor control
480 V switchgear Bus 72F, Position 2C	Bussmann (FRN)	130V dc at swgr. Bus 72F	15 A (3 fuses)	N.A.	G3303-C001A reactor water clean-up system recirculating Pump "A" drive motor control circuit
480V switchgear Bus 72E, Position 2D	Bussmann (FRN)	130V dc at swgr. Bus 72E	15 A (3 fuses)	N.A.	G3303-C001B reactor water clean-up system circulating pump "B" drive motor control circuit

{Unless adequate shutdown margin has been demonstrated,

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 At least 2 source range monitor* (SRM) channels shall be OPERABLE and inserted to the normal operating level with:

a. Annunciation and continuous visual indication in the control room,

b. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and

c. ~~T~~the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.# ~~and shutdown margin demonstrations are in progress.~~

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS and insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

4.9.2 Each of the above required SRM channels shall be demonstrated OPERABLE by:

a. At least once per 12 hours:

1. Performance of a CHANNEL CHECK,
2. Verifying the detectors are inserted to the normal operating level, and
3. During CORE ALTERATIONS, verifying that the detector of an OPERABLE SRM channel is located in the core quadrant where CORE ALTERATIONS are being performed and another is located in an adjacent quadrant.

*The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

#Not required for control rods removed per Specification 3.9.10.1 and 3.9.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verifying that the channel count rate is at least 0.7* cps:

1. Prior to control rod withdrawal,

2. Prior to and at least once per 12 hours during CORE ALTERATIONS, and

3. At least once per 24 hours.

d. Verifying, within 8 hours prior to and at least once per 12 hours during, that the RPS circuitry "shorting links" have been removed during ~~:~~ any

~~1.~~ ~~T~~the time any control rod is withdrawn ~~,~~** ~~or~~ unless

~~2. Shutdown margin demonstrations.~~

adequate shutdown margin has been demonstrated.

*Provided signal-to-noise ratio is ≥ 2. Otherwise, 3 cps.

**Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communication shall be maintained between the control room and refueling platform personnel.

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

When direct communication between the control room and refueling platform personnel cannot be maintained, immediately suspend CORE ALTERATIONS.*

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling platform personnel shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.*

*Except movement of control rods with their normal drive system.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.*

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

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* Except that 3.9.10.1.d and 4.9.10.1.d do not apply if only a single control rod is being withdrawn from the core.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.*

e. All other control rods are inserted.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.10.1 The provisions of Specifications 3.6.1.1, 3.6.1.3 and 3.9.1 and Table 1.2 may be suspended to permit the reactor pressure vessel closure head and the drywell head to be removed and the primary containment air lock doors to be open when the reactor mode switch is in the Startup position during low power PHYSICS TESTS with THERMAL POWER less than 1% of RATED THERMAL POWER and reactor coolant temperature less than 200°F.

APPLICABILITY: OPERATIONAL ~~CONDITION 2,~~ CONDITIONS 2 and 5, during low power PHYSICS TESTS.

ACTION:

With THERMAL POWER greater than or equal to 1% of RATED THERMAL POWER or with the reactor coolant temperature greater than or equal to 200°F, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits at least once per hour during low power PHYSICS TESTS.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

3.10.3 The provisions of Specification 3.9.1, Specification 3.9.3, and Table 1.2 may be suspended to permit the reactor mode switch to be in the Startup position and to allow more than one control rod to be withdrawn for shutdown margin demonstration, provided that at least the following requirements are satisfied.

a. The source range monitors are OPERABLE per Specification 3.9.2 with the RPS circuitry "shorting links" removed. ~~per Specification 3.9.2.~~

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.

c. The "rod-out-notch-override" control shall not be used during out-of-sequence movement of the control rods.

d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

With the requirements of the above specification not satisfied, immediately place the reactor mode switch in the Shutdown or Refuel position.

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and at least once per 12 hours during the performance of a shutdown margin demonstration, verify that;

a. The source range monitors are OPERABLE per Specification 3.9.2,

b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown demonstration procedures, and

c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.2 and the OPERABILITY requirements of the Drywell Oxygen Concentration instrument of Specification 3.3.7.5 may be suspended during the performance of the Startup Test Program until 6 months after initial criticality.

APPLICABILITY: OPERATIONAL CONDITIONS 1 ~~or~~ and 2.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The number of months since criticality shall be verified to be less than or equal to 6 months at least once per 31 days during the Startup Test Program.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.7 The gross radioactivity rate of noble gases measured near the main condenser steam jet air ejectors shall be limited to less than or equal to 340 millicuries/sec after 30 minute decay.

APPLICABILITY: ~~At all times.~~ OPERATIONAL CONDITIONS 1, 2*, and 3*.

ACTION:

With the gross radioactivity rate of noble gases at the main condenser steam jet air ejector exceeding 340 millicuries/sec after 30 minute decay, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases near the outlet of the main condenser steam jet air ejector shall be continuously monitored in accordance with Specification 3.3.7.12.

4.11.2.7.2 The gross radioactivity rate of noble gases from the main condenser steam jet air ejector shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken near the discharge of the main condenser steam jet air ejector:

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the Offgas Radiation Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady-state fission gas release from the primary coolant.

c. The provisions of Specification 4.0.4 are not applicable.

* When the main condenser air ejector is in operation.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1 2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from outside ~~radwaste~~ storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4, ACTION a.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS.

3/4.10.2 ROD SEQUENCE CONTROL SYSTEM

In order to perform the tests required in the technical specifications it is necessary to bypass the sequence restraints on control rod movement. The additional surveillance requirements ensure that the specifications on heat generation rates and shutdown margin requirements are not exceeded during the period when these tests are being performed and that individual rod worths do not exceed the values assumed in the safety analysis.

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

Performance of shutdown margin demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality ~~does not occur.~~ is properly monitored and controlled. These additional restrictions are specified in this LCO.

3/4.10.4 RECIRCULATION LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 OXYGEN CONCENTRATION

Relief from the oxygen concentration specifications is necessary in order to provide access to the primary containment during the initial startup and testing phase of operation. Without this access the startup and test program could be restricted and delayed.

3/4.10.6 TRAINING STARTUPS

This special test exception permits training startups to be performed with the reactor vessel depressurized at low THERMAL POWER and temperature while controlling RCS temperature with one RHR subsystem aligned in the shutdown cooling mode in order to minimize contaminated water discharge to the radioactive waste disposal system.

SHIFT TECHNICAL ADVISOR (Continued)

Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 29, 1980 NRC letter to all licensees.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director, Nuclear Training, shall meet or exceed the requirements and recommendations of Section 5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 29, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 Onsite Review Organization (OSRO)

FUNCTION

6.5.1.1 The OSRO shall function to advise the Superintendent-Nuclear Production on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The OSRO shall be composed of the:

Chairman	Superintendent-Nuclear Production
Vice-Chairman	Assistant Superintendent-Nuclear Production
~~Second~~ Vice-Chairman	Assistant to Superintendent-Nuclear Production
Member ~~Third Vice-Chairman~~	Operations Engineer
~~Secretary~~	~~Assistant Operations Engineer~~
Member	Technical Engineer
Member	Maintenance Engineer
Member	Radiation Protection-Chemical Engineer
Member	Supervisor Operational Assurance
Member	Reactor Engineer
~~Member~~	~~Administrator~~

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSRO Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSRO activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSRO shall meet at least once per calendar month and as convened by the OSRO Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the OSRO necessary for the performance of the OSRO responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and ~~five~~ four members including up to two alternates.

RESPONSIBILITIES

6.5.1.6 The OSRO shall be responsible for:

a. Review of (1) all procedures required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other procedures or changes thereto as determined by the Superintendent-Nuclear Production to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Manager-Nuclear Operations and to the Nuclear Safety Review Group;

f. Review of all REPORTABLE EVENTS;

g. Review of unit operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Superintendent-Nuclear Production or the Nuclear Safety Review Group;

i. Review of the Security Plan and implementing procedures; ~~and submittal of recommended changes to the Nuclear Safety Review Group;~~

j. Review of the Emergency Plan and implementing procedures; ~~and submittal of recommended changes to the Nuclear Safety Review Group.~~

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1.3-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. ~~Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.~~

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

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

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the ODCM.

Attachment 2

Justification for Proposed Changes to Fermi 2 Technical Specifications

Table 1.2

The proposed change along with the proposed change to Specification 3.9.10.1 would allow the withdrawal of a single control rod after adequate shutdown margin has been demonstrated without the requirement to electrically or hydraulically disarm the rods in a five-by-five array centered on the rod to be withdrawn. This significantly shortens the time required to perform control rod friction testing and scram time testing following maintenance. Further, it eliminates the potential for wear and damage to control rod drive circuitry. It is justified by satisfying the one-rod-out interlock requirement and shutdown margin requirements.

The addition of Operational Condition 3 to the Applicability section allows for the possibility that single rod withdrawal for testing purposes may also be required in Operational Condition 3.

Table 2.2.1-1

The proposed changes to the setpoints and allowable values for the Scram Discharge Volume Water Level - High are to provide more efficient and accurate surveillance testing. The use of elevations, which can be established with great precision, eliminates the need to measure volumes of water during the surveillance. Accurate volume measurements have proven difficult. The float switch setpoint elevation is approximately 2 inches above the switch actuation level which is still conservative. The allowable value is based on having sufficient volume remaining above this elevation to accommodate the water from a full scram and has been selected to correspond approximately with the elevation for 160 gallons of water (the present allowable value).

The setpoint for the level transmitter has been selected to correspond approximately with the present 100 gallon setpoint. This is necessary because of the span of the installed instrument and is conservative. The allowable value for the level transmitter is the same as for the float switch.

BASES 2.2.1.8

The proposed change removes the statement regarding water volume to be consistent with the proposed changes to Table 2.2.1-1.

4.1.3.1.4.a

The proposed change would allow the vent and drain valves to be determined OPERABLE without the significant operational impact involved in the current requirement. It is estimated that the current requirement adds between eight and twelve hours to the critical path following a refueling outage. In addition, the scram initiates the requirement to do many other surveillances which would likely have just been performed and thus involve unnecessary exposure. In particular, the requirement to verify proper float switch actuation requires personnel to be in close proximity to the Scram Discharge Volume (SDV) to perform a test which would not otherwise be required.

Every 18 months Table 4.3.1.1-1, Item 11, requires that the Reactor Mode Switch position be tested. This test, which is done during the refueling outage, causes a full scram. This test will permit the vent and drain valves to be tested to meet the revised requirement. In addition, Specification 4.1.3.2.a requires each control rod to be individually scram timed. This test also demonstrates that the 3/4" line from the Hydraulic Control Unit to the SDV is free of obstruction and provides significant assurance that the 8" diameter scram discharge header is unplugged. Given the diameter of the header (8") and the SDV itself (12"), coupled with these tests, a high level of confidence is gained regarding the operability of the SDV. This confidence can be achieved with the proposed change without the significant impact of the current requirement.

Table 3.3.1-1

The revised wording clarifies that the "shorting links" in the RPS circuitry are not required to be removed if adequate shutdown margin (see Specification 3.1.1) has already been demonstrated. This configuration is acceptable in the Refueling Condition with the one-rod-out interlock operative and the shutdown margin determined with the highest worth rod fully withdrawn.

Table 3.3.2-2

The setpoint value for the RCIC and HPCI Steam Line Flow High Differential Pressure Isolations (refer to items 3.a.1 and 4.a.1) are only initial setpoints. The FSAR requires that, during the Startup Test Program, the maximum required steady-state steam flow for the HPCI and RCIC systems be measured and the trip setpoints be adjusted to actuate at 300 percent of the measured value.

Table 3.3.6-2

The proposed elevations for setpoint and allowable value for the Scram Discharge Volume rod block correspond approximately to the current gallon values. Refer to the discussion on Table 2.2.1-1 for more information.

3.3.7.3.a

The meteorological monitoring system has more than one channel per monitored parameter and monitors other parameters in addition to wind speed, wind direction and air temperature difference (refer to FSAR Table 2.3-29). The revised words clarify the intent of the subject action statement.

Table 3.3.7.3-1

The Limiting Condition for Operation (LCO) for 3.3.7.3 is associated with operability of the entire channel, not only the sensing instrument. The replacement of the word "instruments" with "channels" in Table 3.3.7.3-1 is appropriate and better implements the intent of the subject LCO.

Table 3.3.7.9-1

The revised number of detectors are due to changes in the final design associated with licensing condition 2.C(9)(e). The additional detectors are required to comply with applicable regulatory requirements.

(Note: typo in table number on page 3/4 3-69.)

Table 3.3.7.11-1

The proposed changed to Action 111 eliminates the current ambiguity regarding the time period during which the samples may be analyzed. The four-hour period to analyze the samples provides the results in sufficient time, when compared with the sampling frequency, to suspend the release in a timely manner if required.

The addition of the phrase "Otherwise, suspend release of radioactive effluents via this pathway" to Actions 111 and 112 achieves consistency with Action 110. This eliminates a possible misinterpretation on the action to take at the end of the allowed time period, if the required number of channels is not restored.

3.4.1.1

Based on experience to date, the revision provides clarification to the operators and removes some ambiguity.

3.4.2.1.a

The revised wording clarifies the Action statement with respect to the LCO.

4.4.2.1.1

Performance of the Channel Functional Test requires entry into the drywell. Though the existing footnote eliminates the requirement to perform the surveillance when the containment is inerted, during the startup testing phase the plant may be at high power levels prior to inerting. The test requires at least three persons for two eight hour shifts under normal conditions. Because of ALARA concerns and the rather high temperature to be expected during power operation, the test would require an even longer time and more persons. The pressure switches are not the primary means of detecting an open Safety Relief valve. It is proposed that the channel functional test be deleted entirely and that the 18 month channel calibration is sufficient to provide the required assurance of pressure switch function.

3.5.1

The addition of the footnote provides the recognition that the RHR loops may be aligned in the shutdown cooling mode as required by Specification 3.4.9.1 and that while in this mode remote manual operation of the torus suction valve is required upon a LPCI injection signal to establish the flow path.

4.5.1.b.3 and c.2.a

This specification requires that the HPCI pump discharge pressure be greater than/equal to 1100 psig during 1000 psig surveillance testing, and 265 psig during 165 psig surveillance testing. These pump discharge pressure values were derived from the FSAR requirements to throttle the pump discharge pressure to 100 psi above reactor pressure during Startup injection tests to the Condensate Storage Tank in order to compensate for potential line losses when injecting to the reactor. The 100 psi value is a conservatively high estimate of the line losses. The actual line losses, which will be measured during the Startup Test Program, should be used in future surveillance testing. In addition, the proposed revision makes the HPCI specification more consistent with the wording of the 1000 psig RCIC specification (refer to Specification 4.7.4).

4.6.1.7

The air temperature measurement capability at the 574'1" elevation was deleted and replaced with the noted azimuth locations at elevation 590'0". The design modification was implemented in 1984. In addition, the temperature sensing element previously located at the 662'0" level at azimuth 270° was relocated to azimuth 285°.

3.6.2.1.a.2.c/action b/4.6.2.1.b.3

These changes are required to remove the ambiguity which exists relative to the Operational Condition in which the 120°F limit applies. Following a scram from power, the plant is in Operational Condition 3. Under certain conditions following a scram, the suppression pool temperature may increase to greater than 110°F. Since action statement b.2.b requires that the mode switch be placed in the Shutdown position if the suppression pool temperature exceeds 110°F it seems clear that the allowance to exceed 110°F up to 120°F is intended to apply only in Operational Condition 3.

The proposed change to action statement b clarifies that action b.3 must also be taken if the 120°F limit is exceeded in Operational Condition 3.

The proposed change to 4.6.2.1.b clarifies that the 30 minute surveillance applies only in Operational Condition 3.

4.6.3.1

The TIP shear valve footnote properly qualifies this specification in consideration of the operating characteristics of these valves and Specification 4.6.3.5.

3.7.2

The revision to the Applicability statement is appropriate for issuance of the full power license since the footnote will no longer apply, i.e., criticality will already have been achieved in Operational Condition 2.

Table 3.7.3-1

The December 1984 control elevation for survey point 12A should be revised from "581.66" to "581.86" to exactly reflect the original-survey data.

4.7.4.b

The proposed change clarifies that the test flow path system head must include an allowance for the injection line losses and achieves consistency with proposed wording for the other HPCI and RCIC flow tests.

4.7.4.c.2

This specification does not specify a pump discharge pressure for the 150 psig RCIC surveillance test. The proposed revision would make the 150 psig specification consistent with the 1000 psig RCIC specification. The proposed increase for the steam supply pressure tolerance is needed to allow for the difficulty in control during the surveillance at such low pressures.

4.7.5.e(1)

The current specification for functional testing of snubbers requires an initial sample size of 10% of the total number of subject snubbers in the plant. The proposed change reduces the required number of additional snubbers to be tested from 10% to 5% for every failed snubber discovered during the functional testing. In the absence of a suitable snubber failure data base, it was required that for every failed snubber, an additional 10% of that snubber type was to be tested. Subsequently, the ASME OM-4 group developed a sampling plan which determined that 50% of the initial sample size (10%) need be tested for each failed snubber. It is Detroit Edison's understanding that the NRC finds the ASME position acceptable.

4.7.7.3.2.a

The revised upper limit pressure rating reflects the manufacturer's (Chemetron) recommendation and tolerance for equipment accuracy. The CO_2 storage tanks pressure relief valves are set at 341 psig and 357 psig. Therefore, adequate margin exists between the operating range and the relief-valve settings.

3.7.8/4.7.8.2

Specification 4.7.8.2.a determines a fire door to be inoperable if its related supervision system channel is inoperable. The status of the supervision system channel for a fire door does not degrade the door's capability to resist fire. In addition, the current Specification 4.7.8.2.c provides for daily surveillance of unlocked fire doors without electrical supervision, not a fire watch within 1 hour.

The proposed specification change makes the required action and surveillance for loss of the fire door supervision system commensurate with its design function in comparison to the loss of fire-rated assemblies or sealing devices.

(Note: typo in 3.7.8)

4.8.1.1.2/4.8.1.1.3

Introductory Comment

In January 1985, Detroit Edison experienced bearing failures during testing of its emergency diesel generators (EDGs). Subsequent discussions with the EDG vendor and the NRC, in conjunction with knowledge of generic concerns for diesel generator reliability, has led to proposed Technical Specification changes that reduce the load and frequency for surveillance tests. These concerns are also addressed in Generic Letter 84-15 and I&E Information Notice 85-32, and through the approval of changes to the North Anna Technical Specifications.

4.8.1.1.2.a.5

This specification requires the EDG to be loaded to at least its continuous rating (2850 kW) during routine surveillance testing. To ensure that the value of 2850 is met during the surveillance, the operator typically loads the EDG to the nearest higher scale marking (2900 kW) on the load indicator. This fact together with the inherent calibration and instrument inaccuracies, which could cause the actual load to be greater than the indicated load, results in the possibility of routinely overloading the EDG's. Furthermore, the continuous rating of 3850 kW is a conservatively chosen design value. Preoperational test results measured for the auto sequenced post-LOCA and loss-of-offsite power loads for the EDG's were between 2225 and 2500 kW.

In view of these factors, the proposed change to load the EDG's to an indicated 2500-2600 kW is adequate to demonstrate the ability to supply the actual required emergency loads, eliminates the potential for routine overloading and thus contributes significantly to improving EDG reliability.

4.8.1.1.2.a.7

This specification requires the EDG air receivers to be at a pressure of 225 psig or greater. This requirement verifies the air-start capability for the EDGs. The 225 psig value is the actuation point for charging an air receiver and is not a limiting value for EDG air-start capability. An air receiver pressure indicator may read less than 225 psig due to instrument tolerance and thus cause an insignificant reporting condition.

The verification of EDG air-start capability was demonstrated during preoperational testing. An air receiver pressure of 215 psig is more than adequate to meet the EDG air-start requirements. Accordingly, it is recommended that this specification be revised to 215 psig for the air receivers. Independent of this change, the charging actuation point will remain at 225 psig.

4.8.1.1.2.e.8

The justification for reducing the loads in the same as discussed under section 4.8.1.1.2.a.5. The 3135 kW value corresponds to the "short time" rating of the EDG's (110% of the continuous rating).

The likelihood of overloading the EDG is highest for this particular test. The proposed change to an indicated "2800-2900 kW" instead of 3135 kW represents the continuous rating of the EDG as read to the nearest scale marking and is a realistic test load considering the expected emergency loads and the minimization of potential overloading.

FOOTNOTE to 4.8.1.1.2.e.8

For consistency, refer to the discussion under 4.8.1.1.2.a.5.

4.8.1.1.3

For consistency,refer to the discussion under Table 4.8.1.1.2-1.

Table 4.8.1.1.2-1

This specification defines the Diesel Generator Test Schedule based upon the number of failures in the last 100 valid tests on a per nuclear-unit basis. The recommended changes reflect the EDG concerns of Generic Letter 84-15, I&E Information Notice 85-32 and the recent NRC approval of North Anna revised Technical Specifications. The revised table includes the last 100 and last 20 tests in demonstrating 95% EDG reliability. The test frequency is revised to avoid unwarranted and excessive EDG testing. The criteria footnote for determining the number of failures is revised to a per-diesel-generator basis rather than a nuclear-unit basis, since the failure of one diesel does not result in other EDGs being inoperable. The added footnote defines the frequency when reliability does not meet 95% and re-establishes the acceptance of diesel performance without requiring excessive testing.

4.8.4.2.a.3

We believe the technical basis for this surveillance is not well founded and thus it should be deleted. It is our understanding that several other utilities have recently been successful in doing so and have submitted justification for deleting the surveillance.

The justification involves several aspects. First, the only conceivable failure mode caused by aging or environmental effects for fuses is to fail in a conservative manner. Second, the estimated likelihood of creating a non-conservative condition as a result of removing and testing the fuses is believed to be greater than any potential benefits derived from the testing. This can occur either as a result of inadvertently replacing a fuse with one having a larger current rating or by damaging the fuse holding mechanism. Third, since the containment penetrations which are protected by fuses always have two fuses in series, the probability of a non-conservative failure is extremely remote.

Table 3.8.4.2-1

The deletion from the table of all fuse entries is based upon the proposal to delete the surveillance associated with the fuses.

3.9.2.c/4.9.2.d

The revised wording clarifies that the "shorting links" in the RPS circuitry are not required to be removed if adequate shutdown margin (see Specification 3.1.1) has already been demonstrated. This configuration is acceptable in the Refueling Condition with the one-rod-out interlock operative and the shutdown margin determined with the highest worth rod fully withdrawn.

The revised wording for Specification 4.9.2.d is a continuation of the proposed change to 3.9.2.c (see above). Note: requirements for the demonstration of shutdown margin are provided in Specification 3.10.3.

3.9.5

Removal of the comma clarifies that the subject communications are only required during Operational Condition 5 when Core Alterations are in progress.

3.10.1

The addition of Operational Condition 5 to the Applicability statement for this specification reflects the same test exception noted in Table 1.2 for the Refueling Condition. It is prudent to allow the conduct of low power PHYSICS TESTS to verify proper operation of the SRM and IRM detectors in Operational Condition 5, which is not verified through the performance of normal surveillance testing. The limits of 1% of Rated Thermal Power and 200^{o}F reactor coolant temperature provide adequate protection for these tests in both Operational Conditions 2 and 5.

3.10.3.a

The "shorting links" are not necessarily required to be removed per Specification 3.9.2, as discussed above. The revised wording is required to properly reflect the proposed change to Specification 3.9.2.

3.10.5

Specification 3.3.7.5 (specifically Table 3.3.7.5-1, item 9) includes operability requirements for drywell oxygen monitoring instrumentation for Operational Conditions 1 and 2. These operability requirements will prevent entering Operational Condition 2, and thus Operational Condition 1, because the relief in Special Test Exception Specification 3.10.5 for monitoring drywell oxygen concentration only applies to Operational Condition 1.

The Applicability statement of Specification 3.10.5 should be extended to include Operational Condition 2 to eliminate this problem and properly implement the intent of this specification.

3.11.2.7/4.11.2.7.2

The revised Applicability statement (with footnote) reconciles the LCO and surveillance requirements with the operating conditions under which this specification makes sense. It is also consistent with the technical specifications for other recently licensed plants.

The statement relative to Specification 4.0.4 not being applicable is required because the intent of the surveillance requirements, as written, cannot be satisfied without being in an Operational Condition with main condenser steam jet air ejector operation. For example, during plant startup, the surveillance requirements cannot be satisfied before entering Operational Condition 2*.

4.11.4.2

The revision corrects the reference to the proper storage tanks. (Refer to Specification 3.11.4.a)

BASES 3/4.10.3

Most shutdown margin tests performed currently involve pulling control rods in a normal manner using the same pull sequence as used for plant startups. General Electric now normally provides the information needed to support the shutdown margin calculations in a form reflecting this consideration.

The information provided requires taking the reactor critical, measuring the reactor period and factoring the period into the determination of core reactivity at the time of criticality in order to calculate the shutdown margin. The wording in this basis statement should, therefore, be revised as indicated.

6.5.1.2

The revised OSRO composition does not degrade the technical or experience level of the organization. It reflects a more manageable structure and the fact that the Assistant Operations Engineer is the alternate for the Operations Engineer and that the Administrator's function in support of the OSRO is not safety related.

6.5.1.5

The revised quorum number is adequate to execute the responsibilities and authorities of OSRO. At the same time, it allows more expeditious conduct of necessary, but unscheduled meetings for reviews, approvals and decisions. The limitation on the number of alternates assures sufficient senior personnel are present.

6.5.1.6.i and j

Changes to the Security Plan and Emergency Plan are thoroughly reviewed for the Onsite Review Organization (OSRO) by the Security Plan Committee and Emergency Plan Committee respectively. The results of their reviews are reported to OSRO. These reviews and the overview by OSRO assure the applicable rules and regulations for security and emergency plans are implemented by Detroit Edison.

Issues which are safety-related are provided to the Nuclear Safety Review Group (NSRG). NSRG reviews are implemented by Specifications 6.5.1.6.e and k, 6.5.1.7.b, 6.5.1.8, 6.5.2.7 and 6.5.2.8. These specifications assure NSRG reviews are performed for their scope of responsibility. Deletion of the words, as indicated, for Specifications 6.5.1.6.i and j does not degrade the appropriate level of review for the subject plans, but reconciles Specification 6.5.1.6 with the NSRG scope of responsibility and retains the routine, non-safety-related plan changes within the purview of OSRO.

6.9.1.8

The Offsite Dose Calculation Manual (ODCM) contains the parameters and methodology for determining off ite doses. The reference to the ODCM also reconciles this specification with the reference in Specification 3.11.4.