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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 88
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated January 18, 1984, as supplemented, June 29, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the licensee is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

   (2) Technical Specification

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 88 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. DPR-46

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Revise the Appendix A Technical Specifications by removing the pages listed and inserting identically numbered pages. The revised areas are indicated by marginal lines.

Remove	Insert
iii	iii
4	4
6	6
7	7
28	28
29	29
30	30
31	31
40	40
50	50
52	52
52b	52b
61	61
65	65
68	68
80	80
83	83
84	84
121	121
122	122
133a	133a
134	134
1371	1371
159	159
161	161
162a	162a
165	165
166	166
167a	167a
173	173
182	182
193	193
194	194
195	195
196	196

Remove	Insert
197	197
204	204
208	208
215	215
215d	215d
216b	216b

K. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represent a margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

L. Mode - The reactor mode is established by the mode selector switch. The modes include refuel, run, shutdown and startup/hot standby which are defined as follows:

1. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the REFUEL position. When the mode switch is in the REFUEL position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 825 psig and the reactor protection system is energized with APRM protection (excluding the 15% high flux trip) and RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the mode switch is in the SHUTDOWN position.

4. Startup/Hot Standby Mode - In this mode the reactor protection scram trips initiated by the main steam line isolation valve closure are bypassed, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with APRM (15% SCRAM) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

M. Operable - Operable means a system or component is capable of performing its intended function in its required manner.

N. Operating - Operating means a system or component is performing its intended functions in its required manner

O. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment, and which are not required to be open during accident conditions, are closed.

2. At least one door in each airlock is closed and sealed.

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TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Required When Equipment Operability is Not Assured (1)
Mode Switch in Shutdown	X(7)	X	X	X		1	A
Manual Scram	X(7)	X	X	X		1	A
IRM (17) High Flux	X(7)	X	X	(5)	≤ 120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17) High Flux (Flow biased)				X	≤ (0.66W+54%) (14) $\frac{FRP}{MFLPD}$	2	A or C
High Flux	X(7)	X(9)	X(9)	(16)	≤ 15% Rated Power	2	A or C
Inoperative		X(9)	X(9)	X	(13)	2	A or C
Downscale		(11)		X(12)	≥ 2.5% of indicated scale	2	A or C
High Reactor Pressure NBI-PS-55 A,B,C, & D		X(9)	X(10)	X	≤ 1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X(9)(8)	X(8)	X	≤ 2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	≥ + 12.5 in. indicated level	2	A or D
Scram Discharge Instrument Volume High Water Level CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D		X	X(2)	X	≤ 92 inches	3 (18)	A

1.1 (Cont'd)

D. Cold Shutdown

Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 in. above the top of the normal active fuel zone (top of active fuel is defined in Figure 2.1.1).

2.1.A (Cont'd)

a. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\%) \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7x7 fuel and 13.4 KW/ft for 8x8 fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. IRM

The IRM flux scram setting shall be $\leq$120/125 of scale.

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Action

If a Safety Limit is exceeded, the reactor shall be in at least hot shutdown within 2 hours.

Specifications

A. Reactor Pressure $\geq$800 psia and Core Flow $\geq$10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety.

B. Core Thermal Power Limit (Reactor Pressure <800 psia and/or Core Flow <10%)

When the reactor pressure is <800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66\ W + 54\%$

where:

S = Setting in percent of rated thermal power (2381 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate is that recirculation flow rate which provides 100% coreflow at 100% power)

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TABLE 3.1.1 (Page 2)

REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position: Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Require When Equipment Operability is Not Assured (
Main Steam Line High Radiation RMP-RM-251, A,B,C, & D		X(9)		X	$\leq$ 3 Times normal full power back ground	2	A or D
Main Steam Line Isolation Valve Closure		X(6)(9)		X(6)	$\leq$ 10% of valve closure		
MS-LMS-86 A,B,C, & D						4	A or C
MS-LMS-80 A,B,C, & D						4	A or C
Turbine Control Valve Fast Closure TGF-63/OPC-1,2,3,4				X(4)	$\geq$ 1000 psig turbine control fluid	2	A or B
Turbine Stop Valve Closure SVOS-1(1), SVOS-1(2) SVOS-2(1), SVOS-2(2)				X(4)	$\leq$10% of valve Closure	2	A or B
Turbine First Stage Permissive MS-PS-14 A,B,C, & D		X(9)		X	$\leq$ 30% first stage press.	2	A or B

NOTES FOR TABLE 3.1.1

1. There shall be two operable or tripped trip systems for each function. If the minimum number of operable instrument channels for a trip system cannot be met, the affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power to less than 30% of rated.

   C. Reduce power level to IRM range and place mode switch in the Startup position within 8 hours and depressurize to less than 1000 psig.

   D. Reduce turbine load and close main steam line isolation valves within 8 hours.

2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

3. This note deleted.

4. Permissible to bypass when turbine first stage pressure is less than 30% of full load.

5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.

6. The design permits closure of any two lines without a full scram being initiated.

7. When the reactor is subcritical, fuel is in the vessel, and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a. Mode switch in shutdown.
   b. Manual scram.
   c. IRM high flux. 120/125 indicated scale.
   d. APRM (15%) high flux scram.

8. Not required to be operable when primary containment integrity is not required.

9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

10. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the mode switch is not in RUN.

13. An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there is less than 11 operable LPRM detectors to an APRM.

14. W is the recirculation flow in percent of rated flow.

15. This note deleted.

16. The 15% APRM scram is bypassed in the RUN mode.

17. The APRM and IRM instrument channels function in both the Reactor Protection System and Reactor Manual Control System (Control Rod Withdraw Block, Section 3.2.C.). A failure of one channel will affect both of these systems.

18. The minimum number operable associated with the Scram Discharge Instrument Volume are three instruments per Scram Discharge Instrument Volume and three level devices per RPS channel.

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3.1 BASES (cont'd.)

against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volumes accommodate in excess of 18 gallons of water in each volume and are the low points in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, diverse indication (two level switches and two level transmitters for each discharge volume) has been provided in the instrument volumes which alarm and scram the reactor when the volume of water reaches 92 inches in either volume. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph VII.5.4 USAR). Thus, the IRMs and APRMs are required in the "Refuel" and "Startup/Hot Standby" modes. In the power range, the APRM system provides required protection (refer-

4.1 BASES (cont'd.)

revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20×10^{-6} failures/hour. The bi-stable trip circuits are predicted to have an unsafe failure rate of less than 2×10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system; therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a

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TABLE 3.2.A (Page 1)

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required Wh[...] Component Operabil[...] is Not Assured (2)
Main Steam Line High Rad.	RMP-RM-251, A,B,C,&D	≤ 3 Times Full Power	2	A or B
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D	≥+12.5" Indicated Level	2(4)	A or B
Reactor Low Low Water Level	NBI-LIS-57 A & B #2 NBI-LIS-58 A & B #2	≥-37" Indicated Level	2	A or B
Reactor Low Low Low Water Level	NBI-LIS-57 A & B #1 NBI-LIS-58 A & B #1	≥-145.5" Indicated Level	2	A or B
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	≤ 200°F	2(6)	B
Main Steam Line High Flow	MS-dPIS-116 A,B,C,&D 117, 118, 119	≤ 140% of Rated Steam Flow	2(3)	B
Main Steam Line Low Pressure	MS-PS-134, A,B,C,&D	≥ 825 psig	2(5)	B
High Drywell Pressure	PC-PS-12, A,B,C,&D	≤ 2 psig	2(4)	A or B
High Reactor Pressure	RR-PS-128 A & B	≤ 75 psig	1	D
Main Condenser Low Vacuum	MS-PS-103, A,B,C,&D	≥ 7" Hg (7)	2	A or B
Reactor Water Cleanup System High Flow	RWCU-dPIS-170 A & B	≤ 200% of System Flow	1	C

NOTES FOR TABLE 3.2.A

1. Whenever Primary Containment integrity is required there shall be two operable or tripped trip systems for each function.

2. If the minimum number of operable instrument channels per trip system requirement cannot be met by a trip system, that trip system shall be tripped. If the requirements cannot be met by both trip systems, the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in a cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have the Main Steam Isolation Valves shut within 8 hours.

   C. Isolate the Reactor Water Cleanup System.

   D. Isolate the Shutdown Cooling System.

3. Two required for each steam line.

4. These signals also start the Standby Gas Treatment System and initiate Secondary Containment isolation.

5. Not required in the refuel, shutdown, and startup/hot standby modes (interlocked with the mode switch).

6. Requires one channel from each physical location for each trip system.

7. Low vacuum isolation is bypassed when the turbine stop is not full open, manual bypass switches are in bypass and mode switch is not in RUN.

8. The instruments on this table produce primary containment and system isolations. The following listing groups the system signals and the system isolated.

   Group 1

   Isolation Signals:

   1. Reactor Low Low Low Water Level (≥-145.5 in.)
   2. Main Steam Line High Radiation (3 times full power background)
   3. Main Steam Line Low Pressure (≥825 psig in the RUN mode)
   4. Main Steam Line Leak Detection (≤200°F)
   5. Condenser Low Vacuum (≥7" Hg vacuum)
   6. Main Steam Line High Flow (≤140% of rated flow)

   Isolations:

   1. MSIV's
   2. Main Steam Line Drains

NOTES FOR TABLE 3.2.A (cont'd.)

Isolations

1. Secondary Containment Isolation
2. Start Standby Gas Treatment System

Group 7

Isolation Signals:

1. Reactor Low Low Water Level ($\geq$-37 in)
2. Main Steam Line High Radiation ($\leq$3 times full power background)

Isolations:

1. Reactor Water Sample Valves

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq$ (0.66W + 42%) $\frac{\text{FRP}}{\text{MFLPD}}$ (2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ (0.66W + 40%) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (11)	1(1)(6)
SDV Water Level High CRD-231E, 234E	$\leq$ 46 inches	1(12)

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TABLE 3.2.F

PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION

Instrument	Instrument I.D. No.	Range	Minimum Number of Operable Instrument Channels	Action Required When Minimum Condition Not Satisfied (1)
Reactor Water Level	NBI-LI-85A NBI-LI-85B	-150" to +60" -150" to +60"	2	A,B,C
Reactor Pressure	RFC-PI-90A RFC-PI-90B	0 - 1200 psig 0 - 1200 psig	2	A,B,C
Drywell Pressure	PC-PI-512A PC-PR-512B	0 - 80 psia 0 - 80 psia	2	A,B,C
Drywell Temperature	PC-TR-503 PC-TI-505	50 - 170°F 50 - 350°F	2	A,B,C
Suppression Chamber Air Temperature	PC-TR-21A PC-TR-23, Ch 1 & 2	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Temperature	PC-TR-24, Ch 1 to 16	0 - 250°F	4	A,B,C
Suppression Chamber Water Level	PC-LI-10 PC-LR-11	(-4' to +6') (-4' to +6')	2	A,B,C
	PC-LI-12 PC-LI-13	-10" to +10" -10" to +10"	2	A,B,C,E
Suppression Chamber Pressure	PC-PR-20	0 - 2 psig	1	B,C
Control Rod Position	N.A.	Indicating Lights	1	A,B,C,D
Neutron Monitoring	N.A.	S.R.M., I.R.M., LPRM 0 - 100% power	1	A,B,C,D
Torus to Drywell Differential Pressure	PC-dPR-20	0 - 2 psid	1	A,B,C,E
Suppression Chamber/ Drywell Pressure (ΔP)	PC-PR-20/513 (2)	0 - 2 psig	1	

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TABLE 4.2.A (Page 1)

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION SYSTEM

TEST AND CALIBRATION FREQUENCIES

Item	Item I.D. No.	Function Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Water Level	NBI-LIS-57, A & B #2 NBI-LIS-58, A & B #2	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Low Water Level	NBI-LIS-57, A & B #1 NBI-LIS-58, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
Main Steam Line Leak Detection	MS-TE-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	Once/Month (1)	Once/Operating Cycle	None
Main Steam Line High Flow	MS-dPIS-116, A,B,C,&D 117 118 119	Once/Month (1) Once/Month (1) Once/Month (1) Once/Month (1)	Once/3 Months Once/3 Months Once/3 Months Once/3 Months	None None None None
Main Steam Line Low Press.	MS-PS-134, A,B,C,&D	Once/Month (1)	Once/3 Months	None
High Reactor Pressure	RR-PS-128, A & B	Once/Month (1)	Once/3 Months	None
Condenser Low Vacuum	MS-PS-103, A,B,C,&D	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Flow	RWCU-dPIS-170, A & B	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Space Temp.	RWCU-TS-150 A-D, 151, 152, 153, 154, 155, 156, 157, 158, 159, RWCU-TS-81, A,B,E,F RWCU-TS-81 C,D,G,H	Once/Month (1)	Once/Operating Cycle	None

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TABLE 4.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION
TEST AND CALIBRATION FREQUENCIES

Instrument	Instrument I.D. No.	Calibration Frequency	Instrument Check
Reactor Water Level	NBI-LI-85A	Once/6 Months	Each Shift
	NBI-LI-85B	Once/6 Months	Each Shift
Reactor Pressure	RFC-PI-90A	Once/6 Months	Each Shift
	RFC-PI-90B	Once/6 Months	Each Shift
Drywell Pressure	PC-PR-512A	Once/6 Months	Each Shift
	PC-PI-512B	Once/6 Months	Each Shift
Drywell Temperature	PC-TR-503	Once/6 Months	Each Shift
	PC-TI-505	Once/6 Months	Each Shift
Suppression Chamber Air Temperature	PC-TR-21A	Once/6 Months	Each Shift
	PC-TR-23, Ch. 1 & 2	Once/6 Months	Each Shift
Suppression Chamber Water Temperature	PC-TR-24, Ch. 1 to 16	Once/6 Months	Each Shift
Suppression Chamber Water Level	PC-LI-10	Once/6 Months	Each Shift
	PC-LR-11	Once/6 Months	Each Shift
	PC-LI-12	Once/6 Months	Each Shift
	PC-LI-13	Once/6 Months	Each Shift
Suppression Chamber Pressure	PC-PR-20	Once/6 Months	Each Shift
Control Rod Position	N.A.	N.A.	Each Shift
Neutron Monitoring (APRM)	N.A.	Once/Week	Each Shift
Torus to Drywell Differential Pressure	PC-dPR-20	Once/6 Months	Each Shift
Suppression Chamber/ Drywell Pressure (AP)	PC-PR-20/513 (2)	Once/6 Months	Each Shift

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out of service for maintenance, and (2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

A. Primary Containment Isolation Functions

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation, set to trip at 176.5" (+12.5") above the top of the active fuel, closes all isolation valves except those in Groups 1, 4, and 5. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low reactor water level instrumentation is set to trip when reactor water level is 127" (-37") above the top of the active fuel. This trip closes Recirc Sample Valves (Group 7) and initiates the HPCI and RCIC. The low low low reactor water level instrumentation is set to trip when the water level is 19" (-145") above the top of the active fuel. This trip closes Main Steam Line Isolation Valves (Reference 1), Main Steam Drain Valves, and activates the remainder of the CSCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished, and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Paragraph VI.5.3.1 USAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 and 6 isolation valves. For the breaks discussed above, this instrumentation will generally initiate CSCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of all isolation valves except Groups 4 and 5.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case of accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel clad temperatures peak at approximately 1000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section XIV.6.5 USAR.

Temperature monitoring instrumentation is provided in the main steam tunnel and along the steam line in the turbine building to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. See Spec. 3.7 for Valve Group. The setting is 200°F for the main steam leak detection system. For large breaks, the high steam flow instrumentation is a backup to the temp. instrumentation.

High radiation monitors in the main steam tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10CFR100 guidelines are not exceeded for this accident. Reference Section XIV.6.2 USAR.

Pressure instrumentation is provided to close the main steam isolation valves in RUN Mode when the main steam line pressure drops below Specification 2.1.A.6. The Reactor Pressure Vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the RUN Mode is less severe than the loss of feedwater analyzed in Section XIV.5 of the USAR, therefore, closure of the Main Steam Isolation valves for thermal transient protection when not in RUN mode is not required.

The Reactor Water Cleanup System high flow and temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that core uncovery is prevented and fission product release is within limits.

3.5.F (cont'd.)

3. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

4. When irradiated fuel is in the reactor vessel and the reactor is in the Cold Shutdown Condition, both core spray systems, the LPCI and containment cooling subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel. Refueling requirements are as specified in Specification 3.5.F.6.

5. With irradiated fuel in the reactor vessel, one control rod drive housing may be open while the suppression chamber is completely drained provided that:

   a. The reactor vessel head is removed.

   b. The spent fuel pool gates are open and the fuel pool water level is maintained at a level $\geq$ 33 feet.

   c. The condensate transfer system is operable and a minimum of 230,000 gallons of water is in the condensate storage tank.

   d. The automatic mode of the drywell sump pump is disabled.

   e. No maintainance is being conducted which will prevent filling the suppression chamber to a level above the core spray and LPCI suctions.

   f. With the exception of the suppression chamber water supply, both core spray systems and the LPCI system are operable.

   g. The control rod is withdrawn to the backseat.

4.5.F (cont'd.)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.F (cont'd)	4.5.F (cont'd)

h. A special flange, capable of sealing a leaking control rod housing, is available for immediate use.

i. The control rod housing is covered with the special flange following the removal of the control rod drive.

j. No work is being performed in the vessel while the housing is open.

6. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days. Refueling is permitted with the suppression chamber drained provided an operable core spray or LPCI system is aligned to take a suction on the condensate storage tank containing at least 150,000 gallons ($\geq$14 ft. indicated level).

7. The LPCI System is required to be operable while performing training startups at atmospheric pressure at power levels less than 1% of rated thermal power with the exception that the RHR system may be aligned in the shutdown cooling mode rather than the LPCI mode.

G. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystems, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray subsystems, LPCI subsystems, HPCI and RCIC are filled:

1. Whenever the Core Spray, LPCI, HPCI or RCIC systems are made operable, the discharge piping shall be vented from the high point of the system and water flow observed initially and on a monthly basis.

2. The pressure switches which monitor the LPCI, core spray, HPCI and RCIC lines to ensure they are full shall be functionally tested and calibrated every three months.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6 (cont'd.)

B. Coolant Chemistry

1. The reactor coolant radioactivity concentration shall be maintained within the following limits:

a. Whenever the reactor is critical, the reactor coolant activity shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131.

b. The limit of 3.6.B.1.a above may be exceeded by a factor of 10 or less for a maximum of 48 hours following power transients. The reactor shall not be operated more than 5% of its annual power operation under this exception.

c. If the iodine concentration in the coolant exceeds the equilibrium limit by a factor greater than 10, the reactor shall be shutdown in an orderly manner and in the cold shutdown condition within 24 hours, and the steam line isolation valves shall be closed.

4.6 (cont'd.)

B. Coolant Chemistry

1.a. A sample of reactor coolant shall be collected and analyzed for gross gamma activity as follows:

1. At least every 96 hours whenever the reactor is critical.
2. Prior to reactor startup.
3. In the STARTUP mode, at 4-hour intervals following a power change exceeding 5% of rated power in one hour or less.
4. In the RUN mode, at 4-hour intervals following a power change exceeding 20% of rated power in one hour or less.
5. At 4-hour intervals following an off-gas activity increase of 10,000 μCi/sec measured at the SJAE.
6. At 4-hour intervals whenever measurements indicate the equilibrium iodine concentration limit of 3.6.B.1 is exceeded, until a stable value below the equilibrium limit is established.

The samples required in 4.6.B.1.a.3, 4, and 5 shall be collected for 48 hours but may be discontinued if the reactor coolant concentration is shown to be less than 1% of the equilibrium value specified in 3.6.B.1 or when a stable iodine concentration below the limiting equilibrium value is established. Whereas a single measurement may be used to show an activity level below 1%, at least 3 consecutive samples with the last 2 yielding activities below the equilibrium value are required to establish a stable concentration below the equilibrium limit.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.6.B. (cont'd)

2. Prior to startup and during the operation of the reactor up to 10% of rated power, and during hot standby, the reactor coolant shall not exceed the following limits:

a. Conductivity < 5 µmho/cm at 25°C

b. Chloride 0.1 ppm

The reactor shall be shut down if pH is <5.6 or >8.6 for a 24-hour period.

3. During reactor operation in excess of 10% of rated power, the reactor coolant shall not exceed the following limits:

a. Conductivity 1 µmho/cm at 25°C

b. Chloride 0.2 ppm

4. During the reactor operation in excess of 10% of rated power, the reactor coolant may exceed the limits of Paragraph 3.6.B.3 only for the time limits specified here. If these time limits or the following maximum limits are exceeded, the reactor shall be shutdown and placed in the Cold Shutdown Condition.

a. Conductivity Time above 1 µmho/cm at 25°C, 2 weeks/year Maximum limit-10 µmho/cm at 25°C

b. Chloride Time above 0.2 ppm, 2 weeks/year Maximum limit-0.5 ppm

The reactor shall be shut down if pH is <5.6 or >8.6 for a 24-hour period.

5. When the reactor is not pressurized (i.e. at or below 212°F), reactor coolant shall be maintained below the following limits:

a. Conductivity 10µmho/cm at 25°C

b. Chloride 0.5 ppm

4.6.B.1 (cont'd)

b. If the gross activity counts of a sample indicate an activity concentration above 3.1 µCi/gm of dose equivalent I-131, an isotopic analysis shall be performed and quantitative measurements made to determine the dose equivalent I-131 concentration.

c. An isotopic analysis of a reactor coolant sample shall be made at least once per month.

2. Reactor coolant shall be continuously monitored for conductivity.

3. Prior to startup, during the operation of the reactor and during hot standby, a sample of the reactor coolant shall be analyzed:

a. At least every 80 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is ≤0.7 µmho/cm at 25°C.

b. At least every 24 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is >0.7 but ≤2.0 µmho/cm at 25°C.

c. At least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor reading is >2 but ≤3.5 µmho/cm at 25°C.

d. At least every 4 hours for conductivity, chloride ion content, and pH, when the continuous conductivity monitor reading is >3.5 µmho/cm at 25°C or when the continuous conductivity monitor is inoperable.

4. When the reactor is not pressurized, a sample of the reactor coolant shall be analyzed at least every 80 hours for conductivity and chloride ion content.

Table 3.6.3
INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

Snubber	Location
RF-SNUB-(RF-S17)	DW-921
RF-SNUB-(RF-S18)	DW-921
RF-SNUB-(RF-S19)	DW-921
RF-SNUB-(RF-S8)	DW-921
RF-SNUB-(RF-S9)	DW-921
RHR-SNUB-(RH-S10)	DW-901
RHR-SNUB-(RH-S11)	DW-901
RHR-SNUB-(RH-S13)	DW-921
RHR-SNUB-(RH-S14)	DW-921
RHR-SNUB-(RH-14A)	DW-901
RHR-SNUB-(RH-S15)	DW-921
RHR-SNUB-(RH-S16)	DW-901
RHR-SNUB-(RH-S17)	DW-901
RHR-SNUB-(RH-S18)	DW-901
RHR-SNUB-(RH-S19)	DW-901
RHR-SNUB-(RH-S3)	DW-FLG AREA
RHR-SNUB-(RH-S4)	DW-FLG AREA
RHR-SNUB-(RH-S5)	DW-921
RHR-SNUB-(RH-S6)	DW-921
RHR-SNUB-(RH-S67)	DW-901
RHR-SNUB-(RH-S68)	DW-901
RHR-SNUB-(RH-S69A)	DW-901
RHR-SNUB-(RH-S69B)	DW-901
RHR-SNUB-(RH-S7)	DW-921
RHR-SNUB-(RH-S70)	DW-901
RHR-SNUB-(RH-S71)	DW-901
RHR-SNUB-(RH-S72)	DW-901
RHR-SNUB-(RH-S72A)	DW-901
RHR-SNUB-(RH-S73)	DW-901
RHR-SNUB-(RH-S8A)	DW-901
RHR-SNUB-(RH-S8B)	DW-901
RHR-SNUB-(RH-S8C)	DW-901
RHR-SNUB-(RH-S9)	DW-901
RR-SNUB-(SS-1A)	DW-888
RR-SNUB-(SS-1B)	DW-888
RR-SNUB-(SS-2A)	DW-888
RR-SNUB-(SS-2B)	DW-888
RR-SNUB-(SS-3A1)	DW-901
RR-SNUB-(SS-3A2)	DW-901
RR-SNUB-(SS-3B1)	DW-901
RR-SNUB-(SS-3B2)	DW-901
RR-SNUB-(SS-4A)	DW-901
RR-SNUB-(SS-4B)	DW-901
RR-SNUB-(SS-5A)	DW-888
RR-SNUB-(SS-5B)	DW-888
RR-SNUB-(SS-8A1)	DW-901
RR-SNUB-(SS-8A2)	DW-901
RWCU-SNUB-(CU-S3A)	DW-921
RWCU-SNUB-(CU-S3B)	DW-921

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3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Suppression Pool

At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2. and 3.5.F.5.

a. Minimum water volume - 87,650 ft^3

b. Maximum water volume - 91,000 ft^3

c. Maximum suppression pool temperature during normal power operation - 95°F.

d. During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in c. above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in c. above within 24 hours.

e. The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in c. above.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool

a. The suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3.7.A (cont'd.)

4.7.A.2.b. (cont'd.)

where

P_a = peak accident pressure, 58 psig

P_t = appropriately measured test pressures (psig)

for $\frac{L_{tm}}{L_{am}} > 0.7$

c. The ILRT's shall be performed at the following minimum frequency:

1. Prior to initial unit operation.

2. At approximately three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to eight months if necessary to coincide with refueling outage.

d. The measured leakage rates, L_{tm} and L_{am}, shall be less than 0.75 L_t and 0.75 L_a for the reduced pressure tests and peak pressure test respectively.

e. Except for the initial ILRT, all ILRT's shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test. If an ILRT has to be terminated due to excessive leakage through identified leakage paths, the leakage through such paths shall be determined by a local leakage test and recorded. After repairs are made another ILRT shall be conducted.

If an ILRT is completed but the acceptance criteria of Specification 4.7.A.2.d is not satisfied and repairs are necessary, the ILRT need not be

3.7.A (Cont'd)

4.7.A.2.f (cont'd)

4. Main steam line and feedwater line expansion bellows as specified in Table 3.7.3 shall be tested by pressurizing between the laminations of the bellows at a pressure of 5 psig. This is an exemption to Appendix J of 10CFR50.

5. The personnel airlock shall be tested at 58 psig at intervals no longer than six months. This testing may be extended to the next refueling outage (not to exceed 24 months) provided that there have been no airlock openings since the last successful test at 58 psig. In the event the personnel airlock is not opened between refueling outages, it shall be leak checked at 3 psig at intervals no longer than six months. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, test the personnel airlock at 3 psig. This is an exemption to Appendix J of 10CFR50.

g. Continuous Leak Rate Monitor

When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

h. Drywell Surfaces

The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of torus corrosion or leakage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7. (cont'd.)

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both standby gas treatment systems shall be operable at all times when secondary containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$99% radioactive methyl iodide removal at a velocity within 20 percent of actual system design, $\geq$1.75 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H. and $\leq$30°C.

c. Fans shall be shown to operate within $\pm$10% design flow.

3. From and after the date that one standby gas treatment system is made or found to be inoperable for any reason, reactor operation or fuel handling is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the other standby gas treatment system, and its associated diesel generator, shall be operable.

4.7 (cont'd.)

B. Standby Gas Treatment System

1. At least once per operating cycle the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate.

b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H.

2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each system shall be operated with the heaters on at least 10 hours every month.

e. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in conformance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

3. System drains where present shall be inspected quarterly for adequate water level in loop-seals.

LIMITING CONDITIONS FOR OPERATION

3.7.C (cont'd.)

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. Irradiated fuel is not being handled in the secondary containment.

e. If secondary containment integrity cannot be maintained, restore secondary containment integrity within 4 hours or;

   a. Be in at least Hot Shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

   b. Suspend irradiated fuel handling operations in the secondary containment and all core alterations and activities which could reduce the shutdown margin. The provisions of Specification 1.0.J are not applicable.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

SURVEILLANCE REQUIREMENTS

4.7.C (cont'd.)

a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$ mph) conditions with a filter train flow rate of not more than 100% of building volume per day. ($\bar{\mu}$= wind speed)

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$ mph) conditions with a filter train flow rate of not more than 100% of building volume per day, shall be demonstrated at each refueling outage prior to refueling.

d. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4 inch of water negative pressure under calm wind conditions.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

LIMITING CONDITIONS FOR OPERATION

3.7 (cont'd.)

E. Drywell-Suppression Chamber Differential Pressure

1. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.0 psid except as specified in a, b, and c below.

   a. This differential shall be established within 26 hours after placing the mode switch in run.

   b. This differential may be decreased to less than 1.0 psid 24 hours prior to placing mode switch in refuel or shutdown.

   c. This differential may be decreased to less than 1.0 psid for a maximum of four (4) hours during required operability testing of the HPCI system pump, the RCIC system pump and the drywell-pressure suppression chamber vacuum breakers.

2. If the differential pressure of specification 3.7.E.1 cannot be maintained, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in Hot Standby in six (6) hours and in a Cold Shutdown condition within the following 18 hours.

3. The specifications of 3.7.E.1 and 3.7.E.2 are not applicable during a 48 hour continuous period between the dates of March 22, 1982 and March 25, 1982.

SURVEILLANCE REQUIREMENTS

4.7 (cont'd.)

E. Drywell-Suppression Chamber Differential Pressure

1. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

TABLE 3.7.4
PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-7A	MS-AO-80A and MS-AO-86A, Main Steam Isolation Valves	Air
X-7B	MS-AO-80B and MS-AO-86B, Main Steam Isolation Valves	Air
X-7C	MS-AO-80C and MS-AO-86C, Main Steam Isolation Valves	Air
X-7D	MS-AO-80D and MS-AO-86D, Main Steam Isolation valves	Air
X-8	MS-MO-74 and MS-MO-77, Main Steam Line Drain	Air
X-9A	RF-15CV and RF-16CV, Feedwater Check Valves	Air
X-9A	RCIC-AO-22, RCIC-MO-17, and RWCU-15CV, RCIC/RWCU Connection to Feedwater	Air
X-9B	RF-13CV and RF-14CV, Feedwater Check Valves	Air
X-9B	HPCI-AO-18 and HPCI-MO-57, HPCI Connection to Feedwater	Air
X-10	RCIC-MO-15 and RCIC-MO-16, RCIC Steam Line	Air
X-11	HPCI-MO-15 and HPCI-MO-16, HPCI Steam Line	Air
X-12	RHR-MO-17 and RHR-MO-18, RHR Suction Cooling	Air
X-13A	RHR-MO-25A and RHR-MO-27A, RHR Supply to RPV	Air
X-13B	RHR-MO-25B and RHR-MO-27B, RHR Supply to RPV	Air
X-14	RWCU-MO-15 and RWCU-MO-18, Inlet to RWCU System	Air
X-16A	CS-MO-11A and CS-MO-12A, Core Spray to RPV	Air
X-16B	CS-MO-11B and CS-MO-12B, Core Spray to RPV	Air
X-17	RHR-MO-32 and RHR-MO-33, RPV Head Spray	Air
X-18	RW-732AV and RW-733AV, Drywell Equipment Sump Discharge	Air
X-19	RW-765AV and RW-766AV, Drywell Floor Drain Sump Discharge	Air
X-25	PC-232MV and PC-238AV, Purge and Vent Supply to Drywell	Air
X-25	ACAD-1305MV and ACAD-1306MV, Supply to Drywell	Air
X-26	PC-231MV and PC-246AV, Purge and Vent Exhaust from Drywell	Air
X-26	ACAD-1310MV, Bleed from Drywell	Air

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High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither system is operable, the plant is brought to a condition where the standby gas treatment system is not required.

4.7.B & 4.7.C BASES

Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with ANSI N510-1980. The test cannisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

1. The reactor shall not be made critical from a Cold Shutdown Condition unless all of the following conditions are satisfied:

a. Both off-site sources (345 KV and 69 KV) and the startup transformer and emergency transformer are available and capable of automatically supplying power to the 4160 Volt emergency buses 1F and 1G.

b. Both diesel generators shall be operable and there shall be a minimum of 45,000 gal. of diesel fuel in the fuel oil storage tanks.

c. The 4160V critical buses 1F and 1G and the 480V critical buses 1F and 1G are energized.

   1. The loss of voltage relays and their auxiliary relays are operable.

   2. The undervoltage relays and their auxiliary relays are operable.

d. The four unit 125V/250V batteries and their chargers shall be operable.

e. The power monitoring system for the inservice RPS MG set or alternate source shall be operable.

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment

1. Emergency Buses Undervoltage Relays

a. Loss of voltage relays

Once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators.

b. Undervoltage relays

Once every 18 months, low voltage on emergency buses is simulated to demonstrate disconnection of the emergency buses from the offsite power source. The undervoltage relays shall be calibrated once every 18 months.

2. Diesel Generators

a. Each diesel-generator shall be start manually and loaded to not less than 35% of rated load for no less than 2 hours once each month to demonstrate operational readiness.

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A	4.9.A.2 (cont'd)
2. At least one diesel generator shall be operable during fuel handling operations. This one diesel shall be capable of supplying power to an operable Standby Gas Treatment System.	During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps and fuel oil day tank level switches shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.
	b. Once every 18 months the condition under which the diesel generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.
	c. Specification 4.9.A.2.c deleted.
	d. Once a month the quantity of diesel fuel available shall be logged.
	e. Every three months and upon delivery a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 for Nos. 1D or 2D and logged.
	f. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.
	3. Unit Batteries
	a. Every week the specific gravity, the voltage and temperature of the pilot

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.9.A

4.9.A (cont'd.)

cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of the voltage of each cell to nearest 0.1 Volt, specific gravity of each cell, and temperature of every sixth cell. These measurements shall be logged.

c. Once each operating cycle, the stated batteries shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be determined after the discharge and logged.

B. Operation with Inoperable Equipment

1. Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.A.1, except as specified in 3.9.B.1.

a. Incoming Power

1. From and after the date incoming power is not available from a startup or emergency transformer, continued reactor operation is permissible under this condition for seven days. At the end of this period, provided the second source of incoming power has not been made immediately available, the NRC must be notified of the event and the plan to restore this second source. During this period, the two diesel generators and associated critical buses must be demonstrated to be operable.

2. From and after the date that incoming power is not available from both start-up and emergency transformers (i.e., both failed), continued operation is permissible, provided the two diesel generators and associated critical buses are demonstrated to be

4. Power Monitoring System for RPS System

The above specified RPS power monitoring system instrumentation shall be determined operable:

a. At least once per operating cycle by demonstrating the operability of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a channel calibration including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.

1. Over-voltage $\leq$ 132 VAC, with time delay $\leq$ 2 sec.

2. Under-voltage $\geq$ 108 VAC, with time delay $\leq$ 2 sec.

3. Under-frequency $\geq$ 57 Hz. with time delay $\leq$ 2 sec.

3.9.B (cont'd.)

operable, all core and containment cooling systems are operable, reactor power level is reduced to 25% of the rated and NRC is notified within 24 hours of the situation, the precautions to be taken during this period and the plans for prompt restoration of incoming power.

b. Diesel Generators

1. From and after the date that one of the diesel generators or an associated critical bus is made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F.1 if Specification 3.9.A.1 is satisfied.

2. From and after the date that both diesel generators are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 24 hours in accordance with Specification 3.5.F.2 if Specification 3.9.A.1 is satisified.

3. From and after the date that one of the diesel generators or associated critical buses and either the emergency or startup transformer power source are made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F.1, provided the other off-site source, startup transformer or emergency transformer is available and capable of automatically supplying power to the 4160V critical buses and the NRC is notified within 24 hours of the occurrence and the plans for restoration of the inoperable components.

4.9.B

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.B.5 (cont'd.)	4.9.B

c. DC Power

1. From and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding ten days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specifications 3.5.A.5 and 3.5.F are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed components to an operable state.

d. RPS/MG Sets

1. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A Refueling Interlocks	4.10.A (Cont'd)
3. The fuel grapple hoist load switch shall be set at ≤ 650 lbs.	3. Whenever the reactor is in the refuel mode and rod block interlocks are being bypassed (for spiral core unloading), one licensed operator and one member of the reactor engineering staff will verify that all fuel has been removed before the corresponding control rod is withdrawn.
4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at ≤ 400 lbs.	4. Following the withdrawal and bypassing of a control rod, two licensed operator will verify that the interlock bypassed is on the correct control rod.
5. A maximum of two nonadjacent control rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied;	5. Prior to loading fuel in a control cell (using the spiral reload technique), the control room operator and a license operator and a member of the reactor engineering staff on the refueling floo shall verify that the control rod is inserted in the cell to be loaded.
a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.	
b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.	
c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.	
d. An appropriate number of SRM's are available as defined in Specification 3.10.B.	

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. The maintenance is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated or that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that the core is shutdown by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38% Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.6 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Prior to removal of the last two diagonal fuel assemblies, a double blade guide shall be inserted to properly support the control rod and fuel assemblies. After removal of the last two fuel assemblies and withdrawal of the control rod, the double blade guide may be removed.

Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core. The requirements for SRM operability during these core alterations assure sufficient core monitoring.

To minimize the possibility of loading fuel into a cell containing no control rod, when refueling interlock input signals are bypassed for the spiral unload/reload technique, it is required that the control room operator and a licensed operator and a member of the reactor engineering staff on the refueling floor verify that the control rod is inserted in the cell to be loaded. Prior to insertion of the control rod, it shall be verified that a double blade guide was placed in the cell to be loaded to properly support the control rod and fuel assemblies.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.12 Additional Safety Related Plant Capabilities	4.12 Additional Safety Related Plant Capabilities
Applicability:	Applicability:
Applies to the operating status of the main control room ventilation system, the reactor building closed cooling water system and the service water system.	Applies to the surveillance requirements for the main control room ventilation system, the reactor building closed cooling water system and the service water system which are required by the corresponding Limiting Conditions for Operation.
Objective:	Objective:
To assure the availability of the main control room ventilation system, the reactor building closed cooling water system and the service water system upon the conditions for which the capability is an essential response to station abnormalities.	To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.
A. Main Control Room Ventilation	A. Main Control Room Ventilation
1. Except as specified in Specification 3.12.A.3 below, the control room air treatment system, the diesel generators required for operation of this system and the main control room air radiation monitor shall be operable at all times when containment integrity is required.	1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal absorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate.
2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal absorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.	2.a. The tests and sample analysis of Specification 3.12.A.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show $\geq$ 99% radioactive methyl iodide removal at a velocity within 20% of system design, $\geq$1.75 mg/m^3 inlet iodide concentration, $\geq$ 95% R.H. and $\leq$30°C.	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. Fans shall be shown to operate within $\pm$ 10% design flow.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

3.12 BASES

A. Main Control Room Ventilation System

The control room ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours, or refueling operations are terminated.

B. Reactor Building Closed Cooling Water System

The reactor building closed cooling water system has two pumps and one heat exchanger in each of two loops. Each loop is capable of supplying the cooling requirements of the essential services following design accident conditions with only one pump in either loop.

The system has additional flexibility provided by the capability of interconnection of the two loops and the backup water supply to the critical loop by the service water system. This flexibility and the need for only one pump in one loop to meet the design accident requirements justifies the 30 day repair time during normal operation and the reduced requirements during head-off operations requiring the availability of LPCI or the core spray systems.

C. Service Water System

The service water system consists of four vertical service water pumps located in the intake structure, and associated strainers, piping, valving and instrumentation. The pumps discharge to a common header from which independent piping supplies two Seismic Class I cooling water loops and one turbine building loop. Automatic valving is provided to shutoff all supply to the turbine building loop on drop in header pressure thus assuring supply to the Seismic Class I loops each of which feeds one diesel generator, two RHR service water booster pumps, one control room basement fan coil unit and one RBCCW

LIMITING CONDITIONS FOR OPERATION

3.14 FIRE DETECTION SYSTEM

APPLICABILITY

Applies to the operational status of the Fire Detection System.

OBJECTIVE

To assure continuous automatic surveillance throughout the Main Plant.

SPECIFICATIONS

A. The Fire Detection System instrumentation for each fire detection zone shown in Table 3.14 shall be operable.

B. With one or more of the fire detection instrument(s) shown in Table 3.14 inoperable:

1. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

2. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

3.15 FIRE SUPPRESSION WATER SYSTEM

APPLICABILITY

Applies to the availability of water for fire fighting purposes.

OBJECTIVE

To assure a continuous operable water supply for fire fighting systems from 2 fire pumps.

SURVEILLANCE REQUIREMENTS

4.14 FIRE DETECTION SYSTEM

APPLICABILITY

Applies to the operational status of the Fire Detection System.

SPECIFICATIONS

A. Each detector on Table 3.14 shall be demonstrated operable every 6 months by performance of a channel functional test.

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

4.15 FIRE SUPPRESSION WATER SYSTEM

APPLICABILITY

Applies to the availability of water for fire fighting purposes.