



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated November 3, 1998, as supplemented by letter dated December 15, 1998, 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.

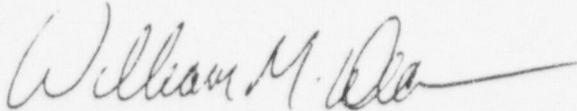
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 164, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



William M. Dean, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: January 5, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 164

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

<u>Remove</u>	<u>Insert</u>	<u>Remove</u>	<u>Insert</u>
ii	ii	63	63
17	17	---	71a
20	20	74	74
21	21	81a	81a
22	22	88	88
23	23	90	90
25	25	92	92
27	27	93	93
29	29	105	105
33	33	106	106
34	34	107	107
35	35	108	108
36	36	118	118
38	38	119	119
39	39	140	140
40	40	142	142
41	41	142a	142a
42	42	166a	166a
43	43	168	168
44	44	193	193
45	45	195	195
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metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the ECCS initiation setpoint would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists.

G. Main Steam Line Isolation Valve Closure Scram

The isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram setpoint at 10% of valve closure, there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

- A. Plant operation at any power level shall be permitted in accordance with Table 3.1.1. The system response time from the opening of the sensor contact up to and including the opening of the scram solenoid relay shall not exceed 50 milliseconds.
- B. During operation with the ratio of MFLPD to FRP greater than 1.0 either:
 - a. The APRM System gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1 and 2.1.B or
 - b. The power distribution shall be changed to reduce the ratio of MFLPD to FRP.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

- A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively
- B. Once a day during reactor power operation the maximum fraction of limiting power density and fraction of rated power shall be determined and the APRM system gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1.a and 2.1.B.

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TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	<u>Trip Function</u>	<u>Trip Settings</u>	<u>Modes in Which Functions Must be Operating</u>			<u>Minimum Number Operating Instrument Channels Per Trip System (2)</u>	<u>Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)</u>
			<u>Refuel (1)</u>	<u>Startup (12)</u>	<u>Run</u>		
1.	Mode Switch in Shutdown (5A-S1)		X	X	X	1	A
2.	Manual Scram (5A-S3A/B)		X	X	X	1	A
3.	IRM (7-41(A-F))						
	High Flux	<u><120/125</u>	X	X	X(11)	2	A
	INOP		X	X	X(11)	2	A
4.	APRM (APRM A-F)						
	High Flux (flow bias)	<u><0.66 (W-ΔW)+54%</u> (4)			X	2	A or B
	High Flux (reduced)	<u><15%</u>	X	X		2	A
	INOP				X	2(5)	A or B
	Downscale	<u>>2/125</u>			X	2	A or B
5.	High Reactor Pressure (PT-2-3-55(A-D) (M))	<u><1055 psig</u>	X	X	X	2	A
6.	High Drywell Pressure (PT-5-12(A-D) (M))	<u><2.5 psig</u>	X	X	X	2	A
7.	Reactor Low (6) Water Level (LT-2-3-57A/B(M)) (LT-2-3-58A/B(M))	<u>>127.0 inches</u>	X	X	X	2	A
8.	Scram Discharge Volume High Level (LT-3-231(A-H) (M))	<u><21 gallons</u>	X	X	X	2 (per volume)	A

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TABLE 3.1.1
(Cont'd)REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	<u>Trip Function</u>	<u>Trip Settings And Allowable Deviations</u>	<u>Modes in Which Functions Must be Operating</u>			<u>Minimum Number Operating Instrument Channels Per Trip System (2)</u>	<u>Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)</u>
			<u>Refuel (1)</u>	<u>Startup</u>	<u>Run</u>		
9.	Main steamline high radiation (7) (RM-17-251(A-D))	3x normal background at rated power(8)	X	X	X	2	A or C
10.	Main steamline isolation valve closure (POS-2-80A-A1,B1 POS-2-86A-A1,B1 POS-2-80B-A1,B2 POS-2-86B-A1,B2 POS-2-80C-A2,B1 POS-2-86C-A2,B1 POS-2-80D-A2,B2 POS-2-86D-A2,B2)	<10% valve closure			X	4	A or C
11.	Turbine control valve fast closure (PS-(37-40))	(9) (10)			X	2	A or D
12.	Turbine stop valve closure (SVOS-5-(1-4))	<10% valve(10) closure			X	2	A or D

TABLE 3.1.1 NOTES

1. When the reactor is subcritical 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) high flux IRM or high flux SRM in coincidence
 - d) scram discharge volume high water level
2. Whenever an instrument system is found to be inoperable, the instrument system output relay shall be tripped immediately. Except for MSIV and Turbine Stop Valve Position, this action shall result in tripping the trip system.
3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, 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 level to IRM and place mode switch in the "Startup/Hot Standby" position within eight hours.
 - c) Reduce turbine load and close main steam line isolation valves within 8 hours.
 - d) Reduce reactor power to less than 30% of rated within 8 hours.
4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48×10^6 lbs/hr core flow. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.
5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.
6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.
7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.
8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

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TABLE 4.1.1

SCRAM INSTRUMENTATION AND LOGIC SYSTEMS FUNCTIONAL TESTSMINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION, LOGIC SYSTEMS AND CONTROL CIRCUITS

<u>Instrument Channel</u>	<u>Group⁽³⁾</u>	<u>Functional Test⁽⁷⁾</u>	<u>Minimum Frequency⁽⁴⁾</u>
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm ⁽⁵⁾	Before Each Startup & Weekly During Refueling ⁽⁶⁾
Inoperative	C	Trip Channel and Alarm	Before Each Startup & Weekly During Refueling ⁽⁶⁾
APRM			
High Flux	B	Trip Output Relays ⁽⁵⁾	Once Each Week
High Flux (Reduced)	B	Trip Output Relays ⁽⁵⁾	Before Each Startup & Weekly During Refueling ⁽⁶⁾
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays ⁽⁵⁾	Once Each Week
Flow Bias	B	Trip Output Relays ⁽⁵⁾	(1)
High Reactor Pressure	B	Trip Channel and Alarm ⁽⁵⁾	(1)
High Drywell Pressure	B	Trip Channel and Alarm ⁽⁵⁾	(1)
Low Reactor Water Level ^{(2),(8)}	B	Trip Channel and Alarm ⁽⁵⁾	(1)
High Water Level in Scram Discharge Volume	B	Trip Channel and Alarm ⁽⁵⁾	(1)
High Main Steam Line Radiation ⁽²⁾	B	Trip Channel and Alarm ⁽⁵⁾	Once Each Week
Main Steam Line Iso. Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Con. Valve Fast Closure	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)
Scram Test Switch (5A-S2 (A-D))	A	Trip Channel and Alarm	Each Refueling Outage
First Stage Turbine Pressure - Permissive (PS-5-14 (A-D))	A	Trip Channel and Alarm	Every 6 Months

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TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATIONMINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

<u>Instrument Channel</u>	<u>Group</u> ⁽¹⁾	<u>Calibration Standard</u> ⁽⁴⁾	<u>Minimum Frequency</u> ⁽²⁾
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Output Signal (Reduced)	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
LPRM (LPRM ND-2-1-104(80))	B(5)	Using TIP System	Every 1000 Equivalent Full Power Hours
High Reactor Pressure	B	Standard Pressure Source	Once/Operating Cycle
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	B	Standard Pressure Source	Once/Operating Cycle
High Water Level in Scram Discharge Volume	B	Water Level	Once/Operating Cycle
Low Reactor Water Level	B	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	(6)	Refueling Outage
High Main Steam Line Radiation	B	Appropriate Radiation Source ⁽³⁾	Refueling Outage
First Stage Turbine Pressure Permissive (PS-5-14(A-D))	A	Pressure Source	Every 6 Months and After Refueling
Main Steam Line Isolation Valve Closure	A	(6)	Refueling Outage

BASES:3.1 Reactor Protection System

The reactor protection system automatically initiates a reactor scram to:

1. preserve the integrity of the fuel barrier;
2. preserve the integrity of the primary system barrier; and
3. minimize the energy which must be absorbed, and prevent criticality following a loss of coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance, testing, or calibration.

The reactor protection system is of the dual channel type. The system is made up of two independent logic channels, each having three subsystems of tripping devices. One of the three subsystems has inputs from the manual scram push buttons and the reactor mode switch. Each of the two remaining subsystems has an input from at least one independent sensor monitoring each of the critical parameters. The outputs of these subsystems are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subsystems will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both logic channels is required to produce a reactor scram.

The required conditions when the minimum instrument logic conditions are not met are chosen so as to bring station operation promptly to such a condition that the particular protection instrument is not required; or the station is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions.

When the minimum requirements for the number of operable or operating trip system and instrumentation channels are satisfied, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scrambling the reactor.

Three APRM instrument channels are provided for each protection trip system to provide for high neutron flux protection. APRM's A and E operate contacts in a trip subsystem, and APRM's C and F operate contacts in the other trip subsystem. APRM's B, D, and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration without changing the minimum number of channels required for inputs to each trip system. Additional IRM channels have also been provided to allow bypassing of one such channel. IRM assignment to the bypass switches is described on FSAR Figure 7.5-9 and in FSAR Section 7.5.5.4.

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specification 2.1.

Instrumentation is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

BASES:4.1 REACTOR PROTECTION SYSTEM

- A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B and C. Sensors that make up Group A are the on-off type and will be tested and calibrated at the indicated intervals. Initially the tests are more frequent than Yankee experience indicates necessary. However, by testing more frequently, the confidence level with this instrumentation will increase and testing will provide data to justify extending the test intervals as experience is accrued.

Group B devices utilize an analog sensor followed by an amplifier and bistable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bistable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during start-up and inactive during full-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

- B. The ratio of MFLPD to FRP shall be checked once per day to determine if the APRM gains require adjustment. Because few control rod movements or power changes occur, checking these parameters daily is adequate.

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the operational status of the plant instrumentation systems which initiate and control a protective function.

Objective:

To assure the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

When the system(s) it initiates or controls is required in accordance with Specification 3.5, the instrumentation which initiates the emergency core cooling system(s) shall be operable in accordance with Table 3.2.1.

B. Primary Containment Isolation

When primary containment integrity is required, in accordance with Specification 3.7, the instrumentation that initiates primary containment isolation shall be operable in accordance with Table 3.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

The instrumentation that initiates the isolation of the reactor building ventilation system and the actuation of the standby gas treatment system shall be operable in accordance with Table 3.2.3.

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the surveillance requirements of the instrumentation systems which initiate and control a protective function.

Objective:

To verify the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.1.

B. Primary Containment Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.3.

3.2 LIMITING CONDITIONS FOR OPERATION

D. Off-Gas System Isolation

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

E. Control Rod Block Actuation

During reactor power operation the instrumentation that initiates control rod block shall be operable in accordance with Table 3.2.5.

F. Mechanical Vacuum Pump Isolation

1. Whenever the main steam line isolation valves are open, the mechanical vacuum pump shall be capable of being automatically isolated and secured by a signal of high radiation in the main steam line tunnel or shall be manually isolated and secured.

2. If Specification 3.2.F.1 is not met following a routine surveillance check, the reactor shall be in the cold shutdown within 24 hours.

G. Post-Accident Instrumentation

During reactor power operation, the instrumentation that displays information in the Control Room necessary for the operator to initiate and control the systems used during and following a postulated accident or abnormal operating condition shall be operable in accordance with Table 3.2.6.

4.2 SURVEILLANCE REQUIREMENTS

D. Off-Gas System Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.

E. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5.

F. Mechanical Vacuum Pump Isolation

During each operating cycle, automatic isolation and securing of the mechanical vacuum pump shall be verified while the reactor is shutdown.

G. Post-Accident Instrumentation

The post-accident instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.6.

3.2 LIMITING CONDITIONS FOR OPERATION

H. Drywell to Torus ΔP Instrumentation

1. During reactor power operation, the Drywell to Torus ΔP Instrumentation (recorder #1-156-3 and instrument DPI-1-158-6) shall be operable except as specified in 3.2.H.2.
2. From and after the date that one of the Drywell to Torus ΔP instruments is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless the instrument is sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.

I. Recirculation Pump Trip Instrumentation

During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operable in accordance with Table 3.2.1.

J. Deleted

K. Degraded Grid Protective System

During reactor power operation, the emergency bus undervoltage instrumentation shall be operable in accordance with Table 3.2.8.

4.2 SURVEILLANCE REQUIREMENTS

H. Drywell to Torus ΔP Instrumentation

The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

I. Recirculation Pump Trip Instrumentation

The Recirculation Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.

J. Deleted

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.8.

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TABLE 3.2.1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

<u>Core Spray - A & B (Note 1)</u>			
<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2	High Drywell Pressure (PT-10-101(A-D) (M))	≤ 2.5 psig	Note 2
2	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D) (M))	≥ 82.5 " above top of enriched fuel	Note 2
1	Low Reactor Pressure (PT-2-3-56C/D(M))	$300 \leq P \leq 350$ psig	Note 2
2	Low Reactor Pressure (PT-2-3-56A/B(M) & PT-2-3-52C/D(M))	$300 \leq P \leq 350$ psig	Note 2
1	Time Delay (14A-K16A & B)	≤ 10 seconds	Note 2
2	Pump (P-46-1A/B) Discharge Pressure (PS-14-44(A-D))	≥ 100 psig	Note 5
1	Auxiliary Power Monitor (LNPX C/D)	--	Note 5
1	Pump Bus Power Monitor (27/3A/B, 27/4A/B)	--	Note 5
1	Trip System Logic	--	Note 5

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TABLE 3.2.1
(Cont'd)EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System A & B (Note 1)			
Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
1	Low Reactor Pressure (PT-2-3-56C/D(M))	$300 \leq p \leq 350$ psig	Note 2
2	High Drywell Pressure (PT-10-101(A-D)(M))	≤ 2.5 psig	Note 2
2	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(S1))	≥ 82.5 " above top of enriched fuel	Note 2
1	Time Delay (10A-K51A & B)	0 seconds	Note 5
1	Reactor Vessel Shroud Level (LT-2-3-73A/B(M))	$\geq 2/3$ core height	Note 5
1	Time Delay (10A-K72A & B)	≤ 60 seconds	Note 5
1	Time Delay (10A-K50A & B)	≤ 5 seconds	Note 5
1	Low Reactor Pressure (PS-2-128A & B)	$100 \leq p \leq 150$ psig	Note 2
2 per pump	RHR Pump (A-D) Discharge Pressure (PS-10-105(A-H))	≥ 100 psig	Note 5
2	High Drywell Pressure (PT-10-101(A-D)(S1))	≤ 2.5 psig	Note 2

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TABLE 3.2.1
(Cont'd)EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

<u>Low Pressure Coolant Injection System A & B (Note 1)</u>			
<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
1	Time Delay (10A-K45A & B)	≤ 6 minutes	Note 5
2	Low Reactor Pressure (PT-2-3-56A/B(M) & PT-2-3-52C/D(M))	$300 \leq p \leq 350$ psig	Note 2
1	Auxiliary Power Monitor (LNPX C/D)	--	Note 5
1	Pump Bus Power Monitor (27/3A/B, 27/4A/B)	--	Note 5
1	Trip System Logic	--	Note 5

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TABLE 3.2.1
(Cont'd)EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

High Pressure Coolant Injection System			
<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2 (Note 3)	Low-Low Reactor Vessel Water Level (LT-2-3-72 (A-D) (S1))	Same as LPCI	Note 5
2 (Note 4)	Low Condensate Storage Tank Water Level (LSL-107-5A/B)	$\geq 3\%$	Note 5
2 (Note 3)	High Drywell Pressure (PT-10-101 (A-D) (M))	Same as LPCI	Note 5
1 (Note 3)	Bus Power Monitor (23A-K41)	--	Note 5
1 (Note 4)	Trip System Logic	--	Note 5
2 (Note 7)	High Reactor Vessel Water Level (LT-2-3-72A/B) (S4)	< 177 inches above top of enriched fuel	Note 5

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TABLE 3.2.1
(Cont'd)EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

<u>Automatic Depressurization</u>			
<u>Minimum Number of Operable Instrument Channels per Trip System (Note 4)</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2	Low-Low Reactor Vessel Water Level (LT-2-3-72 (A-D) (M))	Same as Core Spray	Note 6
2	High Drywell Pressure (PT-10-101 (A-D) (S1))	≤2.5 psig	Note 6
1	Time Delay (2E-K5A/B)	≤120 seconds	Note 6
1	Bus Power Monitor (2E-K1A/B)	--	Note 6
1	Trip System Logic	--	Note 6
2	Time Delay (2E-K16A/B, 2E-K17A/B)	≤8 minutes	Note 6

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TABLE 3.2.1
(Cont'd)RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

<u>Recirculation Pump Trip - A & B (Note 1)</u>			
<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2	Low-Low Reactor Vessel Water Level (LM-2-3-68(A-D))	\geq 6' 10.5" above top of enriched fuel	Note 2
2	High Reactor Pressure (PM-2-3-54(A-D))	\leq 1150 psig	Note 2
2	Time Delays (2-3-68(A-D)(X))	\leq 10 seconds	Note 2
1	Trip Systems Logic	--	Note 2

TABLE 3.2.1 NOTES

1. Each of the two Core Spray, LPCI and RPT, subsystems are initiated and controlled by a trip system. The subsystem "B" is identical to the subsystem "A".
2. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits. If the channel cannot be tripped by the means stated above, that channel shall be made operable within 24 hours or an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.
3. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.
4. One trip system with initiating instrumentation arranged in a one-out-of-two logic.
5. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.
6. Any one of the two trip systems will initiate ADS. If the minimum number of operable channels in one trip system is not available, the requirements of Specification 3.5.F.2 and 3.5.F.3 shall apply. If the minimum number of operable channels is not available in both trip systems, Specifications 3.5.F.3 shall apply.
7. One trip system arranged in a two-out-of-two logic.

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TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied (Note 2)</u>
2	Low-Low Reactor Vessel Water Level (LT-2-3-57A/B(S2), LT-2-3-58A/B(S2))	$\geq 82.5"$ above the top of enriched fuel	A
2 of 4 in each of 2 channels	High Main Steam Line Area Temperature (TS-2-(121-124) (A-D))	$\leq 212^{\circ}\text{F}$	B
2/steam line	High Main Steam Line Flow (DPT-2-(116-119) (A-D) (M))	$\leq 140\%$ of rated flow	B
2/(Note 1)	Low Main Steam Line Pressure (PS-2-134 (A-D))	≥ 800 psig	B
2/(Note 6)	High Main Steam Line Flow (DPT-2-116A, 117B, 118C, 119D (S1))	$\leq 40\%$ of rated flow	B
2	Low Reactor Vessel Water Level (LT-2-3-57A/B(M), LT-2-3-58A/B(M))	Same as Reactor Protection System	A
2	High Main Steam Line Radiation (7) (8) (RM-17-251 (A-D))	≤ 3 x background at rated power (9)	B
2	High Drywell Pressure	Same as Reactor Protection System	A
2/(Note 10)	Condenser Low Vacuum	$\leq 12"$ Hg absolute	A
1	Trip System Logic	--	A

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TABLE 3.2.2
(Cont'd)HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2 per set of 4	High Steam Line Space Temperature (TS-23-(101-104) (B-D))	$\leq 212^{\circ}\text{F}$	Note 3
1	High Steam Line d/p (Steam Line Break) (DPIS-23-77/78)	≤ 195 inches of water	Note 3
4 (Note 5)	Low HPCI Steam Supply Pressure (PS-23-68 (A-D))	≥ 70 psig	Note 3
2	Main Steam Line Tunnel Temperature (TS-23-(101-104)A)	$\leq 212^{\circ}\text{F}$	Note 3
1	Time Delay (23A-K48) (23A-K49)	≤ 35 minutes	Note 3
1	Bus Power Monitor (23A-K38)	--	--
1	Trip System Logic	--	--

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TABLE 3.2.2
(Cont'd)REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied (Note 2)</u>
2	Main Steam Line Tunnel Temperature (TS-13-(79-82)A)	$\leq 212^{\circ}\text{F}$	Note 3
1	Time Delay (13A-K41) (13A-K42)	≤ 35 minutes	Note 3
2 per set of 4	High Steam Line Space Temperature (TS-13-(79-82) (B,C,D))	$\leq 212^{\circ}\text{F}$	Note 3
1	High Steam Line d/p (Steam Line Break) (DPIS-13-83/84)	≤ 195 inches of water	Note 3
4 (Note 5)	Low Steam Supply Pressure (PS-13-87 (A-D))	≥ 50 psig	Note 3
1	Bus Power Monitor (13A-K33)	--	Note 3
1	Trip System Logic	--	Note 3
1	Time Delay (13A-K7) (13A-K31)	$3 \leq t \leq 7$ seconds	Note 3

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TABLE 3.2.3

REACTOR BUILDING VENTILATION ISOLATION & STANDBY GAS TREATMENT SYSTEM INITIATION

<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2	Low Reactor Vessel Water Level (LT-2-3-57A/B(M), LT-2-3-58A/B(M))	Same as PCIS	Note 1
2	High Drywell Pressure (PT-5-12(A-D)(M))	Same as PCIS	Note 1
1	Reactor Building Vent (RM-17-452A/B)	<14 mr/hr	Note 1
1	Refueling Floor Zone Radiation (RM-17-453A/B)	<100 mr/hr	Note 1
1	Reactor Building Vent Trip System Logic	--	Note 1
1	Standby Gas Treatment Trip System Logic	--	Note 1
1	Logic Bus Power Monitor (16A-K52/53)	--	Note 1

Note 1 - If the minimum number of operable instrument channels is not available in either trip system for more than 24 hours, the reactor building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

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TABLE 3.2.4

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
1	Time Delay (Stack Off-Gas Valve Isolation) (15TD & 16TD)	< 2 minutes ≤ 30 minutes	Note 1
1	Trip System Logic	--	Note 1

Note 1 - At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

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TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System		Trip Function	Modes in Which Function Must be Operable			Trip Setting
			Refuel	Startup	Run	
(Note 1)	2 2	Startup Range Monitor				
		a. Upscale (Note 2) (7-40(A-D))	X	X		$\leq 5 \times 10^5$ cps (Note 3)
	b. Detector Not Fully Inserted (7-11(A-D) (LS-4))	X	X			
	Intermediate Range Monitor					
	2	a. Upscale (7-41(A-F))	X	X		$\leq 108/125$ Full Scale
	2	b. Downscale (Note 4) (7-41(A-F))	X	X		$\geq 5/125$ Full Scale
	2	c. Detector Not Fully Inserted (7-11(E, F, G, H, J, K) (LS-4))	X	X		
	Average Power Range Monitor (APRM A-F)					
	2	a. Upscale (Flow Bias)			X	$\leq 0.66(W-\Delta W)+42\%$ (Note 5)
	2	b. Downscale			X	$\geq 2/125$ Full Scale
(Note 9)	1 1	Rod Block Monitor (Note 6) (RBM A/B)				
		a. Upscale (Flow Bias) (Note 7)			X	$\leq 0.66(W-\Delta W)+N$ (Note 5)
		b. Downscale (Note 7)			X	$\geq 2/125$ Full Scale
(Note 8)	1 (per volume) 1	Scram Discharge Volume (LT-3-231A/G (S1))	X	X	X	≤ 12 Gallons
		Trip System Logic	X	X	X	

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TABLE 3.2.6

POST-ACCIDENT INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels</u>	<u>Parameter</u>	<u>Type of Indication</u>	<u>Instrument Range</u>
2	Drywell Atmospheric Temperature (Note 1)	Recorder #TR-16-19-45 (Blue) Meter #TI-16-19-30B	0-350°F 0-350°F
2	Containment Pressure (Note 1)	Meter #PI-16-19-12A Meter #PI-16-19-12B	(-15) -(+260) psig (-15) -(+260) psig
2	Torus Pressure (Note 1)	Meter #PI-16-19-36A Meter #PI-16-19-36B	(-15) -(+65) psig (-15) -(+65) psig
2	Torus Water Level (Note 3)	Meter #LI-16-19-12A Meter #LI-16-19-12B	0-25 ft. 0-25 ft.
2	Torus Water Temperature (Note 1)	Meter #TI-16-19-33A Meter #TI-16-19-33C	0-250°F 0-250°F
2	Reactor Pressure (Note 1)	Meter #PI-2-3-56A Meter #PI-2-3-56B	0-1500 psig 0-1500 psig
2	Reactor Vessel Water Level (Note 1)	Meter #LI-2-3-91A Meter #LI-2-3-91B	(-200)-0-(+200) "H ₂ O (-200)-0-(+200) "H ₂ O
2	Torus Air Temperature (Note 1)	Recorder #TR-16-19-45 (Red) Meter #TI-16-19-41	0-350°F 50-300°F
2/valve	Safety/Relief Valve Position From Pressure Switches (Note 4)	Lights RV-2-71(A-D) From PS-2-71-(1-3) (A-D)	Closed - Open

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TABLE 3.2.6
(Cont'd)POST-ACCIDENT INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels</u>	<u>Parameter</u>	<u>Type of Indication</u>	<u>Instrument Range</u>
1/valve	Safety Valve Position From Acoustic Monitor (Note 5)	Meter ZI-2-1A/B	Closed - Open
2	Containment Hydrogen/Oxygen Monitor (Note 1)	Recorder SR-VG-6A (SI) Recorder SR-VG-6B (SII)	0-30% hydrogen 0-25% oxygen
2	Containment High-Range Radiation Monitor (Note 6)	Meter RM-16-19-1A/B	1 R/hr-10 ⁷ R/hr
1	Stack Noble Gas Effluent (Note 7)	Meter RM-17-155	0.1 - 10 ⁷ mR/hr

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TABLE 3.2.7

(Table 3.2.7 was intentionally deleted from the Technical Specifications)

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TABLE 3.2.8

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

<u>Minimum Number of Operable Instruments</u>	<u>Parameter</u>	<u>Trip Setting</u>	<u>Required Action</u>
2 per bus	Degraded Bus Voltage - Voltage (27/3Z, 27/3W, 27/4Z, 27/4W)	3,700 volts \pm 40 volts	Note 1
2 per bus	Degraded Bus Voltage - Time Delay (62/3W, 62/3Z, 62/4W, 62/4Z)	10 seconds \pm 1 second	Note 2

TABLE 3.2.8 NOTES

1. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits within one hour.
2. If the minimum number of operable instrument channels are not available, reactor power operation is permissible for only 7 successive days unless the system is sooner made operable.

TABLE 3.2.9

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

<u>Minimum Number of Operable Instrument Channels per Trip System</u>	<u>Trip Function</u>	<u>Trip Level Setting</u>	<u>Required Action When Minimum Conditions For Operation Are Not Satisfied</u>
2 (Note 1)	Low-Low Reactor Vessel Water Level (LT-2-3-72 (A-D) (M))	>82.5" Above Top of Enriched Fuel	Note 4
2 (Note 2)	Low Condensate Storage Tank Water Level (LT-107-12A/B (M))	>3%	Note 4
2 (Note 3)	High Reactor Vessel Water Level (LT-2-3-72C/D (S2))	<177" Above Top of Enriched Fuel	Note 4
1	Bus Power Monitor (13A-K36)	--	Note 4
1	Trip System Logic	--	Note 4

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TABLE 4.2.1
(Cont'd)MINIMUM TEST AND CALIBRATION FREQUENCIESEMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

<u>Trip Function</u>	<u>Recirculation Pump Trip Actuation System</u>		
	<u>Functional Test (8)</u>	<u>Calibration (8)</u>	<u>Instrument Check</u>
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
High Reactor Pressure	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle	--

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TABLE 4.2.7

(Table 4.2.7 was intentionally deleted from the Technical Specifications)

3.3 LIMITING CONDITIONS FOR OPERATION

With the required shutdown margin not met and the mode switch in the "Refuel" position, immediately suspend Alteration of the Reactor Core except for control rod insertion and fuel assembly removal; immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies; within 1 hour, initiate action to restore the integrity of the Secondary Containment System.

2. Reactivity Margin - Inoperable Control Rods

Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. The control rod directional control valves for inoperable control rods shall be

4.3 SURVEILLANCE REQUIREMENTS

2. Reactivity Margin - Inoperable Control Rods

Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number

3.3 LIMITING CONDITIONS FOR OPERATION

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed $1\% \Delta k/k$. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken if such actions are appropriate.

- F. If Specifications 3.3.B through 3.3.D above are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.3 SURVEILLANCE REQUIREMENTS

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.
3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.
4. Refer to the Vermont Yankee Core Performance Analysis Report.
5. The Source Range Monitor (SRM) system has no scram functions. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-6} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.
6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR less than the fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable during periods when fuel is in the reactor unless:

1. The reactor is in cold shutdown

and

2. Control rods are fully inserted and Specification 3.3.A is met.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A Minimum flow rate of 35 gpm at 1275 psig shall be verified for each pump by recirculating demineralized water to the test tank.
2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

3. Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.
4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor

3.4 LIMITING CONDITIONS FOR OPERATION

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

C. Liquid Poison Tank - Boron Concentration

At all times when the Standby Liquid Control System is required to be operable, the following conditions shall be met:

1. The net volume versus concentration of the sodium pentaborate solution in the standby liquid control tank shall meet the requirements of Figure 3.4.1.

4.4 SURVEILLANCE REQUIREMENTS

vessel is available by pumping demineralized water into the reactor vessel. Both loops shall be tested over the course of two operating cycles.

5. Testing the new trigger assemblies by installing one of the assemblies in the test block and firing it using the installed circuitry. Install the unfired assemblies, taken from the same batch as the fired one, into the explosion valves.

6. Recirculating the borated solution.

B. Operation with Inoperable Components

When a component becomes inoperable, its redundant component shall be or shall have been demonstrated to be operable within 24 hours.

C. Liquid Poison Tank - Boron Concentration

1. The solution volume in the tank and temperature in the tank and suction piping shall be checked at least daily.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that the Alternate Cooling Tower Subsystem or both Station Service Water Subsystems are made or found inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem(s) are made operable, provided that during such seven days all other active components of the other subsystem(s) are operable.
4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. High Pressure Cooling Injection (HPCI) System

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 150 psig and prior to reactor startup from a cold condition:
 - a. The HPCI System shall be operable.
 - b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

4.5 SURVEILLANCE REQUIREMENT

3. When the Alternate Cooling Subsystem or both Station Service Water Subsystems are made or found to be inoperable, the operable subsystem(s) shall have been or shall be demonstrated to be operable within 24 hours.

E. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI System shall be performed as follows:

1. Testing

<u>Item</u>	<u>Frequency</u>
Simulated Automatic Actuation Test	Each re-fueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Depressurization Subsystems, the Core Spray Subsystems, the LPCI Subsystems, and the RCIC System are operable.
3. If the requirements of Specification 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 120 psig within 24 hours.

F. Automatic Depressurization System

1. Except as specified in Specification 3.5.F.2 below, the entire Automatic Depressurization Relief System shall be operable at any time the reactor pressure is above 100 psig and irradiated fuel is in the reactor vessel.
2. From and after the date that one of the four relief valves of the Automatic Depressurization Subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the

4.5 SURVEILLANCE REQUIREMENT

2. When the HPCI Subsystem is made or found to be inoperable, the Automatic Depressurization System shall have been or shall be demonstrated to be operable within 24 hours.

NOTE: Automatic Depressurization System operability shall be demonstrated by performing a functional test of the trip system logic.

F. Automatic Depressurization System

Surveillance of the Automatic Depressurization System shall be performed as follows:

1. Operability testing of the relief valves shall be in accordance with Specification 4.6.E.
2. When one relief valve of the Automatic Pressure Relief Subsystem is made or found to be inoperable, the HPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding seven days unless such a valve is sooner made operable, provided that during such seven days both the remaining Automatic Relief System valves and the HPCI System are operable.

3. If the requirements of Specification 3.5.F cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 100 psig within 24 hours.

G. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in Specification 3.5.G.2 below, the RCIC System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel.
2. From and after the date that the RCIC System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the HPCI System are operable.

4.5 SURVEILLANCE REQUIREMENT

G. Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Testing

<u>Item</u>	<u>Frequency</u>
Simulated automatic actuation test (testing valve operability)	Each re-fueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The RCIC System shall deliver at least 400 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION

3. If the requirements of Specification 3.5.G cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 120 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the emergency diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the LPCI, Core Spray and Containment Cooling Subsystems connecting to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.
2. 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 core and containment cooling functions.
3. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all Core and Containment Cooling Subsystems may be inoperable provided no work is permitted which has the potential for draining the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT

H. Minimum Core and Containment Cooling System Availability

1. When one of the emergency diesel generators is made or found to be inoperable, the remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.

3.6 LIMITING CONDITIONS FOR OPERATION

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in Specification 3.6.B.3:

Conductivity	5umho/cm
Chloride ion	0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 umho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, in the reactor coolant water for the first 24 hours after placing the reactor in the power operating condition.

4.6 SURVEILLANCE REQUIREMENTS

- e. With the radioiodine concentration in the reactor coolant greater than 1.1 microcuries/gram dose equivalent I-131, a sample of reactor coolant shall be taken every 4 hours and analyzed for radioactive iodines of I-131 through I-135, until the specific activity of the reactor coolant is restored below 1.1 microcuries/gram dose equivalent I-131.
2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.
3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

3.6 LIMITING CONDITIONS FOR OPERATION

4. Except as specified in Specification 3.6.B.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hours.

Conductivity 5 uhmo/cm
Chloride ion 0.5 ppm

5. If Specification 3.6.B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

c. Coolant Leakage

- 1.a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.
- b. While in the run mode, reactor coolant leakage into the primary containment from unidentified sources shall not

4.6 SURVEILLANCE REQUIREMENTS

- b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken every four hours and analyzed for conductivity and chloride ion content.

C. Coolant Leakage

1. Reactor coolant system leakage, for the purpose of satisfying Specification 3.6.C.1, shall be checked and logged once per shift, not to exceed 12 hours.

BASES: 3.6 and 4.6 (Cont'd)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. In order to estimate the material properties at the 1/4 and 3/4 T positions in the vessel plate, the shift in RT_{NDT} is determined in accordance with Regulatory Guide 1.99, Revision 2. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NTT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines, shown on Figure 3.6.1, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10CFR Part 50.

B. Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 μCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in Specification 3.8.E.1, or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 1.1 μCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a $X/Q = 3.9 \times 10^{-3} \text{ sec/m}^3$ (Pasquill D and 0.33 m/sec equivalent), and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR Part 100 dose guidelines.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.1.b may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equivalent drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Safety analyses have shown that only three of the four relief valves are required to provide the recommended pressure margin of 25 psi below the safety valve actuation settings as well as compliance with the MCPR safety limit for the limiting anticipated overpressure transient. For the purposes of this limiting condition, a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

The setpoint tolerance value for as-left or refurbished valves is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of set pressure. However, the code allows a larger tolerance value for the as-found condition if the supporting design analyses demonstrate that the applicable acceptance criteria are met. Safety analysis has been performed which shows that with all safety and safety relief valves within $\pm 3\%$ of the specified set pressures in Table 2.2.1 and with one inoperable safety relief valve, the reactor coolant pressure safety limit of 1375 psig and the MCPR safety limit are not exceeded during the limiting overpressure transient.

BASES: 3.6 and 4.6 (Cont'd)

E. Structural Integrity and Operability Testing

A pre-service inspection of the components listed in Table 4.2-3 of the FSAR was conducted after site erection to assure freedom from defects greater than code allowance; in addition, this serves as a reference base for further inspections. Prior to operation, the reactor primary system was free of gross defects. In addition, the facility has been designed such that gross defects should not occur

BASES: 4.7 (Cont'd)

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

BASES: 4.7 (Cont'd)

The maximum allowable test leak rate at the peak accident pressure of 44 psig (La) is 0.80 weight % per day. The maximum allowable test leak rate at the retest pressure of 24 psig (Lt) has been conservatively determined to be 0.59 weight percent per day. This value was verified to be conservative by actual primary containment leak rate measurements at both 44 psig and 24 psig upon completion of the containment structure.

As most leakage and deterioration of integrity is expected to occur through penetrations, especially those with resilient seals, a periodic leak rate test program of such penetration is conducted at the peak accident pressure of 44 psig to insure not only that the leakage remains acceptably low but also that the sealing materials can withstand the accident pressure.

The Primary Containment Leak Rate Testing Program is based on Option B to 10CFR50, Appendix J, for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression Chamber-Reactor Building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. Operability testing is performed in conjunction with Specification 4.6.E. Inspections and calibrations are performed during the refueling outages; this frequency being based on equipment quality, experience, and engineering judgment.

The ten (10) drywell-suppression vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified. Also it is inspected to assure that it closes freely and operates properly.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.12 ft². This is equivalent to one vacuum breaker open by three-eighths of an inch (3/8") as measured at all points around the circumference of the disk or three-fourths of an inch (3/4") as measured at the bottom of the disk when the top of the disk is on the seat. Since these valves open in a manner that is purely neither mode, a conservative allowance of one-half inch (1/2") has been selected as the maximum permissible valve opening. Assuming that permissible valve opening could be evenly divided among all ten vacuum breakers at once, valve open position assumed to indication for an individual valve must be activated less than fifty-thousandths of an inch (0.050") at all points along the seal surface of the disk. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a nonseated valve.

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TABLE 3.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	Minimum Channels Operable	Notes
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor (RM-17-350)	1*	1,4,5
b. Service Water Discharge Monitor (RM-17-351)	1	2,4,5
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Discharge Flow Rate Monitor (FIT-20-485/442)	1*	3,4

* During releases via this pathway.

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TABLE 3.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Notes
1. Steam Jet Air Ejector (SJAE) a. Noble Gas Activity Monitor (RM-17-150A/B)	1	7, 8, 9
2. Augmented Off-Gas System a. Noble Gas Activity Monitor Between the Charcoal Bed System and the Plant Stack (Providing Alarm and Automatic Termination of Release) (RAN-OG-3127, RAN-OG-3128) b. Flow Rate Monitor (FI-OG-2002, FI-OG-2004, FI-OG-2008) c. Hydrogen Monitor (H2AN-OG-2921A/B, H2AN-OG-2922A/B)	1 1 1	2, 5, 6, 7 1, 5, 6 3, 5, 6
3. Plant Stack a. Noble Gas Activity Monitor (RM-17-156, RM-17-157) b. Iodine Sampler Cartridge c. Particulate Sampler Filter d. Sampler Flow Integrator (FI-17-156/157) e. Stack Flow Rate Monitor (FI-108-22)	1 1 1 1 1	5, 7, 10 4, 5 4, 5 1, 5 1, 5

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TABLE 3.9.3
(Cont'd)RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations ^a	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION ^b	<p>40 routine monitoring stations as follows:</p> <p>16 incident response stations (one in each meteorological sector) within a range of 0 to 4 km⁹;</p> <p>16 incident response stations (one in each meteorological sector) within a range of 2 to 8 km⁹;</p> <p>the balance of the stations to be placed in special interest areas and control station areas.</p>	Quarterly.	<p>Gamma dose, at least once per quarter.</p> <p>Incident response TLDs in the outer monitoring locations, de-dose only quarterly unless gaseous release LCO was exceeded in period.</p>

TABLE 3.9.4

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES^(a)

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m ³)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
H-3	2×10^4 ^(b)					
Mn-54	1×10^3		3×10^4			
Fe-59	4×10^2		1×10^4			
Co-58	1×10^3		3×10^4			
Co-60	3×10^2		1×10^4			3×10^3 ^(c)
Zn-65	3×10^2		2×10^4			
Zr-Nb-95	4×10^2					
I-131		0.9		3	1×10^2	
Cs-134	30	10	1×10^3	60	1×10^3	
Cs-137	50	20	2×10^3	70	2×10^3	
Ba-La-140	2×10^2			3×10^2		

- (a) Reporting levels may be averaged over a calendar quarter. When more than one of the radionuclides in Table 3.9.4 are detected in the sampling medium, the unique reporting requirements are not exercised if the following condition holds:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \dots < 1.0$$

When radionuclides other than those in Table 3.9.4 are detected and are the result of plant effluents, the potential annual dose to a member of the public must be less than or equal to the calendar year limits of Specifications 3.8.B, 3.8.E and 3.8.F.

- (b) Reporting level for drinking water pathways. For nondrinking water pathways, a value of 3×10^4 pCi/l may be used.
- (c) Reporting level for individual grab samples taken at North Storm Drain Outfall only.

TABLE 4.9.2 NOTATION

- (1) The Instrument Functional Test shall also demonstrate that automatic isolation of this pathway and the Control Room alarm annunciation occurs if any of the following conditions exists:
 - (a) Instrument indicates measured levels above the alarm setpoint.
 - (b) Circuit failure.
 - (c) Instrument indicates a downscale failure.
 - (d) Instrument controls not set in operate mode.
- (2) The Instrument Functional Test shall also demonstrate that Control Room alarm annunciation occurs when any of the following conditions exist:
 - (a) Instrument indicates measured levels above the alarm setpoint.
 - (b) Circuit failure.
 - (c) Instrument indicates a downscale failure.
 - (d) Instrument controls are not set in operate mode.
- (3) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Institute for Standards and Technology) radioactive source positioned in a reproducible geometry with respect to the sensor. These standards should permit calibrating the system over its normal operating range of rate capabilities.
- (4) The Instrument Calibration shall include the use of standard gas samples (high range and low range) containing suitable concentrations, hydrogen balance nitrogen, for the detection range of interest per Specification 3.8.J.1.

BASES:3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMSA. Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments are to ensure that the alarm will occur prior to exceeding 10 times the concentration limits of Appendix B to 10CFR20.1001-20.2401, Table 2, Column 2, values.

Automatic isolation function is not provided on the liquid radwaste discharge line due to the infrequent nature of batch, discrete volume, liquid discharges (on the order of once per year or less), and the administrative controls provided to ensure that conservative discharge flow rates/dilution flows are set such that the probability of exceeding the above concentration limits are low, and the potential off-site dose consequences are also low.

B. Gaseous Effluent Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments are provided to ensure that the alarm/trip will occur prior to exceeding design bases dose rates identified in 3.8.E.1. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

C. Radiological Environmental Monitoring Program

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of member(s) of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

Ten years of plant operation, including the years prior to the implementation of the Augmented Off-Gas System, have amply demonstrated via routine effluent and environmental reports that plant effluent measurements and modeling of environmental pathways are adequately conservative. In all cases, environmental sample results have been two to three orders of magnitude less than expected by the model employed, thereby representing small percentages of the ALARA and environmental reporting levels. This radiological environmental monitoring program has therefore been significantly modified as provided for by Regulatory Guide 4.1 (C.2.b), Revision 1, April 1975. Specifically, the air particulate and radioiodine air sampling periods have been increased to semimonthly, based on plant effluent and environmental air sampling data for the previous ten years of operation. An I-131 release rate trigger value of 1×10^{-1} uCi/sec from the plant stack will require that air sample collection be increased to weekly. The

3.12 LIMITING CONDITIONS FOR OPERATION

E. Extended Core Maintenance

One or more control rods may be withdrawn or removed from the reactor core provided the following conditions are satisfied:

1. 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 on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.
2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.12.B.
3. If the spiral unload/reload method of core alteration is to be used, the following conditions shall be met:
 - a. Prior to spiral unload and reload, the SRMs shall be proven operable as stated in Specification 3.12.B1 and 3.12.B2. However, during spiral unloading, the count rate may drop below 3 cps.

4.12 SURVEILLANCE REQUIREMENTS

E. Extended Core Maintenance

Prior to control rod withdrawal for extended core maintenance, that control rod's control cell shall be verified to contain no fuel assemblies.

1. This surveillance requirement is the same as that given in Specification 4.12.A.
2. This surveillance requirement is the same as that given in Specification 4.12.B.

BASES:3.12 & 4.12 REFUELING

- A. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the Reactivity Control System and interlocks at any time during shutdown or the following operating cycle.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist.

Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position, only one control rod can be withdrawn.

- B. The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRMs will drop below 3 cps before all the fuel is unloaded. Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRMs will no longer be required. Requiring the SRMs to be operational prior to fuel removal assures that the SRMs are operable and can be relied on even when the count rate may go below 3 cps.

Prior to spiral reload, two diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, will be loaded into their designated core positions next to each of the 4 SRMs to obtain the required 3 cps. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photo fission, and photo disintegration of deuterium in the moderator. This neutron production is normally great enough to meet the 3 cps minimum SRM requirement, thereby providing a means by which SRM response may be demonstrated before the spiral reload begins. During the spiral reload, the fuel will be loaded in the reverse sequence that it was unloaded with the exception of the initial eight (8) fuel assemblies which are loaded next to the SRMs to provide a means of SRM response.

3.13 LIMITING CONDITIONS FOR OPERATION

C. Fire Hose Stations

1. Except as specified in 3.13.C.2 below, all hose stations inside the Reactor Building, Turbine Building, and those inside the Administration building which provided coverage of the Control Room Building shall be operable whenever equipment in the areas protected by the fire hose stations is required to be operable.
2. With one or more of the fire hose stations specified in 3.13.C.1 above inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an operable hose station within one hour.

4.13 SURVEILLANCE REQUIREMENTS

- 1) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and
- 2) The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

C. Fire Hose Stations

1. Each fire hose station shall be verified to be operable:
 - a. At least monthly by visual inspection of the station to assure all equipment is available.
 - b. At least once each 18 months by removing the hose for inspection and replacing degraded coupling gaskets and reracking.
 - c. At least once each year by hydro-statically testing each outside hose at 250 lbs.
 - d. At least once per 3 years by hydro-statically testing inside hose at 150 lbs.

3.13 LIMITING CONDITIONS FOR
OPERATION

4.13 SURVEILLANCE REQUIREMENTS

- c. At least once per
3 years by
performing an air
flow test through
the Recirculation
M.G. Set foam header
and verifying each
foam nozzle is
unobstructed.

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TABLE 3.13.A.1

FIRE DETECTION SENSORS

<u>Sensor Location</u>		<u>Minimum No. of Sensors Required to Be Operable</u>		
		<u>Heat</u>	<u>Flame</u>	<u>Smoke</u>
1.	Cable Spreading Room & Station Battery Room	-	-	23
2.	Switchgear Room (East)	-	-	10
3.	Switchgear Room (West)	-	-	10
4.	Diesel Generator Room (A)	-	-	2
5.	Diesel Generator Room (B)	-	-	2
6.	Intake Structure (Service Water)	1	1	1
7.	Recirc Motor Generator Set Area	3	-	8
8.a	Control Room Zone 1 (Control Room Ceiling)	-	-	14
8.b	Control Room Zone 2 (Control Room Panels)	-	-	18
8.c	Control Room Zone 3 (Control Room Panels)	-	-	25
8.d	Control Room Zone 4 (Control Room Panels)	-	-	10
8.e	Control Room Zone 5 (Exhaust & Supply Ducts)	-	-	2
9.a	Rx Bldg. Corner Rm NW 232	-	-	1
9.b	Rx Bldg. Corner Rm NW 213 (RCIC)	-	-	1
9.c	Rx Bldg. Corner Rm NE 232	-	-	1
9.d	Rx Bldg. Corner Rm NE 213	-	-	1
9.e	Rx Bldg. Corner Rm SE 232	-	-	1
9.f	Rx Bldg. Corner Rm SE 213	-	-	1
9.g	Rx Bldg. Corner Rm SW 232	-	-	1
10.	HPCI Room	-	-	8
11.	Torus area	12	-	16
12.	Rx Bldg. Cable Penetration Area	-	-	7
13.	Refuel Floor	-	-	13
14.	Diesel Oil Day Tank Room (A)	-	1*	1*
15.	Diesel Oil Day Tank Room (B)	-	1*	1*
16.	Turbine Loading Bay (vehicles)	-	3	-

*NOTE: The Diesel Day Tank Rooms require only one detector operable (1 flame or 1 smoke).