

- (3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (4) Pursuant to the Act and 10 CFR Parts 30, 40, 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
- (5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of 3458 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 186, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

- a) Electrically, or
 - b) Hydraulically by closing the drive water and exhaust water isolation valves.
2. If the inoperable control rod(s) is inserted, within 1 hour disarm the associated directional control valves** either:
- a) Electrically, or
 - b) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

- c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.
- d. With one or more scram discharge volume (SDV) vent or drain lines with one valve inoperable, restore the inoperable valve(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours*** and in COLD SHUTDOWN within the following 24 hours.
- e. With one or more SDV vent or drain lines with both valves inoperable, isolate the associated line within 8 hours **** or be in at least HOT SHUTDOWN within the next 12 hours*** and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

- a. Verifying each valve to be open,* and
- b. Cycling each valve through at least one complete cycle of full travel.

* These valves may be closed intermittently for testing under administrative controls.

** May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

*** Separate Action entry is allowed for each SDV vent and drain line.

**** An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.2 When above the preset power level of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

- a. In accordance with the Surveillance Frequency Control Program, and
- b. Within 24 hours from discovery that a control rod is immovable as a result of excessive friction or mechanical interference.

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

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

- a. The scram discharge volume drain and vent valves OPERABLE in accordance with the Surveillance Frequency Control Program, by verifying that the drain and vent valves;
 1. Close within 30 seconds after receipt of a signal for control rods to scram, and
 2. Open when the scram signal is reset.
- b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation in accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS \

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

- a. Declare the control rod(s) with the slow insertion time inoperable, and
- b. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

- a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER with reactor coolant pressure greater than or equal to 950 psig, following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.
- b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either "1" or "2" as follows:
 - 1.a Specifically affected individual control rods shall be scram time tested at zero reactor coolant pressure and the scram insertion time from the fully withdrawn position to notch position 05 shall not exceed 2.0 seconds, and
 - 1.b Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure prior to exceeding 40% of RATED THERMAL POWER.
 2. Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure.
- c. For at least 10% of the control rods, with reactor coolant pressure greater than or equal to 950 psig, on a rotating basis, and in accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

2. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated pressure is greater than or equal to 955 psig unless the control rod is inserted and disarmed or scrambled.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

- a. In accordance with the Surveillance Frequency Control Program that the position of each control rod is indicated,
- b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and
- c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than 1.70, or THERMAL POWER greater than or equal to 90% of rated with MCPR less than 1.40.

ACTION:

- a. With one RBM channel inoperable:
 1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and
 2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.
- b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

- a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.
- b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

REACTIVITY CONTROL SYSTEMS

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of a minimum of the following:

- a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,
- b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

- a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.
- b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.
 2. The available volume of sodium pentaborate solution is at least 3160 gallons.
 3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. In accordance with the Surveillance Frequency Control Program by: |

1. Verifying the continuity of the explosive charge.
2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:

$$\frac{C}{13\% \text{ wt.}} \times \frac{E}{29 \text{ atom \%}} \times \frac{Q}{86 \text{ gpm}} \geq 1$$

where

C = Sodium pentaborate solution (% by weight)

Q = Two pump flowrate, as determined per
surveillance requirement 4.1.5.c.

E = Boron 10 enrichment (atom % Boron 10)

3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230 ± 25 psig is met.
- d. In accordance with the Surveillance Frequency Control Program by: |
1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.
 2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**
- e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) as shown in the CORE OPERATING LIMITS REPORT (COLR). During operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the appropriate reduction factors for power and flow as defined in the COLR.

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

ACTION:

With an APLHGR exceeding the limiting value, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limiting value

- a. In accordance with the Surveillance Frequency Control Program,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

- a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) EOC-RPT inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.
- b. With MCPR less than the applicable MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.
- c. With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

- a. $\tau = 1.0$ prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2a, and during reactor startups prior to control rod scram time tests in accordance with Specification 4.1.3.2.b.1.b, or
- b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit, including application of the MCPR(P) and MCPR(F) factors as determined from the CORE OPERATING LIMITS REPORT.

- a. In accordance with the Surveillance Frequency Control Program,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
- d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) for each fuel type shall not exceed the value in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

- a. In accordance with the Surveillance Frequency Control Program, |
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR. |
- d. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.1.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program, except Table 4.3.1.1-1 Functions 2.a, 2.b, 2.c, 2.d, 2.e and 2.f. Functions 2.a, 2.b, 2.c, 2.d, and 2.f do not require separate LOGIC SYSTEM FUNCTIONAL TESTS. For Function 2.e, tests shall be performed in accordance with the Surveillance Frequency Control Program. LOGIC SYSTEM FUNCTIONAL TEST for Function 2.e includes simulating APRM and OPRM trip conditions at the APRM channel inputs to the voter channel to check all combinations of two tripped inputs to the 2-Out-Of-4 voter logic in the voter channels.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific reactor trip system.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK(n)</u>	<u>CHANNEL FUNCTIONAL TEST(n)</u>	<u>CHANNEL CALIBRATION(a)(n)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. Intermediate Range Monitors:					
a. Neutron Flux - High	(b)	(j)		2 3(i), 4(i), 5(i)	
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)	
2. Average Power Range Monitor(f):					
a. Neutron Flux - Upscale (Setdown)	(b)	(l)		2	
b. Simulated Thermal Power - Upscale		(e)	(d), (g)	1	
c. Neutron Flux - Upscale			(d)	1	
d. Inoperative	N.A.		N.A.	1, 2	
e. 2-Out-Of-4 Voter			N.A.	1, 2	
f. OPRM Upscale		(e)	(c)(g)	1(m)	
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)	
4. Reactor Vessel Water Level- Low, Level 3				1, 2	
5. Main Steam Line Isolation Valve - Closure	N.A.			1	
6. DELETED					
7. Drywell Pressure - High				1, 2	
8. Scram Discharge Volume Water Level - High					
a. Level Transmitter				1, 2, 5(i)	
b. Float Switch	N.A.			1, 2, 5(i)	

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(n)	CHANNEL FUNCTIONAL TEST(n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
- (c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is $\geq 30\%$ and for recirculation drive flow is $< 60\%$.
- (d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq 25\%$ of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
- (e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
- (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
- (g) The less frequent calibration includes the flow input function.
- (h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
- (i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
- (j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
- (k) DELETED
- (l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
- (m) With THERMAL POWER $\geq 25\%$ of RATED THERMAL POWER.
- (n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.2.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operations of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in accordance with the Surveillance Frequency Control Program, where N is the total number of redundant channels in a specific isolation trip system.

** The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. <u>MAIN STEAM LINE ISOLATION</u>					
a. Reactor Vessel Water Level 1) Low, Low, Level 2 2) Low, Low, Low - Level 1				1, 2, 3 1, 2, 3	
b. DELETED					
c. Main Steam Line Pressure - Low				1	
d. Main Steam Line Flow - High				1, 2, 3	
e. Condenser Vacuum - Low				1, 2**, 3**	
f. Outboard MSIV Room Temperature - High				1, 2, 3	
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High				1, 2, 3	
h. Manual Initiation	N.A.		N.A.	1, 2, 3	
2. <u>RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION</u>					
a. Reactor Vessel Water Level## Low - Level 3				1, 2, 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High				1, 2, 3	
c. Manual Initiation	N.A.		N.A.	1, 2, 3	

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>		<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
3.	<u>REACTOR WATER CLEANUP SYSTEM ISOLATION</u>				
a.	RWCS Δ Flow - High				1, 2, 3
b.	RWCS Area Temperature - High				1, 2, 3
c.	RWCS Area Ventilation Δ Temperature - High				1, 2, 3
d.	SLCS Initiation	N.A.		N.A.	1, 2, 3
e.	Reactor Vessel Water Level Low, Low, - Level 2				1, 2, 3
f.	Manual Initiation	N.A.		N.A.	1, 2, 3
4.	<u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u>				
a.	HPCI Steam Line Δ Pressure - High				1, 2, 3
b.	HPCI Steam Supply Pressure, Low				1, 2, 3
c.	HPCI Turbine Exhaust Diaphragm Pressure - High				1, 2, 3
d.	HPCI Equipment Room Temperature - High				1, 2, 3
e.	HPCI Equipment Room Δ Temperature - High				1, 2, 3
f.	HPCI Pipe Routing Area Temperature - High				1, 2, 3
g.	Manual Initiation	N.A.		N.A.	1, 2, 3
h.	HPCI Steam Line Δ Pressure Timer	N.A.			1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
<u>5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>					
a. RCIC Steam Line Δ Pressure - High				1, 2, 3	
b. RCIC Steam Supply Pressure - Low				1, 2, 3	
c. RCIC Turbine Exhaust Diaphragm Pressure - High				1, 2, 3	
d. RCIC Equipment Room Temperature - High				1, 2, 3	
e. RCIC Equipment Room Δ Temperature - High				1, 2, 3	
f. RCIC Pipe Routing Area Temperature - High				1, 2, 3	
g. Manual Initiation	N.A.		N.A.	1, 2, 3	
h. RCIC Steam Line Δ Pressure Timer	N.A.			1, 2, 3	

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
6. <u>PRIMARY CONTAINMENT ISOLATION</u>					
a. Reactor Vessel Water Level					
1) Low, Low - Level 2				1, 2, 3	
2) Low, Low, Low - Level 1				1, 2, 3	
b. Drywell Pressure## - High				1, 2, 3	
c. North Stack Effluent Radiation - High				1, 2, 3	
d. Deleted					
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High				1, 2, 3	
f. Deleted					
g. Deleted					
h. Drywell Pressure - High/ Reactor Pressure - Low				1, 2, 3	
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.			1, 2, 3	
j. Manual Initiation	N.A.		N.A.	1, 2, 3	

TABLE 4.3.2.1-1 (Continued)

TRIP FUNCTION	ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS			OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED	
	CHANNEL CHECK (a)	CHANNEL FUNCTIONAL TEST (a)	CHANNEL CALIBRATION(a)		
7. <u>SECONDARY CONTAINMENT ISOLATION</u>					
a. Reactor Vessel Water Level Low, Low - Level 2				1, 2, 3	
b. Drywell Pressure## - High				1, 2, 3	
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High				*#	
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High				*#	
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High				1, 2, 3	
e. Deleted					
f. Deleted					
g. Reactor Enclosure Manual Initiation	N.A.		N.A.	1, 2, 3	
h. Refueling Area Manual Initiation	N.A.		N.A.	*	

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

*Required when (1) handling RECENTLY IRRADIATED FUEL in the secondary containment, or (2) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1

ACTION:

- a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to Operable status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.
- c. With either ADS trip system subsystem inoperable, restore the inoperable trip system to OPERABLE status within:
 1. 7 days, provided that the HPCI and RCIC systems are OPERABLE.
 2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.3.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific ECCS trip system.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
1. <u>CORE SPRAY SYSTEM</u>				
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3, 4*, 5*
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure - Low				1, 2, 3, 4*, 5*
d. Manual Initiation	N.A.		N.A.	1, 2, 3, 4*, 5*
2. <u>LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM</u>				
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3, 4*, 5*
b. Drywell Pressure - High				1, 2, 3
c. Reactor Vessel Pressure - Low				1, 2, 3
d. Injection Valve Differential Pressure - Low (Permissive)				1, 2, 3, 4*, 5*
e. Manual Initiation	N.A.		N.A.	1, 2, 3, 4*, 5*
3. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM***</u>				
a. Reactor Vessel Water Level - Low Low, Level 2				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. Condensate Storage Tank Level - Low				1, 2, 3
d. Suppression Pool Water Level - High				1, 2, 3
e. Reactor Vessel Water Level - High, Level 8				1, 2, 3
f. Manual Initiation	N.A.		N.A.	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
4. <u>AUTOMATIC DEPRESSURIZATION SYSTEM</u> [#]				
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3
b. Drywell Pressure - High				1, 2, 3
c. ADS Timer	N.A.			1, 2, 3
d. Core Spray Pump Discharge Pressure - High				1, 2, 3
e. RHR LPCI Mode Pump Discharge Pressure - High				1, 2, 3
f. Reactor Vessel Water Level - Low, Level 3				1, 2, 3
g. Manual Initiation	N.A.		N.A.	1, 2, 3
h. ADS Drywell Pressure Bypass Timer	N.A.			1, 2, 3
5. <u>LOSS OF POWER</u>				
a. 4.16 kV Emergency Bus Under- voltage (Loss of Voltage) ^{###}	N.A.		N.A.	1, 2, 3, 4**, 5**
b. 4.16 kV Emergency Bus Under - voltage (Degraded Voltage)				1, 2, 3, 4**, 5**

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

*** Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

Loss of Voltage Relay 127-11X is not field settable.

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

- a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 24 hours.
- c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:
 1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 24 hours, or, if this action will initiate a pump trip, declare the trip system inoperable.
 2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.
- d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.
- e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each of the required ATWS recirculation pump trip system instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each of the required end-of-cycle recirculation pump trip system instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST, including trip system logic testing, and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested in accordance with the Surveillance Frequency Control Program. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE-RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limit.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured in accordance with the Surveillance Frequency Control Program.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

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

ACTION:

- a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each of the required RCIC system actuation instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program. CHANNEL CHECK and CHANNEL CALIBRATION are not required for manual initiation.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

- a. With a control rod block instrumentation channel trip setpoint** less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition, provided at least one other operable channel in the same trip system is monitoring that parameter.

**The APRM Simulated Thermal Power - Upscale Functional Unit need not be declared inoperable upon entering single reactor recirculation loop operation provided that the flow-biased setpoints are adjusted within 6 hours per Specification 3.4.1.1.

TABLE 4.3.6-1
CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(h)</u>	<u>CHANNEL FUNCTIONAL TEST(h)</u>	<u>CHANNEL CALIBRATION(a)(h)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. <u>ROD BLOCK MONITOR</u>					
a. Upscale	N.A.	(c)		1*	
b. Inoperative	N.A.	(c)	N.A.	1*	
c. Downscale	N.A.	(c)		1*	
2. <u>APRM</u>					
a. Simulated Thermal Power- Upscale	N.A.			1	
b. Inoperative	N.A.		N.A.	1, 2	
c. Neutron Flux - Downscale	N.A.			1	
d. Simulated Thermal Power - Upscale (Setdown)	N.A.			2	
e. Recirculation Flow - Upscale	N.A.			1	
f. LPRM Low Count	N.A.			1, 2	
3. <u>SOURCE RANGE MONITORS</u>					
a. Detector not full in	N.A.	(e)	N.A.	2, 5	
b. Upscale	N.A.	(e)		2, 5	
c. Inoperative	N.A.	(e)	N.A.	2, 5	
d. Downscale	N.A.	(e)		2, 5	
4. <u>INTERMEDIATE RANGE MONITORS</u>					
a. Detector not full in	N.A.		N.A.	2, 5**	
b. Upscale	N.A.			2, 5**	
c. Inoperative	N.A.		N.A.	2, 5**	
d. Downscale	N.A.			2, 5**	
5. <u>SCRAM DISCHARGE VOLUME</u>					
a. Water Level - High	N.A.			1, 2, 5**	
6. <u>DELETED</u>					
7. <u>REACTOR MODE SWITCH SHUTDOWN POSITION</u>	N.A.	(g)	N.A.	3, 4	

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) Deleted.
- (c) Includes reactor manual control multiplexing system input.
- * For OPERATIONAL CONDITION of Specification 3.1.4.3.
- ** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
- *** Deleted.
- (d) Deleted.
- (e) The provisions of Specification 4.0.4 are not applicable provided that the surveillance is performed within 12 hours after the IRMs are on Range 2 or below during a shutdown.
- (f) Deleted.
- (g) The provisions of Specification 4.0.4 are not applicable provided that the surveillance is performed within 1 hour after the Reactor Mode Switch has been placed in the shutdown position.
- (h) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

- a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.
- b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the conditions shown in Table 4.3.7.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.1-1.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENTATION</u>	<u>CHANNEL CHECK(c)</u>	<u>CHANNEL FUNCTIONAL TEST (c)</u>	<u>CHANNEL CALIBRATION(c)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED</u>	
1. Main Control Room Normal Fresh Air Supply Radiation Monitor				1, 2, 3, and *	
2. Area Monitors					
a. Criticality Monitors					
1) Spent Fuel Storage Pool				(a)	
b. Control Room Direct Radiation Monitor				At All Times	
3. Reactor Enclosure Cooling Water Radiation Monitor			(b)	At All Times	

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

- (a) With fuel in the spent fuel storage pool.
- (b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.
- (c) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

- a. With the number of OPERABLE remote shutdown system instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.
- b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-1, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK* and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.7.4.2 Each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) in accordance with the Surveillance Frequency Control Program.

* Control is not required to be transferred to perform the CHANNEL CHECK.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.5-1.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	
1. Reactor Vessel Pressure			
2. Reactor Vessel Water Level			
3. Suppression Chamber Water Level			
4. Suppression Chamber Water Temperature			
5. Suppression Chamber Air Temperature			
6. Primary Containment Pressure			
7. Drywell Air Temperature			
8. Deleted			
9. Deleted			
10. Deleted			
11. Primary Containment Post LOCA Radiation Monitors		**	
12. North Stack Wide Range Accident Monitor***			
13. Neutron Flux			

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table. |

**CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

***High range noble gas monitors.

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

- a. In OPERATIONAL CONDITION 2*, three.
- b. In OPERATIONAL CONDITION 3 and 4, two.

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

ACTION:

- a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least three source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.
- b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

- a. Performance of a:
 1. CHANNEL CHECK in accordance with the Surveillance Frequency Control Program:
 - a) In CONDITION 2*, AND
 - b) In CONDITION 3 or 4.
 2. CHANNEL CALIBRATION** in accordance with the Surveillance Frequency Control Program.
- b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.
- c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3.0 cps*** with the detector fully inserted.

*With IRM's on range 2 or below.

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

***May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.

INSTRUMENTATION

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8.1 Two independent chlorine detection system subsystems shall be OPERABLE with their alarm and trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 0.5 ppm

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

- a. With one chlorine detection subsystem inoperable, restore the inoperable detection system to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.
- b. With both chlorine detection subsystems inoperable, within 1 hour initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.

SURVEILLANCE REQUIREMENTS

4.3.7.8.1 Each of the above required chlorine detection system subsystems shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

INSTRUMENTATION

TOXIC GAS DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8.2 Three independent toxic gas detection system subsystems shall be OPERABLE with their alarm setpoints adjusted to actuate at a toxic gas concentration of less than or equal to:

<u>CHEMICAL</u>	<u>MONITOR SET POINT (ppm)</u>
Ammonia	25
Ethylene Oxide	50
Formaldehyde	5
Vinyl Chloride	10
Phosgene	0.4

APPLICABILITY: ALL OPERATIONAL CONDITIONS.

ACTION:

- a. With one toxic gas detection subsystem inoperable, place the inoperable subsystem in the tripped condition within 24 hours.
- b. With two toxic gas detection system subsystems inoperable, place one inoperable subsystem in the tripped condition within 1 hour, restore one inoperable detection subsystem to OPERABLE status within 7 days, or initiate and maintain operation of at least one control room emergency filtration subsystem in the chlorine isolation mode of operation.
- c. With three toxic gas detection subsystems inoperable, within 1 hour initiate and maintain operation of at least one control room emergency filtration subsystem in the chlorine isolation mode of operation.

SURVEILLANCE REQUIREMENTS

4.3.7.8.2 Each of the above required toxic gas detection system subsystems shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

INSTRUMENTATION

OFFGAS GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The offgas monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.2.5 and 3.11.2.6 respectively, are not exceeded.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

- a. With an offgas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.12-1.
- b. With less than the minimum number of offgas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Annual Radioactive Effluent Release Report.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each offgas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.12-1.

TABLE 4.3.7.12-1

OFFGAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK(5)</u>	<u>SOURCE CHECK(5)</u>	<u>CHANNEL CALIBRATION(5)</u>	<u>CHANNEL FUNCTIONAL TEST(5)</u>	<u>MODES IN WHICH SURVEILLANCE IS REQUIRED</u>	
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM						
a. Hydrogen Monitor		N.A.	(3)		**	
2. (Deleted)						
3. (Deleted)						
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR (STEAM JET AIR EJECTOR)						
a. Noble gas activity monitor			(2)	(1)	**	
5. (Deleted)						

TABLE 4.3.7.12-1 (Continued)

TABLE NOTATIONS

- * (Deleted)
- ** During operation of the main condenser steam jet air ejector and offgas treatment system.
- *** (Deleted)
- (1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:
 - 1. Instrument indicates measured levels above the alarm/trip setpoint.
 - 2. Circuit failure.
 - 3. Instrument indicates a downscale failure.
 - 4. Instrument controls not set in operate mode.
- (2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST), previously National Bureau of Standards, or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.
- (3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:
 - 1. 0.0 volume percent hydrogen, balance nitrogen, and
 - 2. 4 volume percent hydrogen, balance nitrogen
- (4) (Deleted)
- (5) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in the Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

- a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.
- b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.
- c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each of the required feedwater/main turbine trip system actuation instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.9.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition.

INFORMATION ON THIS PAGE HAS BEEN DELETED

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

- a. Reactor THERMAL POWER is $\leq 76.2\%$ of RATED THERMAL POWER,
- b. The recirculation flow control system is in the Local Manual mode, and
- c. The speed of the operating recirculation pump is $\leq 90\%$ of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $\leq 30\%$ of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is $\leq 50\%$ of rated loop flow.

- a. $\leq 145^{\circ}\text{F}$ between reactor vessel steam space coolant and bottom head drain line coolant,
- b. $\leq 50^{\circ}\text{F}$ between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and
- c. $\leq 50^{\circ}\text{F}$ between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 All jet pumps shall be demonstrated OPERABLE as follows:

- a. During two recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.
 1. The indicated recirculation loop flow differs by more than 10% from the established* pump speed-loop flow characteristics.
 2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from recirculation loop flow measurements.
 3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established* patterns by more than 10%.

*To be determined from the startup test program data.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

- b. During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that no two of the following conditions occur:
 - 1. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* pump speed-loop flow characteristics.
 - 2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from single recirculation loop flow measurements.
 - 3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established* single recirculation loop patterns by more than 10%.
- c. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER and upon entering single recirculation loop operation.

*To be determined from the startup test program data.

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

- a. 5% of each other with core flow greater than or equal to 70% of rated core flow.
- b. 10% of each other with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

- a. Restore the recirculation loop flows to within the specified limit within 2 hours, or
- b. Shutdown one of the recirculation loops within the next 8 hours and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program.

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

- 4 safety/relief valves @ 1170 psig $\pm 3\%$
- 5 safety/relief valves @ 1180 psig $\pm 3\%$
- 5 safety/relief valves @ 1190 psig $\pm 3\%$

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. DELETED
- c. DELETED

SURVEILLANCE REQUIREMENTS

4.4.2.1 DELETED

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program. All safety valves will be recertification tested to meet a $\pm 1\%$ tolerance prior to returning the valves to service.

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

Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

REACTOR COOLANT SYSTEM

ACTIONS (Continued)

- F. With any other two or more leak detection systems inoperable other than ACTION E above OR with required Actions and associated Completion Time of ACTIONS A, B, C, D or E not met, be in HOT SHUTDOWN within 12 hours AND in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated operable by:

- a. Perform a CHANNEL CHECK of the primary containment atmosphere gaseous radioactivity monitoring system in accordance with the Surveillance Frequency Control Program.
- b. Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation in accordance with the Surveillance Frequency Control Program. This does not apply to containment pressure and temperature monitoring system.
- c. Perform a CHANNEL CALIBRATION of required leakage detection instrumentation in accordance with the Surveillance Frequency Control Program. This does not apply to containment pressure and temperature monitoring system.
- d. Monitor primary containment pressures AND primary containment temperature in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

- a. Monitoring the primary containment atmospheric gaseous radioactivity in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),
- b. Monitoring the drywell floor drain sump and drywell equipment drain tank flow rate in accordance with the Surveillance Frequency Control Program,
- c. Monitoring the drywell unit coolers condensate flow rate in accordance with the Surveillance Frequency Control Program,
- d. Monitoring the primary containment pressure in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),
- e. Monitoring the reactor vessel head flange leak detection system in accordance with the Surveillance Frequency Control Program, and
- f. Monitoring the primary containment temperature in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

- a. In accordance with the Surveillance Frequency Control Program, and
- b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints set less than the specified allowable values by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS FREQUENCY</u>	<u>OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS IS REQUIRED</u>
1. (Deleted)		
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	In accordance with the Surveillance Frequency Control Program	1
3. (Deleted)		
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1**, 2**, 3**, 4**
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off- gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135, and Kr-88	In accordance with the Surveillance Frequency Control Program	1

**Until the specific activity of the primary coolant system is restored to within its limits.

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period,
- c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
- d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curve A, B, or C as applicable, in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and in accordance with the Surveillance Frequency Control Program during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

- a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:
 - 1. $\leq 100^{\circ}\text{F}$, in accordance with the Surveillance Frequency Control Program.
 - 2. $\leq 90^{\circ}\text{F}$, in accordance with the Surveillance Frequency Control Program.
- b. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during tensioning of the reactor vessel head bolting studs.

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1053 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1053 psig, reduce the pressure to less than 1053 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1053 psig in accordance with the Surveillance Frequency Control Program. |

*Not applicable during anticipated transients.

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two (2) independent RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one (1) RHR shutdown cooling subsystem shall be in operation. * ** ***

Each independent RHR shutdown cooling subsystem shall consist of at least:

- a. One OPERABLE RHR pump, and
- b. One OPERABLE RHR heat exchanger, not common to the two (2) independent subsystems.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

- a. With less than the above required independent RHR shutdown cooling subsystems OPERABLE, immediately initiate corrective action to return the required independent subsystems to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, verify the availability of at least one alternate method capable of decay heat removal for each inoperable independent RHR shutdown cooling subsystem. Be in at least COLD SHUTDOWN within 24 hours.****
- b. With no independent RHR shutdown cooling subsystem in operation, immediately initiate corrective action to return at least one (1) independent subsystem to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one independent RHR shutdown cooling subsystem or alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

*One independent RHR shutdown cooling subsystem may be inoperable for up to 2 hours for surveillance testing provided the other independent subsystem is OPERABLE and in operation.

**The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period provided the other independent subsystem is OPERABLE.

***The independent RHR shutdown cooling subsystem may be removed from operation during hydrostatic testing.

****Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two (2) RHR shutdown cooling subsystems shall be OPERABLE, and with no recirculation pump in operation, at least one (1) RHR shutdown cooling subsystem shall be in operation. * ** ***

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION: #

- a. With one (1) or two (2) RHR shutdown cooling subsystems inoperable:
 1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.
- b. With no RHR shutdown cooling subsystems in operation and no recirculation pump in operation:
 1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulating by an alternate method; and
 2. Once per hour monitor reactor coolant temperature and pressure.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one (1) RHR shutdown cooling subsystem or recirculation pump is operating or an alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

-
- * Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to two (2) hours per eight (8) hour period.
- ** One (1) RHR shutdown cooling subsystem may be inoperable for up to two (2) hours for the performance of Surveillances.
- *** The shutdown cooling subsystem may be removed from operation during hydrostatic testing.
- # Separate Action entry is allowed for each shutdown cooling subsystem.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

- a. In accordance with the Surveillance Frequency Control Program:
 1. For the CSS, the LPCI system, and the HPCI system:
 - a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
 - b) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.
 2. For the LPCI system, verifying that both LPCI system subsystem cross-tie valves (HV-51-182 A, B) are closed with power removed from the valve operators.
 3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.
 4. For the CSS and LPCI system, performance of a CHANNEL FUNCTIONAL TEST of the injection header ΔP instrumentation.
- b. Verifying that, when tested pursuant to Specification 4.0.5:
 1. Each CSS pump in each subsystem develops a flow of at least 3175 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥ 105 psid plus head and line losses.
 2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥ 20 psid plus head and line losses.
 3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure which corresponds to a reactor vessel pressure of 1040 psig plus head and line losses when steam is being supplied to the turbine at 1040, +13, -120 psig.**
- c. In accordance with the Surveillance Frequency Control Program:
 1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72 hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. For the HPCI system, verifying that:
 - a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥ 200 psig plus head and line losses, when steam is being supplied to the turbine at $200 + 15, - 0$ psig.**
 - b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.
 3. Performing a CHANNEL CALIBRATION of the CSS, LPCI, and HPCI system discharge line "keep filled" alarm instrumentation.
 4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be \leq the allowable value of 4.4 psid.
 5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be \leq the allowable value of 3.0 psid.
- d. For the ADS:
1. In accordance with the Surveillance Frequency Control Program, verify ADS accumulator gas supply header pressure is ≥ 90 psig. |
 2. In accordance with the Surveillance Frequency Control Program: |
 - a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.
 - b) Verify that when tested pursuant to Specification 4.0.5 that each ADS valve is capable of being opened.
 - c) DELETED

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If HPCI OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72 hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.*

4.5.2.2 The core spray system shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2a.2.b).

*One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

- a. 22'0" in accordance with the Surveillance Frequency Control Program. |
- b. 16'0" in accordance with the Surveillance Frequency Control Program. |

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, in accordance with the Surveillance Frequency Control Program: |

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

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

- a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing in accordance with the Primary Containment Leakage Rate Testing Program.
- b. In accordance with the Surveillance Frequency Control Program by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.
- c. By verifying the primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

* See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

- 4.6.1.3 The primary containment air lock shall be demonstrated OPERABLE:
- a. By verifying the seal leakage rate is in accordance with the the Primary Containment Leakage Rate Testing Program.
 - b. By conducting an overall air lock leakage test in accordance with the Primary Containment Leakage Rate Testing Program.
 - c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in the air lock can be opened at a time.***

*** Except that the airlock doors need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the airlock doors' interlock is tested within 8 hours after the primary containment has been deinerted and provided the shield door to the airlock is maintained locked closed.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -1.0 and +2.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 145°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 145°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program:

	<u>Approximate Elevation</u>	<u>Number of Installed Sensors*</u>
a.	330'	3
b.	320'	3
c.	260'	3
d.	248'	6

*At least one reading from each elevation is required for a volumetric average calculation.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system may be in operation with the supply and exhaust isolation valves in one supply line and one exhaust line open for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With a drywell and/or suppression chamber purge supply and/or exhaust isolation valve open, except as permitted above, close the valve(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8 In accordance with the Surveillance Frequency Control Program, verify each primary containment purge valve [18" or 24"] is closed.*, ** |

* Only required to be met in OPERATIONAL CONDITIONS 1, 2 and 3.

** Not required to be met when the primary containment purge valves are open for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require these valves to be open.

CONTAINMENT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.
- c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.
- d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

- a. By verifying the suppression chamber water volume to be within the limits in accordance with the Surveillance Frequency Control Program.
- b. In accordance with the Surveillance Frequency Control Program by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:
 1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.
 2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:
 - a) Suppression chamber average water temperature to be less than or equal to 110°F, and
 - b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.
 3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F by verifying suppression chamber average water temperature less than or equal to 120°F.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. By verifying at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program with the temperature alarm setpoint for:
 - 1. High water temperature:
 - a) First setpoint $\leq 95^{\circ}\text{F}$
 - b) Second setpoint $\leq 105^{\circ}\text{F}$
 - c) Third setpoint $\leq 110^{\circ}\text{F}$
 - d) Fourth setpoint $\leq 120^{\circ}\text{F}$
- d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program with the water level alarm setpoint for high water level $\leq 24'1\text{-}1/2"$.
- e. Drywell-to-suppression chamber bypass leak tests shall be conducted to coincide with the Type A test at an initial differential pressure of 4 psi and verifying that the A/\sqrt{k} calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.
- f. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area A/\sqrt{k} contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted.

CONTAINMENT SYSTEMS

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

- a. One OPERABLE RHR pump, and
- b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by | verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and the suppression pool spray sparger when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

- a. One OPERABLE RHR pump, and
- b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the flow path including the RHR heat exchanger and its associated closed bypass valve, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5.

* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

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

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

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

4.6.3.4 A representative sample of reactor instrumentation line excess flow check valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program, such that each valve is tested in accordance with the Surveillance Frequency Control Program by verifying that the valve checks flow.*

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

- a. In accordance with the Surveillance Frequency Control Program by verifying the continuity of the explosive charge.
- b. In accordance with the Surveillance Frequency Control Program by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested in accordance with the Surveillance Frequency Control Program, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

*The reactor vessel head seal leak detection line (penetration 29A) excess flow check valve is not required to be tested pursuant to this requirement.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

- a. Verified closed in accordance with the Surveillance Frequency Control Program.
- b. Demonstrated OPERABLE:
 - 1. In accordance with the Surveillance Frequency Control Program and within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.
 - 2. In accordance with the Surveillance Frequency Control Program by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.
 - 3. In accordance with the Surveillance Frequency Control Program by:
 - a) Verifying each valve's opening setpoint, from the closed position, to be 0.5 psid \pm 5%, and
 - b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.
 - c) Verifying that each outboard valve's position indicator is capable of detecting disk displacement ≥ 0.050 ", and each inboard valve's position indicator is capable of detecting disk displacement ≥ 0.120 ".

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY, restore REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

- a. Verifying in accordance with the Surveillance Frequency Control Program that the pressure within the reactor enclosure secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.
- b. Verifying in accordance with the Surveillance Frequency Control Program that:
 1. All reactor enclosure secondary containment equipment hatches and blowout panels are closed and sealed.
 2. At least one door in each access to the reactor enclosure secondary containment is closed.
 3. All reactor enclosure secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.
- c. In accordance with the Surveillance Frequency Control Program:
 1. Verifying that one standby gas treatment subsystem will draw down the reactor enclosure secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 916 seconds with the reactor enclosure recirc system in operation and
 2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the reactor enclosure secondary containment at a flow rate not exceeding 2500 cfm with wind speeds of ≤ 7.0 mph as measured on the wind instrument on Tower 1, elevation 30' or, if that instrument is unavailable, Tower 2, elevation 159'.

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

- a. Verifying in accordance with the Surveillance Frequency Control Program that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge. |
- b. Verifying in accordance with the Surveillance Frequency Control Program that:
 1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.
 2. At least one door in each access to the refueling area secondary containment is closed.
 3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.
- c. In accordance with the Surveillance Frequency Control Program: |
Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

CONTAINMENT SYSTEMS

REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.1 The reactor enclosure secondary containment ventilation system automatic isolation valves shall be OPERABLE:

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more of the reactor secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

- a. Restore the inoperable valves to OPERABLE status, or
- b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or
- c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

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

SURVEILLANCE REQUIREMENTS

4.6.5.2.1 Each reactor enclosure secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

- a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.
- b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.
- c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

- a. Restore the inoperable valves to OPERABLE status, or
- b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or
- c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, suspend handling of RECENTLY IRRADIATED FUEL in the refueling area secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

- a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.
- b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.
- c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

STANDBY GAS TREATMENT SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and when (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

ACTION:

- a. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:
 1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
 2. When (1) irradiated fuel is being handled in the refueling area secondary containment, or (2) during CORE ALTERATIONS, or (3) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel, suspend handling of IRRADIATED FUEL in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.
- b. With both standby gas treatment subsystems inoperable, if in progress, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:
 - 1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm \pm 10%.
 - 2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.
 - 3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.
 - 4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm \pm 10%.
- c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.
- d. In accordance with the Surveillance Frequency Control Program by:
 - 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm \pm 10%.

CONTAINMENT SYSTEMS

REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly.
- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:
 1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 60,000 cfm \pm 10%.
 2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ATM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
 3. Verifying a subsystem flow rate of 60,000 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows, the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
- d. In accordance with the Surveillance Frequency Control Program by:
 - 1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the filter train at a flow rate of 60,000 cfm \pm 10%, verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.
 - 2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:
 - a. Manual initiation from the control room, and
 - b. Simulated automatic initiation signal.
- e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 60,000 cfm \pm 10%.
- f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 60,000 cfm \pm 10%.

CONTAINMENT SYSTEMS

DRYWELL HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.2 Four independent drywell unit cooler hydrogen mixing subsystems (1AV212, 1BV212, 1GV212, 1HV212) shall be OPERABLE with each subsystem consisting of one unit cooler fan.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each drywell unit cooler hydrogen mixing subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

- a. Starting the system from the control room, and
- b. Verifying that the system operates for at least 15 minutes.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

- a. Within 24 hours** after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to
- b. Within 24 hours** prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The drywell and suppression chamber oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

*See Special Test Exception 3.10.5.

**Specification 3.6.1.8 is applicable during this 24 hour period.

PLANT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

5. With two RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.
 6. With three RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.
 7. With four RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.
- c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required residual heat removal service water system subsystem(s) shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*A RHRSW pump/diesel generator pair consists of a RHRSW pump and its associated diesel generator. If either a RHRSW pump or its associated diesel generator becomes inoperable, then the RHRSW pump/diesel generator pair is inoperable.

PLANT SYSTEMS
LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
 5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In OPERATIONAL CONDITION 4 or 5:
1. With only one emergency service water pump and its associated flowpath OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.
- c. In OPERATIONAL CONDITION *
1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water system loop(s) shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position. |
- b. In accordance with the Surveillance Frequency Control Program by verifying that: |
 1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal.
 2. Each pump starts automatically when its associated diesel generator starts.

*When handling irradiated fuel in the secondary containment.

**An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, then the ESW pump/diesel generator pair is inoperable.

PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

- a. A minimum pond water level at or above elevation 250' 10" Mean Sea Level, and
- b. A pond water temperature of less than or equal to 88°F.

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

ACTION:

With the requirements of the above specification not satisfied:

- a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.
- c. In OPERATIONAL CONDITION *, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

- a. By verifying the pond water level to be greater than its limit in accordance with the Surveillance Frequency Control Program.
- b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:
 1. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than or equal to 80°F; and
 2. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than or equal to 85°F; and
 3. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than 32°F.
- c. By verifying all piping above the frost line is drained:
 1. within one (1) hour after being used when ambient air temperature is below 40°F; or
 2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

*When handling irradiated fuel in the secondary containment.

PLANT SYSTEMS

3/4.7.2 CONTROL ROOM EMERGENCY FRESH AIR SUPPLY SYSTEM - COMMON SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 Two independent control room emergency fresh air supply system subsystems shall be OPERABLE.

APPLICABILITY: All OPERATIONAL CONDITIONS and when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel.

ACTION:

- a. In OPERATIONAL CONDITION 1, 2, or 3 with one control room emergency fresh air supply subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In OPERATIONAL CONDITION 4, 5, or when RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel:
 1. With one control room emergency fresh air supply subsystems inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or initiate and maintain operation of the OPERABLE subsystem in the radiation isolation mode of operation.
 2. With both control room emergency fresh air supply subsystems inoperable, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency fresh air supply subsystem shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying the control room air temperature to be less than or equal to 85°F effective temperature.
- b. In accordance with the Surveillance Frequency Control Program on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.
- c. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:
 1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 3000 cfm \pm 10%.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
3. Verifying a subsystem flow rate of 3000 cfm \pm 10% during subsystem operation when tested in accordance with ANSI N510-1980.
- d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
- e. In accordance with the Surveillance Frequency Control Program by:
 1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm \pm 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.
 2. Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:
 - a) Outside air intake high chlorine, and
 - b) Manual initiation from the control room.
 3. Verifying that on each of the below radiation isolation mode actuation test signals, the subsystem automatically switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm:
 - a) Outside air intake high radiation, and
 - b) Manual initiation from control room.

PLANT SYSTEMS

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

- a. With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.
- b. In the event the RCIC system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.
- c. Specification 3.0.4.b is not applicable to RCIC.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by: |
 1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
 2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
 3. Verifying that the pump flow controller is in the correct position.
- b. In accordance with the Surveillance Frequency Control Program by | verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1040 + 13, - 120 psig.*

* The provisions of Specification 4.0.4 are not applicable, provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam pressure to less than 150 psig within the following 72 hours.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. In accordance with the Surveillance Frequency Control Program by: |
1. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.
 2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 psig.*
 3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.
 4. Performing a CHANNEL CALIBRATION of the RCIC system discharge line "keep filled" level alarm instrumentation.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam pressure to less than 150 psig within the following 72 hours.

PLANT SYSTEMS

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

- a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:
 - 1. Decontaminate and repair the sealed source, or
 - 2. Dispose of the sealed source in accordance with Commission Regulations.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

- a. The licensee, or
- b. Other persons specifically authorized by the Commission or an Agreement State.

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

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

- a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:
 - 1. With a half-life greater than 30 days, excluding Hydrogen 3, and
 - 2. In any form other than gas.

PLANT SYSTEMS

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 The main turbine bypass system shall be OPERABLE as determined by the number of operable main turbine bypass valves being greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

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

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or take the ACTION required by Specification 3.2.3.c.

SURVEILLANCE REQUIREMENTS

4.7.8 The main turbine bypass system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program:

- a. By cycling each turbine bypass valve through at least one complete cycle of full travel,
- b. By performing a system functional test which includes simulated automatic actuation, and by verifying that each automatic valve actuates to its correct position, and
- c. By determining TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to the value specified in the CORE OPERATING LIMITS REPORT.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

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

- a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability, and
- b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

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

- a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:
 1. Verifying the fuel level in the day fuel tank.
 2. Verifying the fuel level in the fuel storage tank.
 3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.
 4. Verify that the diesel can start* and gradually accelerate to synchronous speed with generator voltage and frequency at 4280 ± 120 volts and 60 ± 1.2 HZ.
 5. Verify diesel is synchronized, gradually loaded* to an indicated 2700-2800 KW** and operates with this load for at least 60 minutes.
 6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.
 7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 225 psig.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine pre-lube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. By removing accumulated water:
 - 1) From the day tank in accordance with the Surveillance Frequency Control Program and after each occasion when the diesel is operated for greater than 1 hour, and
 - 2) From the storage tank in accordance with the Surveillance Frequency Control Program.
- c. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:
 - 1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:
 - a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.
 - b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.
 - c) A flash point equal to or greater than 125°F, and
 - d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.
 - 2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.
- d. In accordance with the Surveillance Frequency Control Program by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A, except that the filters specified in ASTM D2276-78, Sections 5.1.6 and 5.1.7, may have a nominal pore size of up to three (3) microns.
- e. In accordance with the Surveillance Frequency Control Program by:
 - 1. Deleted
 - 2. Verifying each diesel generator's capability to reject a load of greater than or equal to that of its single largest post-accident load while maintaining voltage at 4285 ± 420 volts and frequency at 60 ± 1.2 hz and after steady state conditions are reached, voltage is maintained at 4280 ± 120 volts.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the diesel generator capability to reject a load of 2850 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection. |
4. Simulating a loss-of-offsite power by itself, and: |
 - a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.
 - b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses within 10 seconds, energizes the auto-connected loads through the individual load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4280 ± 120 volts and 60 ± 1.2 Hz during this test.
5. Verifying that on an ECCS actuation test signal, without loss-of-offsite power, the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall reach 4280 ± 120 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. |
6. Simulating a loss-of-offsite power in conjunction with an ECCS actuation test signal, and: |
 - a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.
 - b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses within 10 seconds, energizes the auto-connected shutdown loads through the individual load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4280 ± 120 volts and 60 ± 1.2 Hz during this test.
7. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential over-current are automatically bypassed upon an ECCS actuation signal. |

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warm-up procedures, and as applicable regarding loading and shutdown recommendations.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

8. a) Verifying the diesel generator operates* for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 2950-3050 kW** and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 2700-2800 kW**.
- b) Verifying that, within 5 minutes of shutting down the diesel generator after the diesel generator has operated* for at least 2 hours at an indicated 2700-2800 kW**, the diesel generator starts*. The generator voltage and frequency shall reach 4280 ± 120 volts and 60 ± 1.2 Hz within 10 seconds after the start signal.
9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW.
10. Verifying the diesel generator's capability to:
 - a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
 - b) Transfer its loads to the offsite power source, and
 - c) Be restored to its standby status.
11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.
12. Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within $\pm 10\%$ of its design interval.

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:
 - a) Control Room Switch In Pull-To-Lock (With Local/Remote Switch in Remote)
 - b) Local/Remote Switch in Local
 - c) Emergency Stop
- f. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting* all four diesel generators simultaneously, during shutdown, and verifying that all four diesel generators accelerate to at least 882 rpm in less than or equal to 10 seconds.
- g. In accordance with the Surveillance Frequency Control Program by:
 1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and
 2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

h. In accordance with the Surveillance Frequency Control Program the diesel generator shall be started* and verified to accelerate to synchronous speed in less than or equal to 10 seconds. The generator voltage and frequency shall reach 4280 ± 120 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

- a) Manual***
- b) Simulated loss-of-offsite power by itself.
- c) Simulated loss-of-offsite power in conjunction with an ECCS actuation test signal.
- d) An ECCS actuation test signal by itself.

The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 2700-2800 KW** in less than or equal to 200 seconds, and operate for at least 60 minutes. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

***If diesel generator started manually from the control room, 10 seconds after the automatic prelube period.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN
LAST 20 VALID DEMANDS*

TEST FREQUENCY

less than or equal
to 1 or,

In accordance with the
Surveillance Frequency
Control Program

greater than or
equal to 2**

In accordance with the
Surveillance Frequency
Control Program

*Criteria for determining the number of failures and number of valid demands shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 demands have been reduced to one.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required division batteries and chargers shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. Each Division 1 and 2 battery float current is ≤ 2 amps, and Division 3 and 4 battery float current is ≤ 1 amp when battery terminal voltage is greater than or equal to the minimum established float voltage of 4.8.2.1.a.2, and
 2. Total battery terminal voltage for each 125-volt battery is greater than or equal to the minimum established float voltage.
- b. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. Each battery pilot cell voltage is ≥ 2.07 volts,
 2. Each battery connected cell electrolyte level is greater than or equal to minimum established design limits, and
 3. The electrolyte temperature of each pilot cell is greater than or equal to minimum established design limits.
- c. In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell voltage is ≥ 2.07 volts.
- d. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. The battery chargers will supply the currents listed below at greater than or equal to the minimum established float voltage for at least 4 hours.

<u>Charger</u>	<u>Current (Amperes)</u>
1BCA1	300
1BCA2	300
1BCB1	300
1BCB2	300
1BCC	75
1BCD	75

2. The battery capacity is adequate to supply and maintain in OPERABLE status the required emergency loads for the design duty cycle when subjected to a battery service test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- e. In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test or modified performance discharge test. The modified performance discharge test may be performed in lieu of the battery service test (Specification 4.8.2.1.d.2).
- f. Performance discharge tests or modified performance discharge tests of battery capacity shall be given as follows:
 - 1. In accordance with the Surveillance Frequency Control Program when:
 - (a) The battery shows degradation or
 - (b) The battery has reached 85% of expected life with battery capacity < 100% of manufacturer's rating, and
 - 2. In accordance with the Surveillance Frequency Control Program when battery has reached 85% of expected life with battery capacity ≥ 100% of manufacturer's rating.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one of the above required Unit 1 A.C. distribution system divisions not energized, reenergize the division within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With one of the above required Unit 1 D.C. distribution system divisions not energized, reenergize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- c. With any of the above required Unit 2 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system divisions shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5, and *.

ACTION:

- a. With less than two divisions of the above required Unit 1 A.C. distribution systems energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.
- b. With less than two divisions of the above required Unit 1 D.C. distribution systems energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.
- c. With any of the above required Unit 2 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.
- d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

*When handling irradiated fuel in the secondary containment.

ELECTRICAL POWER SYSTEMS

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of all Class 1E motor operated valves shall be either:

- a. Continuously bypassed for all valves with maintained position control switches; or,
- b. Bypassed only under accident conditions for all valves with spring-return-to-normal control switches.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously or only under accident conditions, as applicable, restore the thermal overload bypass within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves which are continuously bypassed and temporarily placed in force only when the valve motor is undergoing periodic or maintenance testing shall be verified to be bypassed following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.

4.8.4.2.2 In accordance with the Surveillance Frequency Control Program, a CHANNEL FUNCTIONAL TEST of all those valves which are bypassed only under accident conditions (valves with spring- return-to-normal control switches) shall be performed to verify that the thermal overload protection will be bypassed under accident conditions. |

ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two reactor protection system (RPS) electric power monitoring channels for each inservice RPS Inverter or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

- a. With one RPS electric power monitoring channel for an inservice RPS Inverter or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS Inverter or alternate power supply from service.
- b. With both RPS electric power monitoring channels for an inservice RPS Inverter or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 24 hours or remove the associated RPS Inverter or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

- a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.
- b. In accordance with the Surveillance Frequency Control Program by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency 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 Allowable Values.
 1. Overvoltage ≤ 127.6 VAC,
 2. Undervoltage ≥ 110.7 VAC,
 3. Underfrequency ≥ 57.05 Hz.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified, in accordance with the Surveillance Frequency Control Program.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

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

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

- a. Continuous visual indication in the control room,
- b. At least one with audible alarm in the control room,
- c. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and
- d. Unless adequate shutdown margin has been demonstrated, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.**

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

*These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core. The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

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

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

- b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.
- c. Verifying that the channel count rate is at least 3.0 cps: *
 - 1. Prior to control rod withdrawal,
 - 2. Prior to and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS, and
 - 3. In accordance with the Surveillance Frequency Control Program.
- d. Verifying, within 8 hours prior to and in accordance with the Surveillance Frequency Control Program, that the RPS circuitry "shorting links" have been removed during:
 - 1. The time any control rod is withdrawn**, unless adequate shutdown margin has been demonstrated, or
 - 2. Shutdown margin demonstrations.

*May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1. These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

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

REFUELING OPERATIONS

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified, in accordance with the Surveillance Frequency Control Program. |

*Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**See Special Test Exception 3.10.3.

REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS.*

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

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks which prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and in accordance with the Surveillance Frequency Control Program during crane operation. |

REFUELING OPERATIONS

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program during handling of fuel assemblies or control rods within the reactor pressure vessel.

REFUELING OPERATIONS

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 22 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

- a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.
- b. The SRM channels are OPERABLE per Specification 3.9.2.
- c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1c.
- d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.
- e. All other control rods are inserted.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

- a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.
- b. The SRM channels are OPERABLE per Specification 3.9.2.
- c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.
- d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.
- e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

REFUELING OPERATIONS

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

- 3.9.11.1 One (1) RHR shutdown cooling subsystem shall be OPERABLE and in operation. *

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet above the top of the reactor pressure vessel flange.

ACTION:

- a. With the required RHR shutdown cooling subsystem inoperable:
 1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available.
- b. With the required action and associated completion time of Action "a" above not met.
 1. Immediately suspend loading of irradiated fuel assemblies into the reactor pressure vessel; and
 2. Immediately initiate action to restore REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY to OPERABLE status; and
 3. Immediately initiate action to restore one (1) Standby Gas Treatment subsystem to OPERABLE status; and
 4. Immediately initiate action to restore isolation capability in each required Refueling Floor secondary containment penetration flow path not isolated.
- c. With no RHR shutdown cooling subsystem in operation:
 1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and
 2. Once per hour monitor reactor coolant temperature.

SURVEILLANCE REQUIREMENTS

- 4.9.11.1 At least one (1) RHR shutdown cooling subsystem, or an alternate method, shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* The required RHR shutdown cooling subsystem may be removed from operation for up to two (2) hours per eight (8) hour period.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two (2) RHR shutdown cooling subsystems shall be OPERABLE, and one (1) RHR shutdown cooling subsystem shall be in operation. *

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor pressure vessel flange.

ACTION:

- a. With one (1) or two (2) required RHR shutdown cooling subsystems inoperable:
 1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.
- b. With the required action and associated completion time of Action "a" above not met:
 1. Immediately initiate action to restore REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY to OPERABLE status; and
 2. Immediately initiate action to restore one (1) Standby Gas Treatment subsystem to OPERABLE status; and
 3. Immediately initiate action to restore isolation capability in each required Refueling Floor secondary containment penetration flow path not isolated.
- c. With no RHR shutdown cooling subsystem in operation:
 1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and
 2. Once per hour monitor reactor coolant temperature.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one (1) RHR shutdown cooling subsystem, or an alternate method, shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* The required operating shutdown cooling subsystem may be removed from operation for up to two (2) hours per eight (8) hour period.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during low power PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

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

- a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.
- b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.
- c. The "continuous rod withdrawal" control shall not be used during out-of-sequence movement of the control rods.
- d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during the performance of a shutdown margin demonstration, verify that;

- a. The source range monitors are OPERABLE per Specification 3.9.2,
- b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown margin demonstration procedures, and
- c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation may be suspended for up to 24 hours for the performance of:

- a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or
- b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

- a. With the above specified time limit exceeded, insert all control rods.
- b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.3 may be suspended during the performance of the Startup Test Program until either the required 100% of RATED THERMAL POWER trip tests have been completed or the reactor has operated for 120 Effective Full Power Days.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The Effective Full Power Days of operation shall be verified to be less than 120, by calculation, in accordance with the Surveillance Frequency Control Program during the Startup Test Program.

SPECIAL TEST EXCEPTIONS

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during training startups. |

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

- a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.6 The rate of the sum of the activities of the noble gases Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, and Xe-138 measured at the recombiner after-condenser discharge shall be limited to less than or equal to 330 millicuries/second.

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

ACTION:

With the rate of the sum of the activities of the specified noble gases at the recombiner after-condenser discharge exceeding 330 millicuries/second, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.6.1 The rate of the sum of the activities of noble gases at the recombiner after-condenser discharge shall be continuously monitored in accordance with Specification 3.3.7.12.

4.11.2.6.2 The rate of the sum of the activities of the specified noble gases from the recombiner after-condenser discharge shall be determined to be within the limits of Specification 3.11.2.6 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner after condenser discharge:

- a. In accordance with the Surveillance Frequency Control Program.
- b. Within 4 hours following an increase, as indicated by the Main Condenser Off-Gas Pretreatment Radioactivity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level or air in-leakage, in the nominal steady-state fission gas release from the primary coolant.
- c. The provisions of Specification 4.0.4 are not applicable.

*When the main condenser air ejector is in operation.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

- a. Preserve the integrity of the fuel cladding.
- b. Preserve the integrity of the reactor coolant system.
- c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and
- d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The APRM system is divided into four APRM channels and four 2-Out-Of-4 Voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed.

The system meets the intent of IEEE-279 for nuclear power plant protection systems. Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System" and NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function." The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The APRM Functions include five Functions accomplished by the four APRM channels (Functions 2.a, 2.b, 2.c, 2.d, and 2.f) and one accomplished by the four 2-Out-Of-4 Voter channels (Function 2.e). Two of the five Functions accomplished by the APRM channels are based on neutron flux only (Functions 2.a and 2.c), one Function is based on neutron flux and recirculation drive flow (Function 2.b) and one is based on equipment status (Function 2.d). The fifth Function accomplished by the APRM channels is the Oscillation Power Range Monitor (OPRM) Upscale trip Function 2.f, which is based on detecting oscillatory characteristics in the neutron flux. The OPRM Upscale Function is also dependent on average neutron flux (Simulated Thermal Power) and recirculation drive flow, which are used to automatically enable the output trip.

The Two-Out-Of-Four Logic Module includes 2-Out-Of-4 Voter hardware and the APRM Interface hardware. The 2-Out-Of-4 Voter Function 2.e votes APRM Functions 2.a, 2.b, 2.c, and 2.d independently of Function 2.f. This voting is accomplished by the 2-Out-Of-4 Voter hardware in the Two-Out-Of-Four Logic Module. The voter includes separate outputs to RPS for the two independently voted sets of Functions, each of which is redundant (four total outputs). The analysis in Reference 2 took credit for this redundancy in the justification of the 12-hour allowed out-of-service time for

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION (continued)

10% of RATED THERMAL POWER. Plant operational experience has shown that this flow correlation methodology is consistent with the guidance and intent in Reference 8. Changes throughout the cycle in the drive flow / core flow relationship due to the changing thermal hydraulic operating conditions of the core are accounted for in the margins included in the bases or analyses used to establish the setpoints for the APRM Simulated Thermal Power - Upscale Function and the OPRM Upscale Function.

As noted in Table 3.3.1-2, Note "*", the redundant outputs from the 2-Out-Of-4 Voter channel are considered part of the same channel, but the OPRM and APRM outputs are considered to be separate channels, so N = 8 to determine the interval between tests for application of Specification 4.3.1.3 (REACTOR PROTECTION SYSTEM RESPONSE TIME). The note further requires that testing of OPRM and APRM outputs shall be alternated.

Each test of an OPRM or APRM output tests each of the redundant outputs from the 2-Out-Of-4 Voter channel for that function, and each of the corresponding relays in the RPS. Consequently, each of the RPS relays is tested every fourth cycle. This testing frequency is twice the frequency justified by References 2 and 3.

Automatic reactor trip upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A.C. Thadani, NRC, dated May 15, 1991).

The measurement of response time at the frequencies specified in the Surveillance Frequency Control Program provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable except for the APRM Simulated Thermal Power - Upscale and Neutron Flux - Upscale trip functions and the OPRM Upscale trip function (Table 3.3.1-2, Items 2.b, 2.c, and 2.f). Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in-place, on-site or off-site test measurements, or (2) utilizing replacement sensors with certified response times. Response time testing for the sensors as noted in Table 3.3.1-2 is not required based on the analysis in NEDO-32291-A. Response time testing for the remaining channel components is required as noted. For the digital electronic portions of the APRM functions, performance characteristics that determine response time are checked by a combination of automatic self-test, calibration activities, and response time tests of the 2-Out-Of-4 Voter (Table 3.3.1-2, Item 2.e).

REACTIVITY CONTROL SYSTEMS

BASES

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillances are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3. Additional pertinent analysis is also contained in Amendment 17 to the Reference 4 topical report.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density over the range of power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. RBM OPERABILITY is required when the limiting condition described in Specification 3.1.4.3 exists.

REACTIVITY CONTROL SYSTEMS

BASES

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume assures that the solution is available for use. |

Replacement of the explosive charges in the valves will assure that these valves will not fail because of deterioration of the charges. |

The Standby Liquid Control System also has a post-DBA LOCA safety function to buffer Suppression Pool pH in order to maintain bulk pH above 7.0. The buffering of Suppression Pool pH is necessary to prevent iodine re-evolution to satisfy the methodology for Alternative Source Term. Manual initiation is used, and the minimum amount of total boron required for Suppression Pool pH buffering is 240 lbs. Given that at least 185 lbs of Boron-10 is maintained in the tank, the total boron in the tank will be greater than 240 lbs for the range of enrichments from 29% to 62%.

ACTION Statement (a) applies only to OPERATIONAL CONDITIONS 1 and 2 because a single pump can satisfy both the reactor control function and the post-DBA LOCA function to control Suppression Pool pH since boron injection is not required until 13 hours post-LOCA. ACTION Statement (b) applies to OPERATIONAL CONDITIONS 1, 2 and 3 to address the post-LOCA safety function of the SLC system.

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1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972.
 2. C. J. Paone, R. C. Stirn, and R. M. Young, Supplement 1 to NEDO-10527, July 1972.
 3. J. M. Haun, C. J. Paone, and R. C. Stirn, Addendum 2, "Exposed Cores." Supplement 2 to NEDO-10527, January 1973.
 4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".
 5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," NEDC-32193P, Revision 2, October 1993.

INSTRUMENTATION

BASES

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to D.N. Grace from C.E. Rossi dated January 6, 1989) and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," as approved by the NRC and documented in the NRC SER (letter to S.D. Floyd from C.E. Rossi dated June 18, 1990).

Automatic closure of the MSIVs upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A.C. Thadani, NRC, dated May 15, 1991).

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Response time testing for sensors are not required based on the analysis in NEDO 32291-A. Response time testing of the remaining channel components is required as noted in Table 3.3.2-3.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. Primary containment isolation valves that are actuated by the isolation signals specified in Technical Specification Table 3.3.2-1 are identified in Technical Requirements Manual Table 3.6.3-1.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

INSTRUMENTATION

BASES

3/4.3.3 EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION (Continued)

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30936P, Parts 1 and 2, "Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," as approved by the NRC and documented in the SER (letter to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1) and letter to D. N. Grace from C. E. Rossi dated December 9, 1988 (Part 2)).

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power for energizing various components such as pump motors, motor operated valves, and the associated control components. If the loss of power instrumentation detects that voltage levels are too low, the buses are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources. The loss of power relays in each channel have sufficient overlapping detection characteristics and functionality to permit operation subject to the conditions in Action Statement 37. Bases 3/4.8.1, 3/4.8.2, and 3/4.8.3 provide discussion regarding parametric bounds for determining operability of the offsite sources. Those Bases assume that the loss of power relays are operable. With an inoperable 127Z-11X0X relay, the grid voltage is monitored to 230kV (for the 101 Safeguard Bus Source) or 525kV (for the 201 Safeguard Bus Source) to increase the margin for the operation of the 127Z-11X0X relay.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

INSTRUMENTATION

BASES

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been specified in accordance with recommendations made by GE in their letter to the BWR Owner's Group dated August 7, 1989, SUBJECT: "Clarification of Technical Specification changes given in ECCS Actuation Instrumentation Analysis."

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," as approved by the NRC and documented in the SER (letter to D. N. Grace from C. E. Rossi dated September 22, 1988).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

INSTRUMENTATION

BASES

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

The surveillance interval for the Main Control Room Normal Fresh Air Supply Radiation Monitor is determined in accordance with the Surveillance Frequency Control Program.

3/4.3.7.2 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE UFSAR.

3/4.3.7.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50, Appendix A.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

INSTRUMENTATION

BASES

3/4.3.7.7 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

There are three toxic gas detection subsystems. The high toxic chemical concentration alarm in the Main Control Room annunciates when two of the three subsystems detect a high toxic gas concentration. An Operate/Inop keylock switch is provided for each subsystem which allows an individual subsystem to be placed in the tripped condition. Placing the keylock switch in the INOP position initiates one of the two inputs required to initiate the alarm in the Main Control Room.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

3/4.3.7.9 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

REACTOR COOLANT SYSTEM

BASES

ACTIONS (Continued)

- C. With the required primary containment air cooler condensate flow rate monitoring system inoperable, SR 4.4.3.1.a must be performed every 8 hours to provide periodic information of activity in the primary containment of more frequent interval than the routine frequency of every 12 hours. The 8 hour interval provides periodic information that is adequate to detect leakage and recognizes that other forms of leakage detection are available. The required ACTION has been clarified to state that the additional surveillance requirement is not applicable if the required primary containment atmosphere gaseous radioactivity monitoring system is also inoperable. Consistent with SR 4.0.3, surveillances are not required to be performed on inoperable equipment. In this case, ACTION Statement A. and E. requirements apply.
- D. With the primary containment pressure and temperature monitoring system inoperable, operation may continue for up to 30 days given the system's indirect capability to detect RCS leakage. However, other more limiting Tech Spec requirements associated with the primary containment pressure/temperature monitoring system will still apply.
- E. With both the primary containment atmosphere gaseous radioactivity monitor and the primary containment air cooler condensate flow rate monitor inoperable, the only means of detecting leakage is the drywell floor drain sump monitor and the drywell pressure/temperature instrumentation. This condition does not provide the required diverse means of leakage detection. The required ACTION is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period. While in the primary containment atmosphere gases radioactivity monitor is INOPERABLE, Primary containment atmospheric grab samples will be taken and analyzed every 12 hours since ACTION Statement A. requirements also apply.
- F. If any required ACTION of Conditions A, B, C, D or E cannot be met within the associated Completion Time, the plant must be brought to an OPERATIONAL CONDITION in which the LCO does not apply. To achieve this status, the plant must be brought to at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the next 24 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the ACTIONS in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 4.4.3.1.a

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmospheric monitoring system. The check gives reasonable confidence that the channel is operating properly.

REACTOR COOLANT SYSTEM

BASES

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.4.3.1.b

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string.

SR 4.4.3.1.c

The SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment.

SR 4.4.3.1.d

This SR provides a routine check of primary containment pressure and temperature for indirect evidence of RCS leakage.

REFERENCES

1. LGS UFSAR, Section 5.2.5.1.
2. Regulatory Guide 1.45, May 1973.
3. LGS UFSAR, Section 5.2.5.2.1.3.
4. LGS UFSAR, Section 5.2.5.2.1.5.
5. LGS UFSAR, Section 5.2.5.2.1.4.
6. LGS UFSAR, Section 5.2.5.2.1.1(2).
7. GEAP-5620, April 1968
8. NUREG-75/067, October 1975.
9. LGS UFSAR, Section 5.2.5.6.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. The limit of 2 gpm increase in UNIDENTIFIED LEAKAGE over a 24-hour period and the monitoring of drywell floor drain sump and drywell equipment drain tank flow rate at least once every eight (8) hours conforms with NRC staff positions specified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as revised by NRC Safety Evaluation dated March 6, 1990. The ACTION requirement for the 2 gpm increase in UNIDENTIFIED LEAKAGE limit ensures that such leakage is identified or a plant shutdown is initiated to allow further investigation and corrective action. Once identified, reactor operation may continue dependent upon the impact on total leakage.

EMERGENCY CORE COOLING SYSTEM

BASES

ECCS - OPERATING and SHUTDOWN (Continued)

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system. The HPCI system, and one LPCI subsystem, and/or one CSS subsystem out-of-service period of 8 hours ensures that sufficient ECCS, comprised of a minimum of one CSS subsystem, three LPCI subsystems, and all of the ADS will be available to 1) provide for safe shutdown of the facility, and 2) mitigate and control accident conditions within the facility. A Note prohibits the application of Specification 3.0.4.b to an inoperable HPCI subsystem. There is an increased risk associated with entering an OPERATIONAL CONDITION or other specified condition in the Applicability with an inoperable HPCI subsystem and the provisions of Specification 3.0.4.b, which allow entry into an OPERATIONAL CONDITION or other specified condition in the Applicability with the Limiting Condition for Operation not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety/relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls five selected safety-relief valves. The safety analysis assumes all five are operable. The allowed out-of-service time for one valve for up to fourteen days is determined in a similar manner to other ECCS sub-system out-of-service time allowances.

Verification that ADS accumulator gas supply header pressure is ≥ 90 psig ensures adequate gas pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator at least two valve actuations can occur with the drywell at 70% of design pressure. The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 90 psig is provided by the PCIG supply.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum calculated pressure in the event of a LOCA. A visual inspection in accordance with the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the calculated containment peak pressure does not exceed the design pressure of 55 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5.0 psid. The limit of - 1.0 to + 2.0 psig for initial containment pressure will limit the total pressure to ≤ 44 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. Limiting the use of the drywell and suppression chamber purge system to specific criteria is imposed to protect the integrity of the SGTS filters. Analysis indicates that should a LOCA occur while this pathway is being utilized, the associated pressure surge through the (18 or 24") purge lines will adversely affect the integrity of SGTS. This condition is not imposed on the 1 and 2 inch valves used for pressure control since a surge through these lines does not threaten the operability of SGTS.

Surveillance requirement 4.6.1.8 ensures that the primary containment purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. The SR is modified by a Note stating that primary containment purge valves are only required to be closed in OPERATIONAL CONDITIONS 1, 2 and 3. The SR is also modified by a Note stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. The 18 or 24 inch purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time.

CONTAINMENT SYSTEMS

BASES

DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test is not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The scram discharge volume vent and drain valves serve a dual function, one of which is primary containment isolation. Since the other safety functions of the scram discharge volume vent and drain valves would not be available if the normal PCIV actions were taken, actions are provided to direct the user to the scram discharge volume vent and drain operability requirements contained in Specification 3.1.3.1. However, since the scram discharge volume vent and drain valves are PCIVs, the Surveillance Requirements of Specification 4.6.3 still apply to these valves.

The opening of a containment isolation valve that was locked or sealed closed to satisfy Technical Specification 3.6.3 Action statements, may be reopened on an intermittent basis under administrative controls. These controls consist of stationing a dedicated individual at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

Primary containment isolation valves governed by this Technical Specification are identified in Table 3.6.3-1 of the TRM.

This Surveillance Requirement requires a demonstration that a representative sample of reactor instrument line excess flow check valves (EFCVs) is OPERABLE by verifying that the valve actuates to the isolation position on a simulated instrument line break signal. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested in accordance with the Surveillance Frequency Control Program. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This Surveillance Requirement provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. Furthermore, any EFCV failures will be evaluated to determine if additional testing in the test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint. For some EFCVs, this Surveillance can be performed with the reactor at power.

CONTAINMENT SYSTEMS

BASES

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system in accordance with the Surveillance Frequency Control Program, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY and Control Room radiation doses associated with containment leakage. The operation of these systems and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analysis. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

As a result of the Alternative Source Term (AST) project, secondary containment integrity of the refueling area is not required during certain conditions when handling irradiated fuel, during CORE ALTERATIONS, or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel. The control room dose analysis for the fuel handling accident (FHA) is based on unfiltered releases from the South Stack and therefore, does not require the standby gas treatment system to be aligned to the refueling area whenever irradiated fuel is being handled, during CORE ALTERATIONS or operations are being conducted that have the potential to drain the reactor vessel. The OPERABILITY of the standby gas treatment system however, ensures that release pathways do not result in a control room dose higher than determined in the refueling accident dose analysis. This is accomplished by assuring that the Standby Gas Treatment System is OPERABLE if releases from the refueling area are made through the North Stack following a FHA.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

The SGTS fans are sized for three zones and therefore, when aligned to a single zone or two zones, will have excess capacity to more quickly drawdown the affected zones. There is no maximum flow limit to individual zones or pairs of zones and the air balance and drawdown time are verified when all three zones are connected to the SGTS.

The three zone air balance verification and drawdown test will be done after any major system alteration, which is any modification which will have an effect on the SGTS flowrate such that the ability of the SGTS to drawdown the reactor enclosure to greater than or equal to 0.25 inch of vacuum water gage in less than or equal to 916 seconds could be affected.

ELECTRICAL POWER SYSTEMS

BASES

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

"Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 except for paragraphs C.2.a(3), C.2.c(1), C.2.c(2), C.2.d(3) and C.2.d(4), and the periodic testing will be performed in accordance with the Surveillance Frequency Control Program. The exceptions to Regulatory Guide 1.108 allow for gradual loading of diesel generators during testing and decreased surveillance test frequencies (in response to Generic Letter 84-15). The single largest post-accident load on each diesel generator is the RHR pump.

The Surveillance Requirement for removal of accumulated water from the fuel oil storage tanks is for preventive maintenance. The presence of water does not necessarily represent failure of the Surveillance Requirement, provided the accumulated water is removed during performance of the Surveillance. Accumulated water in the fuel oil storage tanks constitutes a collection of water at a level that can be consistently and reliably measured. The minimum level at which accumulated water can be consistently and reliably measured in the fuel oil storage tank sump is 0.25 inches. Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of accumulated water from the fuel storage tanks once every (31) days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137.

The surveillance requirements for demonstrating the OPERABILITY of the units batteries are in accordance with the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

Verifying battery float current while on float charge (4.8.2.1.a.1) is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery.

Use of float current to determine the state of charge of the battery is consistent with IEEE Standard 450-1995.

This Surveillance Requirement (4.8.2.1.a.1) states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of 4.8.2.1.a.2. When this float voltage is not maintained, the Actions of 3.8.2.1 Action a., provides the necessary and appropriate verifications of the battery condition. Furthermore, the float current limits are established based on the float voltage range and is not directly applicable when this voltage is not maintained.

ELECTRICAL POWER SYSTEMS

BASES

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Verifying, per 4.8.2.1.a.2, battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the minimum float voltage established by the battery manufacturer (2.20 Vpc, average, or 132 V at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years).

Surveillance Requirements 4.8.2.1.b.1 and 4.8.2.1.c require verification that the cell float voltages are equal to or greater than 2.07 V.

The limit specified in 4.8.2.1.b.2 for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability.

Surveillance Requirement 4.8.2.1.b.3 verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60 degrees Fahrenheit). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity.

Surveillance Requirement 4.8.2.1.d.1 verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32, the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

Surveillance Requirement 4.8.2.1.d.1 requires that each battery charger be capable of supplying the amps listed for the specified charger at the minimum established float voltage for 4 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. This time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

A battery service test, per 4.8.2.1.d.2, is a special test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in the UFSAR.

ADMINISTRATIVE CONTROLS
PROCEDURES AND PROGRAMS (Continued)

i. Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications," of the following:

- a. Actions to restore battery cells with float voltage < 2.13 volts, and
- b. Actions to equalize and test battery cells that have been discovered with electrolyte level below the minimum established design limit.

j. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 0.
- c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

- (4) Pursuant to the Act and 10 CFR Parts 30, 40, 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and
 - (5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.
- C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- (1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels of 3458 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.
 - (2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 147 are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.
 - (3) Fire Protection (Section 9.5, SSER-2, -4)*

Exelon Generation Company shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report dated August 1983 through Supplement 9, dated August 1989, and Safety Evaluation dated November 20, 1995, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

*The parenthetical notation following the title of license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

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REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

- a) Electrically, or
 - b) Hydraulically by closing the drive water and exhaust water isolation valves.
2. If the inoperable control rod(s) is inserted, within 1 hour disarm the associated directional control valves** either:
- a) Electrically, or
 - b) Hydraulically by closing the drive water and exhaust water isolation valves.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

- c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.
- d. With one or more scram discharge volume (SDV) vent or drain lines with one valve inoperable, restore the inoperable valve(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours*** and in COLD SHUTDOWN within the following 24 hours.
- e. With one or more SDV vent or drain lines with both valves inoperable, isolate the associated line within 8 hours **** or be in at least HOT SHUTDOWN within the next 12 hours*** and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by: |

- a. Verifying each valve to be open,* and |
- b. Cycling each valve through at least one complete cycle of full travel. |

* These valves may be closed intermittently for testing under administrative controls.

** May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

***Separate Action entry is allowed for each SDV vent and drain line.

****An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.2 When above the preset power level of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

- a. In accordance with the Surveillance Frequency Control Program, and
- b. Within 24 hours from discovery that a control rod is immovable as a result of excessive friction or mechanical interference.

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

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

- a. The scram discharge volume drain and vent valves OPERABLE in accordance with the Surveillance Frequency Control Program, by verifying that the drain and vent valves:
 1. Close within 30 seconds after receipt of a signal for control rods to scram, and
 2. Open when the scram signal is reset.
- b. Proper level sensor response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation in accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

- a. Declare the control rod(s) with the slow insertion time inoperable, and
- b. Perform the Surveillance Requirements of Specification 4.1.3.2c. at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of 7.0 seconds.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement and, during single control rod scram time tests, the control rod drive pumps shall be isolated from the accumulators:

- a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER with reactor coolant pressure greater than or equal to 950 psig, following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days.
- b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods in accordance with either "1" or "2" as follows:
 - 1.a Specifically affected individual control rods shall be scram time tested at zero reactor coolant pressure and the scram insertion time from the fully withdrawn position to notch position 05 shall not exceed 2.0 seconds, and
 - 1.b Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure prior to exceeding 40% of RATED THERMAL POWER.
 2. Specifically affected individual control rods shall be scram time tested at greater than or equal to 950 psig reactor coolant pressure.
- c. For at least 10% of the control rods, with reactor coolant pressure greater than or equal to 950 psig, on a rotating basis, and in accordance with the Surveillance Frequency Control Program.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

2. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

4.1.3.5 Each control rod scram accumulator shall be determined OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that the indicated pressure is greater than or equal to 955 psig unless the control rod is inserted and disarmed or scrambled.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod position indication system shall be determined OPERABLE by verifying:

- a. In accordance with the Surveillance Frequency Control Program that the position of each control rod is indicated,
- b. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.1.3.1.2, and
- c. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.1.3.6b.

REACTIVITY CONTROL SYSTEMS

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than 1.70, or THERMAL POWER greater than or equal to 90% of rated with MCPR less than 1.40.

ACTION:

- a. With one RBM channel inoperable:
 1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and
 2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.
- b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

- a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.
- b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

REACTIVITY CONTROL SYSTEMS

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of a minimum of the following:

- a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,
- b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3

ACTION:

- a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.
- b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that:
 - 1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.
 - 2. The available volume of sodium pentaborate solution is at least 3160 gallons.
 - 3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. In accordance with the Surveillance Frequency Control Program by:
1. Verifying the continuity of the explosive charge.
 2. Determining by chemical analysis and calculation* that the available weight of Boron-10 is greater than or equal to 185 lbs; the concentration of sodium pentaborate in solution is less than or equal to 13.8% and within the limits of Figure 3.1.5-1 and; the following equation is satisfied:
$$\frac{C}{13\% \text{ wt.}} \times \frac{E}{29 \text{ atom \%}} \times \frac{Q}{86 \text{ gpm}} \geq 1$$
where
C = Sodium pentaborate solution (% by weight)
Q = Two pump flowrate, as determined per surveillance requirement 4.1.5.c.
E = Boron 10 enrichment (atom % Boron 10)
 3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm per pump at a pressure of greater than or equal to 1230±25 psig is met.
- d. In accordance with the Surveillance Frequency Control Program by:
1. Initiating at least one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of the batch successfully fired. All injection loops shall be tested in 3 operating cycles.
 2. Verify all heat-treated piping between storage tank and pump suction is unblocked.**
- e. Prior to addition of Boron to storage tank verify sodium pentaborate enrichment to be added is ≥ 29 atom % Boron 10.

* This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after water or boron addition or solution temperature is restored.

** This test shall also be performed whenever suction piping temperature drops below the limits of Figure 3.1.5-1 for the most recent concentration analysis, within 24 hours after solution temperature is restored.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) as shown in the CORE OPERATING LIMITS REPORT (COLR). During operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the appropriate reduction factors for power and flow as defined in the COLR.

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

ACTION:

With an APLHGR exceeding the limiting value, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limiting value:

- a. In accordance with the Surveillance Frequency Control Program, |
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and in accordance with the Surveillance Frequency Control Program | when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

- a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) EOC-RPT inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.
- b. With MCPR less than the applicable MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.
- c. With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

- a. $\tau = 1.0$ prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2a and during reactor startups prior to control rod scram time tests in accordance with Specification 4.1.3.2.b.1.b, or
- b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit including application of the MCPR(P) and MCPR(F) factors as determined from the CORE OPERATING LIMITS REPORT.

- a. In accordance with the Surveillance Frequency Control Program,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
- d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the value in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGRs shall be determined to be equal to or less than the limit:

- a. In accordance with the Surveillance Frequency Control Program, |
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR. |
- d. The provisions of Specification 4.0.4 are not applicable.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.1.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program, except Table 4.3.1.1-1 Functions 2.a, 2.b, 2.c, 2.d, 2.e, and 2.f. Functions 2.a, 2.b, 2.c, 2.d, and 2.f do not require separate LOGIC SYSTEM FUNCTIONAL TESTS. For Function 2.e, tests shall be performed in accordance with the Surveillance Frequency Control Program. LOGIC SYSTEM FUNCTIONAL TEST for Function 2.e includes simulating APRM and OPRM trip conditions at the APRM channel inputs to the voter channel to check all combinations of two tripped inputs to the 2-Out-Of-4 voter logic in the voter channels.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific reactor trip system.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK (n)</u>	<u>CHANNEL FUNCTIONAL TEST (n)</u>	<u>CHANNEL CALIBRATION(a)(n)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. Intermediate Range Monitors:					
a. Neutron Flux - High	(b)	(j)		2 3(i), 4(i), 5(i)	
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)	
2. Average Power Range Monitor(f):					
a. Neutron Flux - Upscale (Setdown)	(b)	(l)		2	
b. Simulated Thermal Power - Upscale		(e)	(d), (g)	1	
c. Neutron Flux - Upscale			(d)	1	
d. Inoperative	N.A.		N.A.	1, 2	
e. 2-Out-Of-4 Voter			N.A.	1, 2	
f. OPRM Upscale		(e)	(c)(g)	1(m)	
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)	
4. Reactor Vessel Water Level- Low, Level 3				1, 2	
5. Main Steam Line Isolation Valve - Closure	N.A.			1	
6. DELETED					
7. Drywell Pressure - High				1, 2	
8. Scram Discharge Volume Water Level - High					
a. Level Transmitter				1, 2, 5(i)	
b. Float Switch	N.A.			1, 2, 5(i)	

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK (n)</u>	<u>CHANNEL FUNCTIONAL TEST (n)</u>	<u>CHANNEL CALIBRATION(a)(n)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
- (c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is $\geq 30\%$ and for recirculation drive flow is $< 60\%$.
- (d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq 25\%$ of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
- (e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
- (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
- (g) The less frequent calibration includes the flow input function.
- (h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
- (i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
- (j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
- (k) DELETED
- (l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
- (m) With THERMAL POWER $\geq 25\%$ of RATED THERMAL POWER.
- (n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.2-1.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.2.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operations of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in accordance with the Surveillance Frequency Control Program, where N is the total number of redundant channels in a specific isolation trip system.

** The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. <u>MAIN STEAM LINE ISOLATION</u>					
a. Reactor Vessel Water Level					
1) Low, Low, Level 2				1, 2, 3	
2) Low, Low, Low - Level 1				1, 2, 3	
b. DELETED				DELETED	
c. Main Steam Line Pressure - Low				1	
d. Main Steam Line Flow - High				1, 2, 3	
e. Condenser Vacuum - Low				1, 2**, 3**	
f. Outboard MSIV Room Temperature - High				1, 2, 3	
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High				1, 2, 3	
h. Manual Initiation	N.A.		N.A.	1, 2, 3	
2. <u>RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION</u>					
a. Reactor Vessel Water Level## Low - Level 3				1, 2, 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High				1, 2, 3	
c. Manual Initiation	N.A.		N.A.	1, 2, 3	

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
3. <u>REACTOR WATER CLEANUP SYSTEM ISOLATION</u>					
a. RWCS Δ Flow - High				1, 2, 3	
b. RWCS Area Temperature - High				1, 2, 3	
c. RWCS Area Ventilation Δ Temperature - High				1, 2, 3	
d. SLCS Initiation	N.A.		N.A.	1, 2, 3	
e. Reactor Vessel Water Level Low, Low, - Level 2				1, 2, 3	
f. Manual Initiation	N.A.		N.A.	1, 2, 3	
4. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION</u>					
a. HPCI Steam Line Δ Pressure - High				1, 2, 3	
b. HPCI Steam Supply Pressure, Low				1, 2, 3	
c. HPCI Turbine Exhaust Diaphragm Pressure - High				1, 2, 3	
d. HPCI Equipment Room Temperature - High				1, 2, 3	
e. HPCI Equipment Room Δ Temperature - High				1, 2, 3	
f. HPCI Pipe Routing Area Temperature - High				1, 2, 3	
g. Manual Initiation	N.A.		N.A.	1, 2, 3	
h. HPCI Steam Line Δ Pressure Timer	N.A.			1, 2, 3	

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
5. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>					
a. RCIC Steam Line Δ Pressure - High				1, 2, 3	
b. RCIC Steam Supply Pressure - Low				1, 2, 3	
c. RCIC Turbine Exhaust Diaphragm Pressure - High				1, 2, 3	
d. RCIC Equipment Room Temperature - High				1, 2, 3	
e. RCIC Equipment Room Δ Temperature - High				1, 2, 3	
f. RCIC Pipe Routing Area Temperature - High				1, 2, 3	
g. Manual Initiation	N.A.		N.A.	1, 2, 3	
h. RCIC Steam Line Δ Pressure Timer	N.A.			1, 2, 3	

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
6. <u>PRIMARY CONTAINMENT ISOLATION</u>					
a. Reactor Vessel Water Level					
1) Low, Low - Level 2				1, 2, 3	
2) Low, Low, Low - Level 1				1, 2, 3	
b. Drywell Pressure ## - High				1, 2, 3	
c. North Stack Effluent Radiation - High				1, 2, 3	
d. Deleted					
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High				1, 2, 3	
f. Deleted					
g. Deleted					
h. Drywell Pressure - High/ Reactor Pressure - Low				1, 2, 3	
i. Primary Containment Instrument Gas to Drywell Δ Pressure - Low	N.A.			1, 2, 3	
j. Manual Initiation	N.A.		N.A.	1, 2, 3	

TABLE 4.3.2.1-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST(a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
7. <u>SECONDARY CONTAINMENT ISOLATION</u>					
a. Reactor Vessel Water Level Low, Low - Level 2				1, 2, 3	
b. Drywell Pressure## - High				1, 2, 3	
c.1. Refueling Area Unit 1 Ventilation Exhaust Duct Radiation - High				*#	
2. Refueling Area Unit 2 Ventilation Exhaust Duct Radiation - High				*#	
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High				1, 2, 3	
e. Deleted					
f. Deleted					
g. Reactor Enclosure Manual Initiation	N.A.		N.A.	1, 2, 3	
h. Refueling Area Manual Initiation	N.A.		N.A.	*	

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

*Required when (1) handling RECENTLY IRRADIATED FUEL in the secondary containment, or (2) during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

**When not administratively bypassed and/or when any turbine stop valve is open.

#During operation of the associated Unit 1 or Unit 2 ventilation exhaust system.

##These trip functions (2a, 6b, and 7b) are common to the RPS actuation trip function.

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1

ACTION:

- a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to Operable status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.
- c. With either ADS trip system subsystem inoperable, restore the inoperable trip system to OPERABLE status within:
 1. 7 days, provided that the HPCI and RCIC systems are OPERABLE.
 2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.3.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific ECCS trip system.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK(a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. <u>CORE SPRAY SYSTEM</u>					
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3, 4*, 5*	
b. Drywell Pressure - High				1, 2, 3	
c. Reactor Vessel Pressure - Low				1, 2, 3, 4*, 5*	
d. Manual Initiation	N.A.		N.A.	1, 2, 3, 4*, 5*	
2. <u>LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM</u>					
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3, 4*, 5*	
b. Drywell Pressure - High				1, 2, 3	
c. Reactor Vessel Pressure - Low				1, 2, 3	
d. Injection Valve Differential Pressure - Low (Permissive)				1, 2, 3, 4*, 5*	
e. Manual Initiation	N.A.		N.A.	1, 2, 3, 4*, 5*	
3. <u>HIGH PRESSURE COOLANT INJECTION SYSTEM***</u>					
a. Reactor Vessel Water Level - Low Low, Level 2				1, 2, 3	
b. Drywell Pressure - High				1, 2, 3	
c. Condensate Storage Tank Level - Low				1, 2, 3	
d. Suppression Pool Water Level - High				1, 2, 3	
e. Reactor Vessel Water Level - High, Level 8				1, 2, 3	
f. Manual Initiation	N.A.		N.A.	1, 2, 3	

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL FUNCTIONAL TEST (a)</u>	<u>CHANNEL CALIBRATION(a)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
4. <u>AUTOMATIC DEPRESSURIZATION SYSTEM#</u>					
a. Reactor Vessel Water Level - Low Low Low, Level 1				1, 2, 3	
b. Drywell Pressure - High				1, 2, 3	
c. ADS Timer	N.A.			1, 2, 3	
d. Core Spray Pump Discharge Pressure - High				1, 2, 3	
e. RHR LPCI Mode Pump Discharge Pressure - High				1, 2, 3	
f. Reactor Vessel Water Level - Low, Level 3				1, 2, 3	
g. Manual Initiation	N.A.		N.A.	1, 2, 3	
h. ADS Drywell Pressure Bypass Timer	N.A.			1, 2, 3	
5. <u>LOSS OF POWER</u>					
a. 4.16 kV Emergency Bus Under voltage (Loss of Voltage)##	N.A.		N.A.	1, 2, 3, 4**, 5**	
b. 4.16 kV Emergency Bus Under- voltage (Degraded Voltage)				1, 2, 3, 4**, 5**	

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

* When the system is required to be OPERABLE per Specification 3.5.2.

** Required OPERABLE when ESF equipment is required to be OPERABLE.

*** Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 200 psig.

Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 100 psig.

Loss of Voltage Relay 127-11X is not field settable.

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

- a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 24 hours.
- c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:
 1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 24 hours, or, if this action will initiate a pump trip, declare the trip system inoperable.
 2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.
- d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.
- e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each of the required ATWS recirculation pump trip system instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each of the required end-of-cycle recirculation pump trip system instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST, including trip system logic testing, and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit in accordance with the Surveillance Frequency Control Program. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested in accordance with the Surveillance Frequency Control Program. The measured time shall be added to the most recent breaker arc suppression time and the resulting END-OF-CYCLE-RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be verified to be within its limit.

4.3.4.2.4 The time interval necessary for breaker arc suppression from energization of the recirculation pump circuit breaker trip coil shall be measured in accordance with the Surveillance Frequency Control Program.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

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

ACTION:

- a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each of the required RCIC system actuation instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program. CHANNEL CHECK and CHANNEL CALIBRATION are not required for manual initiation.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

- a. With a control rod block instrumentation channel trip setpoint** less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod block trip systems and instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS shown in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition, provided at least one other operable channel in the same trip system is monitoring that parameter.

**The APRM Simulated Thermal Power - Upscale Functional Unit need not be declared inoperable upon entering single reactor recirculation loop operation provided that the flow-biased setpoints are adjusted within 6 hours per Specification 3.4.1.1.

TABLE 4.3.6-1
CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>TRIP FUNCTION</u>	<u>CHANNEL CHECK (h)</u>	<u>CHANNEL FUNCTIONAL TEST (h)</u>	<u>CHANNEL CALIBRATION(a)(h)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>	
1. <u>ROD BLOCK MONITOR</u>					
a. Upscale	N.A.	(c)		1*	
b. Inoperative	N.A.	(c)	N.A.	1*	
c. Downscale	N.A.	(c)		1*	
2. <u>APRM</u>					
a. Simulated Thermal Power - Upscale	N.A.			1	
b. Inoperative	N.A.		N.A.	1, 2	
c. Neutron Flux - Downscale	N.A.			1	
d. Simulated Thermal Power - Upscale (Setdown)	N.A.			2	
e. Recirculation Flow - Upscale	N.A.			1	
f. LPRM Low Count	N.A.			1, 2	
3. <u>SOURCE RANGE MONITORS</u>					
a. Detector not full in	N.A.	(e)	N.A.	2, 5	
b. Upscale	N.A.	(e)		2, 5	
c. Inoperative	N.A.	(e)	N.A.	2, 5	
d. Downscale	N.A.	(e)		2, 5	
4. <u>INTERMEDIATE RANGE MONITORS</u>					
a. Detector not full in	N.A.		N.A.	2, 5**	
b. Upscale	N.A.			2, 5**	
c. Inoperative	N.A.		N.A.	2, 5**	
d. Downscale	N.A.			2, 5**	
5. <u>SCRAM DISCHARGE VOLUME</u>					
a. Water Level - High	N.A.			1, 2, 5**	
6. <u>DELETED</u>					
7. <u>REACTOR MODE SWITCH SHUTDOWN POSITION</u>	N.A.	(g)	N.A.	3, 4	

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) Deleted.
- (c) Includes reactor manual control multiplexing system input.
- * For OPERATIONAL CONDITION of Specification 3.1.4.3.
- ** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
- *** Deleted.
- (d) Deleted
- (e) The provisions of Specification 4.0.4 are not applicable provided that the surveillance is performed within 12 hours after the IRMs are on Range 2 or below during a shutdown.
- (f) Deleted
- (g) The provisions of Specification 4.0.4 are not applicable provided that the surveillance is performed within 1 hour after the Reactor Mode Switch has been placed in the shutdown position.
- (h) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

- a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.
- b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the conditions shown in Table 4.3.7.1-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.1-1.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENTATION</u>	<u>CHANNEL CHECK(c)</u>	<u>CHANNEL FUNCTIONAL TEST(c)</u>	<u>CHANNEL CALIBRATION(c)</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED</u>	
1. Main Control Room Normal Fresh Air Supply Radiation Monitor				1, 2, 3, and *	
2. Area Monitors					
a. Criticality Monitors					
1) Spent Fuel Storage Pool				(a)	
b. Control Room Direct Radiation Monitor				At All Times	
3. Reactor Enclosure Cooling Water Radiation Monitor			(b)	At All Times	

TABLE 4.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

*When RECENTLY IRRADIATED FUEL is being handled in the secondary containment or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel.

- (a) With fuel in the spent fuel storage pool.
- (b) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.
- (c) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

- a. With the number of OPERABLE remote shutdown system instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.
- b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-1, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK* and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.7.4.2 Each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) in accordance with the Surveillance Frequency Control Program.

* Control is not required to be transferred to perform the CHANNEL CHECK.

INFORMATION ON THIS PAGE HAS BEEN DELETED

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.5 The accident monitoring instrumentation channels shown in Table 3.3.7.5-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3.7.5-1.

ACTION:

With one or more accident monitoring instrumentation channels inoperable, take the ACTION required by Table 3.3.7.5-1.

SURVEILLANCE REQUIREMENTS

4.3.7.5 Each of the above required accident monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.5-1.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK (a)</u>	<u>CHANNEL CALIBRATION (a)</u>	
1. Reactor Vessel Pressure			
2. Reactor Vessel Water Level			
3. Suppression Chamber Water Level			
4. Suppression Chamber Water Temperature			
5. Suppression Chamber Air Temperature			
6. Primary Containment Pressure			
7. Drywell Air Temperature			
8. Deleted			
9. Deleted			
10. Deleted			
11. Primary Containment Post LOCA Radiation Monitors		**	
12. North Stack Wide Range Accident Monitor***			
13. Neutron Flux			

(a) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

**CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

***High range noble gas monitors.

INSTRUMENTATION

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

- a. In OPERATIONAL CONDITION 2*, three.
- b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*#, 3, and 4.

ACTION:

- a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least three source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.
- b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

- a. Performance of a:
 - 1. CHANNEL CHECK in accordance with the Surveillance Frequency Control Program:
 - a) in CONDITION 2*, and
 - b) in CONDITION 3 or 4.
 - 2. CHANNEL CALIBRATION** in accordance with the Surveillance Frequency Control Program.
- b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.
- c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 3.0 cps*** with the detector fully inserted.#

*With IRM's on range 2 or below in CONDITION 2.

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

***May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1.

#During initial startup test program, SRM detectors may be partially withdrawn prior to IRM on-scale indication provided that the SRM channels remain on scale above 100 cps and respond to changes in the neutron flux.

INSTRUMENTATION

CHLORINE DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8.1 Two independent chlorine detection system subsystems shall be OPERABLE with their alarm and trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 0.5 ppm

APPLICABILITY: ALL OPERATIONAL CONDITIONS.

ACTION:

- a. With one chlorine detection subsystem inoperable, restore the inoperable detection system to OPERABLE status within 7 days or, within the next 6 hours, initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.
- b. With both chlorine detection subsystem inoperable, within 1 hour initiate and maintain operation of at least one control room emergency filtration system subsystem in the chlorine isolation mode of operation.

SURVEILLANCE REQUIREMENTS

4.3.7.8.1 Each of the above required chlorine detection system subsystems shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

INSTRUMENTATION

TOXIC GAS DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.8.2 Three independent toxic gas detection system subsystems shall be OPERABLE with their alarm setpoints adjusted to actuate at a toxic gas concentration of less than or equal to:

<u>CHEMICAL</u>	<u>MONITOR SET POINT (ppm)</u>
Ammonia	25
Ethylene Oxide	50
Formaldehyde	5
Vinyl Chloride	10
Phosgene	0.4

APPLICABILITY: All OPERATIONAL CONDITIONS.

ACTION:

- With one toxic gas detection subsystem inoperable, place the inoperable subsystem in the tripped condition within 24 hours.
- With two toxic gas detection system subsystems inoperable, place one inoperable subsystem in the tripped condition within 1 hour, restore one inoperable detection subsystem to OPERABLE status within 7 days, or initiate and maintain operation of at least one control room emergency filtration subsystem in the chlorine isolation mode of operation.
- With three toxic gas detection subsystems inoperable, within 1 hour initiate and maintain operation of at least one control room emergency filtration subsystem in the chlorine isolation mode of operation.

SURVEILLANCE REQUIREMENTS

4.3.7.8.2 Each of the above required toxic gas detection system subsystems shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION in accordance with the Surveillance Frequency Control Program.

INSTRUMENTATION

OFFGAS GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The offgas monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.2.5 and 3.11.2.6 respectively, are not exceeded.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

- a. With an offgas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.12-1.
- b. With less than the minimum number of offgas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Annual Radioactive Effluent Release Report.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each offgas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.7.12-1.

TABLE 4.3.7.12-1

OFFGAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK (5)</u>	<u>SOURCE CHECK (5)</u>	<u>CHANNEL CALIBRATION(5)</u>	<u>CHANNEL FUNCTIONAL TEST (5)</u>	<u>MODES IN WHICH SURVEILLANCE IS REQUIRED</u>	
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM						
a. Hydrogen Monitor		N.A.	(3)		**	
2. (Deleted)						
3. (Deleted)						
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR (STEAM JET AIR EJECTOR)						
a. Noble gas activity monitor			(2)	(1)	**	
5. (Deleted)						

TABLE 4.3.7.12-1 (Continued)

TABLE NOTATIONS

- * (Deleted)
- ** During operation of the main condenser steam jet air ejector and offgas treatment system.
- *** (Deleted)
- (1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:
 - 1. Instrument indicates measured levels above the alarm/trip setpoint.
 - 2. Circuit failure.
 - 3. Instrument indicates a downscale failure.
 - 4. Instrument controls not set in operate mode.
- (2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST), previously National Bureau of Standards, or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.
- (3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:
 - 1. 0.0 volume percent hydrogen, balance nitrogen, and
 - 2. 4 volume percent hydrogen, balance nitrogen.
- (4) (Deleted)
- (5) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

INSTRUMENTATION

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The feedwater/main turbine trip system actuation instrumentation channels shown in the Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

- a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.
- b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.
- c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.9.1 Each of the required feedwater/main turbine trip system actuation instrumentation channels shall be demonstrated OPERABLE* by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies specified in the Surveillance Frequency Control Program.

4.3.9.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed in accordance with the Surveillance Frequency Control Program.

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition.

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REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

- a. Reactor THERMAL POWER is $\leq 76.2\%$ of RATED THERMAL POWER,
- b. The recirculation flow control system is in the Local Manual mode, and
- c. The speed of the operating recirculation pump is $\leq 90\%$ of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is $\leq 30\%$ of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is $\leq 50\%$ of rated loop flow.

- a. $\leq 145^{\circ}\text{F}$ between reactor vessel steam space coolant and bottom head drain line coolant,
- b. $\leq 50^{\circ}\text{F}$ between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and
- c. $\leq 50^{\circ}\text{F}$ between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2 All jet pumps shall be demonstrated OPERABLE as follows:

- a. During two recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.
 1. The indicated recirculation loop flow differs by more than 10% from the established* pump speed-loop flow characteristics.
 2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from recirculation loop flow measurements.
 3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from the established* patterns by more than 10%.

*During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

SURVEILLANCE REQUIREMENTS (Continued)

- b. During single recirculation loop operation, each of the above required jet pumps shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that no two of the following conditions occur:
 - 1. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* pump speed-loop flow characteristics.
 - 2. The indicated total core flow differs by more than 10% from the established* total core flow value derived from single recirculation loop flow measurements.
 - 3. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established* single recirculation loop patterns by more than 10%.
- c. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER and upon entering single recirculation loop operation.

*During the startup test program, data shall be recorded for the parameters listed to provide a basis for establishing the specified relationships. Comparisons of the actual data in accordance with the criteria listed shall commence upon the conclusion of the startup test program.

REACTOR COOLANT SYSTEM

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

- a. 5% of each other with core flow greater than or equal to 70% of rated core flow.
- b. 10% of each other with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With the recirculation loop flows different by more than the specified limits, either:

- a. Restore the recirculation loop flows to within the specified limit within 2 hours, or
- b. Shutdown one of the recirculation loops within the next 8 hours and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program. |

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

- 4 safety/relief valves @ 1170 psig $\pm 3\%$
- 5 safety/relief valves @ 1180 psig $\pm 3\%$
- 5 safety/relief valves @ 1190 psig $\pm 3\%$

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. DELETED
- c. DELETED

SURVEILLANCE REQUIREMENTS

4.4.2.1 DELETED

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations in accordance with the Surveillance Frequency Control Program. All safety valves will be recertification tested to meet a $\pm 1\%$ tolerance prior to returning the valves to service.

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

Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

REACTOR COOLANT SYSTEM

ACTIONS (Continued)

- F. With any other two or more leak detection systems inoperable other than ACTION E above OR with required Actions and associated Completion Time of ACTIONS A, B, C, D or E not met, be in HOT SHUTDOWN within 12 hours AND in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated operable by:

- a. Perform a CHANNEL CHECK of the primary containment atmosphere gaseous radioactivity monitoring system in accordance with the Surveillance Frequency Control Program.
- b. Perform a CHANNEL FUNCTIONAL TEST or required leakage detection instrumentation in accordance with the Surveillance Frequency Control Program. This does not apply to containment pressure and temperature monitoring system.
- c. Perform a CHANNEL CALIBRATION of required leakage detection instrumentation in accordance with the Surveillance Frequency Control Program. This does not apply to containment pressure and temperature monitoring system.
- d. Monitor primary containment pressure AND primary containment temperature in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

- a. Monitoring the primary containment atmospheric gaseous radioactivity in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),
- b. Monitoring the drywell floor drain sump and drywell equipment drain tank flow rate in accordance with the Surveillance Frequency Control Program,
- c. Monitoring the drywell unit coolers condensate flow rate in accordance with the Surveillance Frequency Control Program,
- d. Monitoring the primary containment pressure in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),
- e. Monitoring the reactor vessel head flange leak detection system in accordance with the Surveillance Frequency Control Program, and
- f. Monitoring the primary containment temperature in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

- a. In accordance with the Surveillance Frequency Control Program, and
- b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints set less than the specified allowable values by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS FREQUENCY</u>	<u>OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS IS REQUIRED</u>
1. (Deleted)		
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	In accordance with the Surveillance Frequency Control Program	1
3. (Deleted)		
4. Isotopic Analysis for Iodine	a) At least once per 4 hours whenever the specific activity exceeds a limit, as required by ACTION b. b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1**, 2**, 3**, 4** 1, 2
5. Isotopic Analysis of an Off- gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135, and Kr-88	In accordance with the Surveillance Frequency Control Program	1

**Until the specific activity of the primary coolant system is restored to within its limits.

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period,
- c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
- d. The reactor vessel flange and head flange temperature greater than or equal to 70°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curves A, B or C as applicable, in accordance with the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and in accordance with the Surveillance Frequency Control Program during system heatup.

4.4.6.1.3 DELETED

4.4.6.1.4 DELETED

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

- a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:
 1. $\leq 100^{\circ}\text{F}$, in accordance with the Surveillance Frequency Control Program.
 2. $\leq 90^{\circ}\text{F}$, in accordance with the Surveillance Frequency Control Program.
- b. Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during tensioning of the reactor vessel head bolting studs.

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1053 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1053 psig, reduce the pressure to less than 1053 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1053 psig in accordance with the Surveillance Frequency Control Program.

*Not applicable during anticipated transients.

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two (2) independent RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one (1) RHR shutdown cooling subsystem shall be in operation. * ** ***

Each independent RHR shutdown cooling subsystem shall consist of at least:

- a. One OPERABLE RHR pump, and
- b. One OPERABLE RHR heat exchanger, not common to the two (2) independent subsystems.

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

- a. With less than the above required independent RHR shutdown cooling subsystems OPERABLE, immediately initiate corrective action to return the required independent subsystems to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, verify the availability of at least one alternate method capable of decay heat removal for each inoperable independent RHR shutdown cooling subsystem. Be in at least COLD SHUTDOWN within 24 hours.****
- b. With no independent RHR shutdown cooling subsystem in operation, immediately initiate corrective action to return at least one (1) independent subsystem to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.1 At least one independent RHR shutdown cooling subsystem or alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

*One independent RHR shutdown cooling subsystem may be inoperable for up to 2 hours for surveillance testing provided the other independent subsystem is OPERABLE and in operation.

**The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period provided the other independent subsystem is OPERABLE.

***The independent RHR shutdown cooling subsystem may be removed from operation during hydrostatic testing.

****Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two (2) RHR shutdown cooling subsystems shall be OPERABLE, and with no recirculation pump in operation, at least one (1) RHR shutdown cooling subsystem shall be in operation. * ** ***

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION: #

- a. With one (1) or two (2) RHR shutdown cooling subsystems inoperable:
 1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.
- b. With no RHR shutdown cooling subsystems in operation and no recirculation pump in operation:
 1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulating by an alternate method; and
 2. Once per hour monitor reactor coolant temperature and pressure.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one (1) RHR shutdown cooling subsystem or recirculation pump is operating or an alternate method shall be determined to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to two (2) hours per eight hour (8) period.

** One (1) RHR shutdown cooling subsystem may be inoperable for up to two (2) hours for the performance of Surveillances.

*** The shutdown cooling subsystem may be removed from operation during hydrostatic testing.

Separate Action entry is allowed for each shutdown cooling subsystem.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

- a. In accordance with the Surveillance Frequency Control Program:
 1. For the CSS, the LPCI system, and the HPCI system:
 - a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
 - b) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.
 2. For the LPCI system, verifying that both LPCI system subsystem cross-tie valves (HV-51-282 A, B) are closed with power removed from the valve operators.
 3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.
 4. For the CSS and LPCI system, performance of a CHANNEL FUNCTIONAL TEST of the injection header ΔP instrumentation.
- b. Verifying that, when tested pursuant to Specification 4.0.5:
 1. Each CSS pump in each subsystem develops a flow of at least 3175 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥ 105 psid plus head and line losses.
 2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥ 20 psid plus head and line losses.
 3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure which corresponds to a reactor vessel pressure of 1040 psig plus head and line losses when steam is being supplied to the turbine at 1040, +13, -120 psig.**
- c. In accordance with the Surveillance Frequency Control Program:
 1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

* Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72-hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. For the HPCI system, verifying that:
 - a) The system develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥ 200 psig plus head and line losses, when steam is being supplied to the turbine at $200 + 15, - 0$ psig.**
 - b) The suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.
 3. Performing a CHANNEL CALIBRATION of the CSS, LPCI, and HPCI system discharge line "keep filled" alarm instrumentation.
 4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be \leq the allowable value of 4.4 psid.
 5. Performing a CHANNEL CALIBRATION of the LPCI header ΔP instrumentation and verifying the setpoint to be \leq the allowable value of 3.0 psid.
- d. For the ADS:
1. In accordance with the Surveillance Frequency Control Program, verify ADS accumulator gas supply header pressure is ≥ 90 psig. |
 2. In accordance with the Surveillance Frequency Control Program: |
 - a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.
 - b) Verify that when tested pursuant to Specification 4.0.5 that each ADS valve is capable of being opened.
 - c) DELETED

** The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If HPCI OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 200 psig within the following 72 hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.*

4.5.2.2 The core spray system shall be determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2a.2.b).

*One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying the water level to be greater than or equal to, as applicable:

- a. 22'0" in accordance with the Surveillance Frequency Control Program. |
- b. 16'0" in accordance with the Surveillance Frequency Control Program. |

4.5.3.2 With the suppression chamber level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, in accordance with the Surveillance Frequency Control Program: |

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

*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

- a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing in accordance with the Primary Containment Leakage Rate Testing Program.
- b. In accordance with the Surveillance Frequency Control Program by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are opened under administrative control as permitted by Specification 3.6.3.
- c. By verifying the primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

* See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 The primary containment air lock shall be demonstrated OPERABLE:

- a. By verifying the seal leakage rate is in accordance with the Primary Containment Leakage Rate Testing Program.
- b. By conducting an overall air lock leakage test in accordance with the Primary Containment Leakage Rate Testing Program.
- c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in the air lock can be opened at a time.***

***Except that the airlock doors need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the airlock doors' interlock is tested within 8 hours after the primary containment has been deinerted and provided the shield door to the airlock is maintained locked closed.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Drywell and suppression chamber internal pressure shall be maintained between -1.0 and +2.0 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell and/or suppression chamber internal pressure outside of the specified limits, restore the internal pressure to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The drywell and suppression chamber internal pressure shall be determined to be within the limits in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS\

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 145°F**.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 145°F**, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The drywell average air temperature shall be the volumetric average of the temperatures at the following locations and shall be determined to be within the limit in accordance with the Surveillance Frequency Control Program:

	<u>Approximate Elevation</u>	<u>Number of Installed Sensors*</u>
a.	330'	3
b.	320'	3
c.	260'	3
d.	248'	6

* At least one reading from each elevation is required for a volumetric average calculation.

** Operation may continue with drywell average air temperature no greater than 148°F for the remainder of the current operating cycle (Cycle 9), or until the next shutdown of sufficient duration to allow for unit cooler fan repairs, whichever comes first.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and suppression chamber purge system may be in operation with the supply and exhaust isolation valves in one supply line and one exhaust line open for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With a drywell and/or suppression chamber purge supply and/or exhaust isolation valve open, except as permitted above, close the valve(s) within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.8 In accordance with the Surveillance Frequency Control Program, verify each primary containment purge valve [18" or 24"] is closed.*, **

* Only required to be met in OPERATIONAL CONDITIONS 1, 2, and 3.

** Not required to be met when the primary containment purge valves are open for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require these valves to be open.

CONTAINMENT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.
- c. With only one suppression chamber water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression chamber water level and/or temperature to be within the limits at least once per 12 hours.
- d. With no suppression chamber water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- e. With the drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

- a. By verifying the suppression chamber water volume to be within the limits in accordance with the Surveillance Frequency Control Program.
- b. In accordance with the Surveillance Frequency Control Program by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:
 1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.
 2. At least once per hour when suppression chamber average water temperature is greater than or equal to 95°F, by verifying:
 - a) Suppression chamber average water temperature to be less than or equal to 110°F, and
 - b) THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER 12 hours after suppression chamber average water temperature has exceeded 95°F for more than 24 hours.
 3. At least once per 30 minutes following a scram with suppression chamber average water temperature greater than or equal to 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. By verifying at least 8 suppression pool water temperature indicators in at least 8 locations, OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program with the temperature alarm setpoint for:
 - 1. High water temperature:
 - a) First setpoint $\leq 95^{\circ}\text{F}$
 - b) Second setpoint $\leq 105^{\circ}\text{F}$
 - c) Third setpoint $\leq 110^{\circ}\text{F}$
 - d) Fourth setpoint $\leq 120^{\circ}\text{F}$
- d. By verifying at least two suppression chamber water level indicators OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program with the water level alarm setpoint for high water level $\leq 24'1\text{-}1/2"$.
- e. Drywell-to-suppression chamber bypass leak tests shall be conducted to coincide with the Type A test at an initial differential pressure of 4 psi and verifying that the A/\sqrt{k} calculated from the measured leakage is within the specified limit. If any drywell-to-suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 24 months until two consecutive tests meet the specified limit, at which time the test schedule may be resumed.
- f. By conducting a leakage test on the drywell-to-suppression chamber vacuum breakers at a differential pressure of at least 4.0 psi and verifying that the total leakage area A/\sqrt{k} contributed by all vacuum breakers is less than or equal to 24% of the specified limit and the leakage area for an individual set of vacuum breakers is less than or equal to 12% of the specified limit. The vacuum breaker leakage test shall be conducted during each refueling outage for which the drywell-to-suppression chamber bypass leak test in Specification 4.6.2.1.e is not conducted.

CONTAINMENT SYSTEMS

SUPPRESSION POOL SPRAY

LIMITING CONDITION FOR OPERATION

3.6.2.2 The suppression pool spray mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

- a. One OPERABLE RHR pump, and
- b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger and the suppression pool spray sparger(s).

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one suppression pool spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With both suppression pool spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN* within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The suppression pool spray mode of the RHR system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- b. By verifying that each of the required RHR pumps develops a flow of at least 500 gpm on recirculation flow through the RHR heat exchanger and the suppression pool spray sparger when tested pursuant to Specification 4.0.5.

* Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

CONTAINMENT SYSTEMS

SUPPRESSION POOL COOLING

LIMITING CONDITION FOR OPERATION

3.6.2.3 The suppression pool cooling mode of the residual heat removal (RHR) system shall be OPERABLE with two independent loops, each loop consisting of:

- a. One OPERABLE RHR pump, and
- b. An OPERABLE flow path capable of recirculating water from the suppression chamber through an RHR heat exchanger.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN* within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The suppression pool cooling mode of the RHR system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- b. By verifying that each of the required RHR pumps develops a flow of at least 10,000 gpm on recirculation flow through the flow path including the RHR heat exchanger and its associated closed bypass valve, the suppression pool and the full flow test line when tested pursuant to Specification 4.0.5.

*Whenever both RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

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

4.6.3.2 Each primary containment automatic isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

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

4.6.3.4 A representative sample of instrumentation line excess flow check valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program, such that each valve is tested in accordance with the Surveillance Frequency Control Program, by verifying that the valve checks flow.*

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

- a. In accordance with the Surveillance Frequency Control Program by verifying the continuity of the explosive charge.
- b. In accordance with the Surveillance Frequency Control Program by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested in accordance with the Surveillance Frequency Control Program, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and/or operating life, as applicable.

*The reactor vessel head seal leakage detection line (penetration 29A) excess flow check valve is not required to be tested pursuant to this requirement.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

- a. Verified closed in accordance with the Surveillance Frequency Control Program.
- b. Demonstrated OPERABLE:
 1. In accordance with the Surveillance Frequency Control Program and within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.
 2. In accordance with the Surveillance Frequency Control Program by verifying both position indicators OPERABLE by observing expected valve movement during the cycling test.
 3. In accordance with the Surveillance Frequency Control Program by:
 - a) Verifying each valve's opening setpoint, from the closed position, to be 0.5 psid \pm 5%, and
 - b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.
 - c) Verifying that each outboard valve's position indicator is capable of detecting disk displacement ≥ 0.050 ", and each inboard valve's position indicator is capable of detecting disk displacement ≥ 0.120 ".

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY, restore REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1.1 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

- a. Verifying in accordance with the Surveillance Frequency Control Program that the pressure within the reactor enclosure secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.
- b. Verifying in accordance with the Surveillance Frequency Control Program that:
 1. All reactor enclosure secondary containment equipment hatches and blowout panels are closed and sealed.
 2. At least one door in each access to the reactor enclosure secondary containment is closed.
 3. All reactor enclosure secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.
- c. In accordance with the Surveillance Frequency Control Program:
 1. Verifying that one standby gas treatment subsystem will draw down the reactor enclosure secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 916 seconds with the reactor enclosure recirc system in operation, and
 2. Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the reactor enclosure secondary containment at a flow rate not exceeding 2500 cfm with wind speeds of ≤ 7.0 mph as measured on the wind instrument on Tower 1, elevation 30' or, if that instrument is unavailable, Tower 2, elevation 159'.

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

REFUELING AREA SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, suspend handling of RECENTLY IRRADIATED FUEL in the secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

- a. Verifying in accordance with the Surveillance Frequency Control Program that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.
- b. Verifying in accordance with the Surveillance Frequency Control Program that:
 1. All refueling area secondary containment equipment hatches and blowout panels are closed and sealed.
 2. At least one door in each access to the refueling area secondary containment is closed.
 3. All refueling area secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers/valves and required to be closed during accident conditions are closed by valves, blind flanges, slide gate dampers or deactivated automatic dampers/valves secured in position.
- c. In accordance with the Surveillance Frequency Control Program:

Operating one standby gas treatment subsystem for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the refueling area secondary containment at a flow rate not exceeding 764 cfm.

CONTAINMENT SYSTEMS

REACTOR ENCLOSURE SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.1 The reactor enclosure secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one or more of the reactor secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

- a. Restore the inoperable valves to OPERABLE status, or
- b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or
- c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

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

SURVEILLANCE REQUIREMENTS

4.6.5.2.1 Each reactor enclosure secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

- a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.
- b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.
- c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

REFUELING AREA SECONDARY CONTAINMENT AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2.2 The refueling area secondary containment ventilation system automatic isolation valves shall be OPERABLE.

APPLICABILITY: When RECENTLY IRRADIATED FUEL is being handled in the secondary containment, or during operations with a potential for draining the reactor vessel, with the vessel head removed and fuel in the vessel.

ACTION:

With one or more of the refueling area secondary containment ventilation system automatic isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

- a. Restore the inoperable valves to OPERABLE status, or
- b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or
- c. Isolate each affected penetration by use of at least one closed manual valve, blind flange or slide gate damper.

Otherwise, suspend handling of RECENTLY IRRADIATED FUEL in the refueling area secondary containment, and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2.2 Each refueling area secondary containment ventilation system automatic isolation valve shall be demonstrated OPERABLE:

- a. Prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.
- b. In accordance with the Surveillance Frequency Control Program by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.
- c. By verifying the isolation time to be within its limit in accordance with the Surveillance Frequency Control Program.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each standby gas treatment subsystem shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:
 - 1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 5764 cfm \pm 10%.
 - 2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.
 - 3. Verify that when the fan is running the subsystem flowrate is 2800 cfm minimum from each reactor enclosure (Zones I and II) and 2200 cfm minimum from the refueling area (Zone III) when tested in accordance with ANSI N510-1980.
 - 4. Verify that the pressure drop across the refueling area to SGTS prefilter is less than 0.25 inches water gage while operating at a flow rate of 2400 cfm \pm 10%.
- c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 0.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F), at a relative humidity of 70% and at a face velocity of 66 fpm.
- d. In accordance with the Surveillance Frequency Control Program by:
 - 1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 9.1 inches water gauge while operating the filter train at a flow rate of 8400 cfm \pm 10%.

CONTAINMENT SYSTEMS

REACTOR ENCLOSURE RECIRCULATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent reactor enclosure recirculation subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one reactor enclosure recirculation subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With both reactor enclosure recirculation subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each reactor enclosure recirculation subsystem shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates properly.
- b. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by:
 1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 60,000 cfm \pm 10%.
 2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
 3. Verifying a subsystem flow rate of 60,000 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
- d. In accordance with the Surveillance Frequency Control Program by:
 - 1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the filter train at a flow rate of 60,000 cfm \pm 10%, verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.
 - 2. Verifying that the filter train starts and the isolation valves which take suction on and return to the reactor enclosure open on each of the following test signals:
 - a. Manual initiation from the control room, and
 - b. Simulated automatic initiation signal.
- e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 60,000 cfm \pm 10%.
- f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 60,000 cfm \pm 10%.

CONTAINMENT SYSTEMS

DRYWELL HYDROGEN MIXING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.2 Four independent drywell unit cooler hydrogen mixing subsystems (2AV212, 2BV212, 2GV212, 2HV212) shall be OPERABLE with each subsystem consisting of one unit cooler fan.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.6.2 Each drywell unit cooler hydrogen mixing subsystem shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

- a. Starting the system from the control room, and
- b. Verifying that the system operates for at least 15 minutes.

CONTAINMENT SYSTEMS

DRYWELL AND SUPPRESSION CHAMBER OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.6.6.3 The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume.

APPLICABILITY: OPERATIONAL CONDITION 1*, during the time period:

- a. Within 24 hours** after THERMAL POWER is greater than 15% of RATED THERMAL POWER, following startup, to
- b. Within 24 hours** prior to reducing THERMAL POWER to less than 15% of RATED THERMAL POWER, preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.3 The drywell and suppression chamber oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is greater than 15% of RATED THERMAL POWER and in accordance with the Surveillance Frequency Control Program thereafter.

*See Special Test Exception 3.10.5.

**Specification 3.6.1.8 is applicable during this 24 hour period.

PLANT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

5. With two RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.
 6. With three RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.
 7. With four RHRSW pump/diesel generator pairs* inoperable, restore at least one inoperable RHRSW pump/diesel generator pair* to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In OPERATIONAL CONDITION 3 or 4 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.4.9.1 or 3.4.9.2, inoperable, declare the associated RHR loop inoperable and take the ACTION required by Specification 3.4.9.1 or 3.4.9.2, as applicable.
 - c. In OPERATIONAL CONDITION 5 with the RHRSW subsystem(s), which is associated with an RHR loop required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required residual heat removal service water system subsystem(s) shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*A RHRSW pump/diesel generator pair consists of a RHRSW pump and its associated diesel generator. If either a RHRSW pump or its associated diesel generator becomes inoperable, then the RHRSW pump/diesel generator pair is inoperable.

PLANT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

4. With three ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
 5. With four ESW pump/diesel generator pairs** inoperable, restore at least one inoperable ESW pump/diesel generator pair** to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. In OPERATIONAL CONDITION 4 or 5:
1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or declare the associated safety related equipment inoperable and take the ACTION required by Specifications 3.5.2 and 3.8.1.2.
- c. In OPERATIONAL CONDITION *
1. With only one emergency service water pump and its associated flow path OPERABLE, restore at least two pumps with at least one flow path to OPERABLE status within 72 hours or verify adequate cooling remains available for the diesel generators required to be OPERABLE or declare the associated diesel generator(s) inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENT

4.7.1.2 At least the above required emergency service water system loop(s) shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) that is not locked, sealed, or otherwise secured in position, is in its correct position. |
- b. In accordance with the Surveillance Frequency Control Program by verifying that: |
 1. Each automatic valve actuates to its correct position on its appropriate ESW pump start signal.
 2. Each pump starts automatically when its associated diesel generator starts.

* When handling irradiated fuel in the secondary containment.

** An ESW pump/diesel generator pair consists of an ESW pump and its associated diesel generator. If either an ESW pump or its associated diesel generator becomes inoperable, then the ESW pump/diesel generator pair is inoperable.

PLANT SYSTEMS

ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The spray pond shall be OPERABLE with:

- a. A minimum pond water level at or above elevation 250'-10" Mean Sea Level, and
- b. A pond water temperature of less than or equal to 88°F.

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

ACTION:

With the requirements of the above specification not satisfied:

- a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
- b. In OPERATIONAL CONDITION 4 or 5, declare the RHRSW system and the emergency service water system inoperable and take the ACTION required by Specifications 3.7.1.1 and 3.7.1.2.
- c. In OPERATIONAL CONDITION *, declare the emergency service water system inoperable and take the ACTION required by Specification 3.7.1.2. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The spray pond shall be determined OPERABLE:

- a. By verifying the pond water level to be greater than its limit in accordance with the Surveillance Frequency Control Program. |
- b. By verifying the water surface temperature (within the upper two feet of the surface) to be less than or equal to 88°F:
 1. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than or equal to 80°F; and |
 2. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than or equal to 85°F; and |
 3. in accordance with the Surveillance Frequency Control Program when the spray pond temperature is greater than 32°F. |
- c. By verifying all piping above the frost line is drained:
 1. within one (1) hour after being used when ambient air temperature is below 40°F; or
 2. when ambient air temperature falls below 40°F if the piping has not been previously drained.

*When handling irradiated fuel in the secondary containment.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.2 Each control room emergency fresh air supply subsystem shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying the control room air temperature to be less than or equal to 85°F effective temperature. |
- b. In accordance with the Surveillance Frequency Control Program on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates with the heaters OPERABLE. |
- c. In accordance with the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the subsystem by: |
 - 1. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 3000 cfm \pm 10%.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
 3. Verifying a subsystem flow rate of 3000 cfm \pm 10% during subsystem operation when tested in accordance with ANSI N510-1980.
- d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration of less than 2.5% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and a relative humidity of 70%.
- e. In accordance with the Surveillance Frequency Control Program by:
1. Verifying that the pressure drop across the combined prefilter, upstream and downstream HEPA filters, and charcoal adsorber banks is less than 6 inches water gauge while operating the subsystem at a flow rate of 3000 cfm \pm 10%; verifying that the prefilter pressure drop is less than 0.8 inch water gauge and that the pressure drop across each HEPA is less than 2 inches water gauge.
 2. Verifying that on each of the below chlorine isolation mode actuation test signals, the subsystem automatically switches to the chlorine isolation mode of operation and the isolation valves close within 5 seconds:
 - a) Outside air intake high chlorine, and
 - b) Manual initiation from the control room.
 3. Verifying that on each of the below radiation isolation mode actuation test signals, the subsystem automatically switches to the radiation isolation mode of operation and the control room is maintained at a positive pressure of at least 1/8 inch water gauge relative to the turbine enclosure and auxiliary equipment room and outside atmosphere during subsystem operation with an outdoor air flow rate less than or equal to 525 cfm:
 - a) Outside air intake high radiation, and
 - b) Manual initiation from control room.

PLANT SYSTEMS

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

- a. With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.
- b. In the event the RCIC system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.
- c. Specification 3.0.4.b is not applicable to RCIC.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by:
 1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
 2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
 3. Verifying that the pump flow controller is in the correct position.
- b. In accordance with the Surveillance Frequency Control Program by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1040 + 13, - 120 psig.*

* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 150 psig within the following 72 hours.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. In accordance with the Surveillance Frequency Control Program by:
1. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded.
 2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, - 0 psig.*
 3. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level-low signal.
 4. Performing a CHANNEL CALIBRATION of the RCIC system discharge line "keep filled" level alarm instrumentation.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests. If OPERABILITY is not successfully demonstrated within the 12-hour period, reduce reactor steam dome pressure to less than 150 psig within the following 72 hours.

PLANT SYSTEMS

3/4.7.5 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.5 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

- a. With a sealed source having removable contamination in excess of the above limit, withdraw the sealed source from use and either:
 1. Decontaminate and repair the sealed source, or
 2. Dispose of the sealed source in accordance with Commission Regulations.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

- a. The licensee, or
- b. Other persons specifically authorized by the Commission or an Agreement State.

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

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

- a. Sources in use - In accordance with the Surveillance Frequency Control Program for all sealed sources containing radioactive material:
 1. With a half-life greater than 30 days, excluding Hydrogen 3, and
 2. In any form other than gas.

PLANT SYSTEMS

3/4.7.8 MAIN TURBINE BYPASS SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 The main turbine bypass system shall be OPERABLE as determined by the number of operable main turbine bypass valves being greater than or equal to that specified in the CORE OPERATING LIMITS REPORT.

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

ACTION: With the main turbine bypass system inoperable, restore the system to OPERABLE status within 1 hour or take the ACTION required by Specification 3.2.3.c.

SURVEILLANCE REQUIREMENTS

4.7.8 The main turbine bypass system shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program:

- a. By cycling each turbine bypass valve through at least one complete cycle of full travel,
- b. By performing a system functional test which includes simulated automatic actuation, and by verifying that each automatic valve actuates to its correct position, and
- c. By determining TURBINE BYPASS SYSTEM RESPONSE TIME to be less than or equal to the value specified in the CORE OPERATING LIMITS REPORT.

ELECTRICAL POWER SYSTEMS \

SURVEILLANCE REQUIREMENTS

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

- a. Determined OPERABLE in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignments and indicated power availability, and
- b. Demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by transferring, manually and automatically, unit power supply from the normal circuit to the alternate circuit.

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

- a. In accordance with the frequency specified in Table 4.8.1.1.2-1 on a STAGGERED TEST BASIS by:
 1. Verifying the fuel level in the day fuel tank.
 2. Verifying the fuel level in the fuel storage tank.
 3. Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day fuel tank.
 4. Verify that the diesel can start* and gradually accelerate to synchronous speed with generator voltage and frequency at 4280 ± 120 volts and 60 ± 1.2 Hz.
 5. Verify diesel is synchronized, gradually loaded* to an indicated 2700-2800 kW** and operates with this load for at least 60 minutes.
 6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.
 7. Verifying the pressure in all diesel generator air start receivers to be greater than or equal to 225 psig.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. By removing accumulated water:
 - 1) From the day tank in accordance with the Surveillance Frequency Control Program and after each occasion when the diesel is operated for greater than 1 hour, and
 - 2) From the storage tank in accordance with the Surveillance Frequency Control Program.
- c. By sampling new fuel oil in accordance with ASTM D4057-81 prior to addition to the storage tanks and:
 - 1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:
 - a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.
 - b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.
 - c) A flash point equal to or greater than 125°F, and
 - d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.
 - 2) By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.
- d. In accordance with the Surveillance Frequency Control Program by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A, except that the filters specified in ASTM D2276-78, Sections 5.1.6 and 5.1.7, may have a nominal pore size of up to three (3) microns.
- e. In accordance with the Surveillance Frequency Control Program by:
 - 1) Deleted
 - 2) Verifying each diesel generator's capability to reject a load of greater than or equal to that of its single largest post-accident load while maintaining voltage at 4285 ± 420 volts and frequency at 60 ± 1.2 hz and after steady state conditions are reached, voltage is maintained at 4280 ± 120 volts.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the diesel generator capability to reject a load of 2850 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection. |
4. Simulating a loss-of-offsite power by itself, and: |
 - a) Verifying deenergization of the emergency buses and load shedding from the emergency buses.
 - b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency buses within 10 seconds, energizes the auto-connected loads through the individual load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency buses shall be maintained at 4280 ± 120 volts and 60 ± 1.2 Hz during this test.
5. Verifying that on an ECCS actuation test signal, without loss-of-offsite power, the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall reach 4280 ± 120 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. |
6. Simulating a loss-of-offsite power in conjunction with an ECCS actuation test signal, and: |
 - a) Verifying deenergization of the emergency buses and load shedding from the emergency buses.
 - b) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency buses within 10 seconds, energizes the auto-connected shutdown loads through the individual load timers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency buses shall be maintained at 4280 ± 120 volts and 60 ± 1.2 Hz during this test.
7. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential over-current are automatically bypassed upon an ECCS actuation signal. |

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warm up procedures, and as applicable regarding loading and shutdown recommendations.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

8. a) Verifying the diesel generator operates* for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to an indicated 2950-3050 kW** and during the remaining 22 hours of this test, the diesel generator shall be loaded to an indicated 2700-2800 kW**.
- b) Verifying that, within 5 minutes of shutting down the diesel generator after the diesel generator has operated* for at least 2 hours at an indicated 2700-2800 kW**, the diesel generator starts*. The generator voltage and frequency shall reach 4280 ± 120 volts and 60 ± 1.2 Hz within 10 seconds after the start signal.
9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3100 kW.
10. Verifying the diesel generator's capability to:
 - a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
 - b) Transfer its loads to the offsite power source, and
 - c) Be restored to its standby status.
11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to standby operation, and (2) automatically energizes the emergency loads with offsite power.
12. Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within $\pm 10\%$ of its design interval.

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:
 - a) Control Room Switch In Pull-To-Lock (With Local/Remote Switch in Remote)
 - b) Local/Remote Switch in Local.
 - c) Emergency Stop
- f. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting* all four diesel generators simultaneously, during shutdown, and verifying that all four diesel generators accelerate to at least 882 rpm in less than or equal to 10 seconds.
- g. In accordance with the Surveillance Frequency Control Program by:
 1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and
 2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section XI Article IWD-5000.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- h. In accordance with the Surveillance Frequency Control Program the diesel generator shall be started* and verified to accelerate to synchronous speed in less than or equal to 10 seconds. The generator voltage and frequency shall reach 4280 ± 120 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The diesel generator shall be started for this test by using one of the following signals:

- a) Manual***
- b) Simulated loss-of-offsite power by itself.
- c) Simulated loss-of-offsite power in conjunction with an ECCS actuation test signal.
- d) An ECCS actuation test signal by itself.

The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 2700-2800 KW** in less than or equal to 200 seconds, and operate for at least 60 minutes. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

*This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading and shutdown recommendations.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

***If diesel generator started manually from the control room, 10 seconds after the automatic prelube period.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN
LAST 20 VALID DEMANDS*

TEST FREQUENCY

less than or equal to 1 or,

In accordance with the
Surveillance Frequency
Control Program

greater than or equal to 2**

In accordance with the
Surveillance Frequency
Control Program

*Criteria for determining number of failures and number of valid demands shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 demands have been reduced to one.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required division batteries and chargers shall be demonstrated OPERABLE:

- a. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. Each Division 1 and 2 battery float current is ≤ 2 amps, and Division 3 and 4 battery float current is ≤ 1 amp when battery terminal voltage is greater than or equal to the minimum established float voltage of 4.8.2.1.a.2, and
 2. Total battery terminal voltage for each 125-volt battery is greater than or equal to the minimum established float voltage.
- b. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. Each battery pilot cell voltage is ≥ 2.07 volts,
 2. Each battery connected cell electrolyte level is greater than or equal to minimum established design limits, and
 3. The electrolyte temperature of each pilot cell is greater than or equal to minimum established design limits.
- c. In accordance with the Surveillance Frequency Control Program by verifying that each battery connected cell voltage is ≥ 2.07 volts.
- d. In accordance with the Surveillance Frequency Control Program by verifying that:
 1. The battery chargers will supply the currents listed below at greater than or equal to the minimum established float voltage for at least 4 hours:

<u>Charger</u>	<u>Current (Amperes)</u>
2BCA1	300
2BCA2	300
2BCB1	300
2BCB2	300
2BCC	75
2BCD	75

2. The battery capacity is adequate to supply and maintain in OPERABLE status the required emergency loads for the design duty cycle when subjected to a battery service test.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- e. In accordance with the Surveillance Frequency Control Program by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test or modified performance discharge test. The modified performance discharge test may be performed in lieu of the battery service test (Specification 4.8.2.1.d.2).
- f. Performance discharge tests or modified performance discharge tests of battery capacity shall be given as follows:
 - 1. In accordance with the Surveillance Frequency Control Program when:
 - (a) The battery shows degradation or
 - (b) The battery has reached 85% of expected life with battery capacity < 100% of manufacturer's rating, and
 - 2. In accordance with the Surveillance Frequency Control Program when battery has reached 85% of expected life with battery capacity \geq 100% of manufacturer's rating.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

- a. With one of the above required Unit 2 A.C. distribution system divisions not energized, reenergize the division within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- b. With one of the above required Unit 2 D.C. distribution system divisions not energized, reenergize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
- c. With any of the above required Unit 1 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each of the above required power distribution system divisions shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- b. With less than two divisions of the above required Unit 2 D.C. distribution systems energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.
- c. With any of the above required Unit 1 and common AC and/or DC distribution system divisions not energized, declare the associated common equipment inoperable, and take the appropriate ACTION for that system.
- d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2 At least the above required power distribution system divisions shall be determined energized in accordance with the Surveillance Frequency Control Program by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

ELECTRICAL POWER SYSTEMS

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of all Class 1E motor operated valves shall be either:

- a. Continuously bypassed for all valves with maintained position control switches; or,
- b. Bypassed only under accident conditions for all valves with spring-return-to-normal control switches.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously or only under accident conditions, as applicable, restore the thermal overload bypass within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION statement(s) for the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves which are continuously bypassed and temporarily placed in force only when the valve motor is undergoing periodic or maintenance testing shall be verified to be bypassed following periodic or maintenance testing during which the thermal overload protection was temporarily placed in force.

4.8.4.2.2 In accordance with the Surveillance Frequency Control Program, a CHANNEL FUNCTIONAL TEST of all those valves which are bypassed only under accident conditions (valves with spring-return-to-normal control switches) shall be performed to verify that the thermal overload protection will be bypassed under accident conditions.

ELECTRICAL POWER SYSTEMS

REACTOR PROTECTION SYSTEM ELECTRICAL POWER MONITORING

LIMITING CONDITION FOR OPERATION

3.8.4.3 Two reactor protection system (RPS) electric power monitoring channels for each inservice RPS Inverter or alternate power supply shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

- a. With one RPS electric power monitoring channel for an inservice RPS Inverter or alternate power supply inoperable, restore the inoperable power monitoring channel to OPERABLE status within 72 hours or remove the associated RPS Inverter or alternate power supply from service.
- b. With both RPS electric power monitoring channels for an inservice RPS Inverter or alternate power supply inoperable, restore at least one electric power monitoring channel to OPERABLE status within 24 hours or remove the associated RPS Inverter or alternate power supply from service.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above specified RPS electric power monitoring channels shall be determined OPERABLE:

- a. By performance of a CHANNEL FUNCTIONAL TEST each time the plant is in COLD SHUTDOWN for a period of more than 24 hours, unless performed in the previous 6 months.
- b. In accordance with the Surveillance Frequency Control Program by demonstrating the OPERABILITY of overvoltage, undervoltage, and underfrequency 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 Allowable Values.
 1. Overvoltage ≤ 127.6 VAC,
 2. Undervoltage ≥ 110.7 VAC,
 3. Underfrequency ≥ 57.05 Hz.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

4.9.1.1 The reactor mode switch shall be verified to be locked in the Shutdown or Refuel position as specified in accordance with the Surveillance Frequency Control Program.

4.9.1.2 Each of the above required reactor mode switch Refuel position interlocks* shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program during control rod withdrawal or CORE ALTERATIONS, as applicable.

4.9.1.3 Each of the above required reactor mode switch Refuel position interlocks* that is affected shall be demonstrated OPERABLE by performance of a CHANNEL FUNCTIONAL TEST prior to resuming control rod withdrawal or CORE ALTERATIONS, as applicable, following repair, maintenance or replacement of any component that could affect the Refuel position interlock.

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

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

- a. Continuous visual indication in the control room,
- b. At least one with audible alarm in the control room,
- c. One of the required SRM detectors located in the quadrant where CORE ALTERATIONS are being performed and the other required SRM detector located in an adjacent quadrant, and
- d. Unless adequate SHUTDOWN MARGIN has been demonstrated, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.**

APPLICABILITY: OPERATIONAL CONDITION 5.***

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

*These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core. The use of special movable detectors during CORE ALTERATIONS in place of the normal SRM nuclear detectors is permissible as long as these special detectors are connected to the normal SRM circuits.

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

***See Special Test Exception, Specification 3/4.10.7.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

- b. Performance of a CHANNEL FUNCTIONAL TEST in accordance with the Surveillance Frequency Control Program.
- c. Verifying that the channel count rate is at least 3.0 cps: *
 - 1. Prior to control rod withdrawal,
 - 2. Prior to and in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS, and
 - 3. In accordance with the Surveillance Frequency Control Program.
- d. Verifying, within 8 hours prior to and in accordance with the Surveillance Frequency Control Program, that the RPS circuitry "shorting links" have been removed during:
 - 1. The time any control rod is withdrawn**, unless adequate shutdown margin has been demonstrated, or
 - 2. Shutdown margin demonstrations.

*May be reduced, provided the source range monitor has an observed count rate and signal-to-noise ratio on or above the curve shown in Figure 3.3.6-1. These channels are not required when sixteen or fewer fuel assemblies, adjacent to the SRMs, are in the core.

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

REFUELING OPERATIONS

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be inserted.*

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.**

ACTION:

With all control rods not inserted, suspend all other CORE ALTERATIONS, except that one control rod may be withdrawn under control of the reactor mode switch Refuel position one-rod-out interlock.

SURVEILLANCE REQUIREMENTS

4.9.3 All control rods shall be verified to be inserted, except as above specified in accordance with the Surveillance Frequency Control Program. |

*Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**See Special Test Exception 3.10.3.

REFUELING OPERATIONS \

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 5, during CORE ALTERATIONS.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communication between the control room and refueling floor personnel shall be demonstrated in accordance with the Surveillance Frequency Control Program during CORE ALTERATIONS.*

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

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1200 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks which prevent crane travel over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and in accordance with the Surveillance Frequency Control Program during crane operation.

REFUELING OPERATIONS

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth in accordance with the Surveillance Frequency Control Program during handling of fuel assemblies or control rods within the reactor pressure vessel.

REFUELING OPERATIONS

3/4.9.9 WATER LEVEL - SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 22 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth in accordance with the Surveillance Frequency Control Program.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

- a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.
- b. The SRM channels are OPERABLE per Specification 3.9.2.
- c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1c.
- d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.
- e. All other control rods are inserted.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and in accordance with the Surveillance Frequency Control Program thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

- a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.
- b. The SRM channels are OPERABLE per Specification 3.9.2.
- c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.
- d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.
- e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

REFUELING OPERATIONS

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

- 3.9.11.1 One (1) RHR shutdown cooling subsystem shall be OPERABLE and in operation. *

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet above the top of the reactor pressure vessel flange.

ACTION:

- a. With the required RHR shutdown cooling subsystem inoperable:
 1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available.
- b. With the required action and associated completion time of Action "a" above not met.
 1. Immediately suspend loading of irradiated fuel assemblies into the reactor pressure vessel; and
 2. Immediately initiate action to restore REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY to OPERABLE status; and
 3. Immediately initiate action to restore one (1) Standby Gas Treatment subsystem to OPERABLE status; and
 4. Immediately initiate action to restore isolation capability in each required Refueling Floor secondary containment penetration flow path not isolated.
- c. With no RHR shutdown cooling subsystem in operation:
 1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and
 2. Once per hour monitor reactor coolant temperature.

SURVEILLANCE REQUIREMENTS

- 4.9.11.1 At least one (1) RHR shutdown cooling subsystem, or an alternate method, shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* The required RHR shutdown cooling subsystem may be removed from operation for up to two (2) hours per eight (8) hour period.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two (2) RHR shutdown cooling subsystems shall be OPERABLE, and one (1) RHR shutdown cooling subsystem shall be in operation. *

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet above the top of the reactor pressure vessel flange.

ACTION:

- a. With one (1) or two (2) required RHR shutdown cooling subsystems inoperable:
 1. Within one (1) hour, and once per 24 hours thereafter, verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.
- b. With the required action and associated completion time of Action "a" above not met:
 1. Immediately initiate action to restore REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY to OPERABLE status; and
 2. Immediately initiate action to restore one (1) Standby Gas Treatment subsystem to OPERABLE status; and
 3. Immediately initiate action to restore isolation capability in each required Refueling Floor secondary containment penetration flow path not isolated.
- c. With no RHR shutdown cooling subsystem in operation:
 1. Within one (1) hour from discovery of no reactor coolant circulation, and once per 12 hours thereafter, verify reactor coolant circulation by an alternate method; and
 2. Once per hour monitor reactor coolant temperature.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one (1) RHR shutdown cooling subsystem, or an alternate method, shall be verified to be in operation and circulating reactor coolant in accordance with the Surveillance Frequency Control Program.

* The required operating shutdown cooling subsystem may be removed from operation for up to two (2) hours per eight (8) hour period.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: OPERATIONAL CONDITION 2, during low power PHYSICS TESTS.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.1 The THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during low power PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.3 SHUTDOWN MARGIN DEMONSTRATIONS

LIMITING CONDITION FOR OPERATION

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

- a. The source range monitors are OPERABLE with the RPS circuitry "shorting links" removed per Specification 3.9.2.
- b. The rod worth minimizer is OPERABLE per Specification 3.1.4.1 and is programmed for the shutdown margin demonstration, or conformance with the shutdown margin demonstration procedure is verified by a second licensed operator or other technically qualified member of the unit technical staff.
- c. The "continuous rod withdrawal" control shall not be used during out-of-sequence movement of the control rods.
- d. No other CORE ALTERATIONS are in progress.

APPLICABILITY: OPERATIONAL CONDITION 5, during shutdown margin demonstrations.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.3 Within 30 minutes prior to and in accordance with the Surveillance Frequency Control Program during the performance of a shutdown margin demonstration, verify that;

- a. The source range monitors are OPERABLE per Specification 3.9.2,
- b. The rod worth minimizer is OPERABLE with the required program per Specification 3.1.4.1 or a second licensed operator or other technically qualified member of the unit technical staff is present and verifies compliance with the shutdown margin demonstration procedures, and
- c. No other CORE ALTERATIONS are in progress.

SPECIAL TEST EXCEPTIONS

3/4.10.4 RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.4.1.1 and 3.4.1.3 that recirculation loops be in operation may be suspended for up to 24 hours for the performance of:

- a. PHYSICS TESTS, provided that THERMAL POWER does not exceed 5% of RATED THERMAL POWER, or
- b. The Startup Test Program.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2, during PHYSICS TESTS and the Startup Test Program.

ACTION:

- a. With the above specified time limit exceeded, insert all control rods.
- b. With the above specified THERMAL POWER limit exceeded during PHYSICS TESTS, immediately place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The time during which the above specified requirement has been suspended shall be verified to be less than 24 hours in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS and the Startup Test Program.

4.10.4.2 THERMAL POWER shall be determined to be less than 5% of RATED THERMAL POWER in accordance with the Surveillance Frequency Control Program during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 OXYGEN CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.10.5 The provisions of Specification 3.6.6.3 may be suspended until completion of the Startup Test Program or the reactor has operated for 120 Effective Full Power Days.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION

With the requirements of the above specification not satisfied, be in at least STARTUP within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.5 The Effective Full Power Days of operation shall be verified to be less than 120, by calculation, in accordance with the Surveillance Frequency Control Program during the Startup Test Program.

SPECIAL TEST EXCEPTIONS

3/4.10.6 TRAINING STARTUPS

LIMITING CONDITION FOR OPERATION

3.10.6 The provisions of Specification 3.5.1 may be suspended to permit one RHR subsystem to be aligned in the shutdown cooling mode during training startups provided that the reactor vessel is not pressurized, THERMAL POWER is less than or equal to 1% of RATED THERMAL POWER and reactor coolant temperature is less than 200°F.

APPLICABILITY: OPERATIONAL CONDITION 2, during training startups.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.10.6 The reactor vessel shall be verified to be unpressurized and the THERMAL POWER and reactor coolant temperature shall be verified to be within the limits in accordance with the Surveillance Frequency Control Program during training startups. |

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

- a. With the quantity of radioactive material in any of the above tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents in accordance with the Surveillance Frequency Control Program when radioactive materials are being added to the tank. |

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.6 The rate of the sum of the activities of the noble gases Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, and Xe-138 measured at the recombiner after-condenser discharge shall be limited to less than or equal to 330 millicuries/second.

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

ACTION:

With the rate of the sum of the activities of the specified noble gases at the recombiner after-condenser discharge exceeding 330 millicuries/second, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.6.1 The rate of the sum of the activities of noble gases at the recombiner after-condenser discharge shall be continuously monitored in accordance with Specification 3.3.7.12.

4.11.2.6.2 The rate of the sum of the activities of the specified noble gases from the recombiner after-condenser discharge shall be determined to be within the limits of Specification 3.11.2.6 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the recombiner after condenser discharge:

- a. In accordance with the Surveillance Frequency Control Program.
- b. Within 4 hours following an increase, as indicated by the Main Condenser Off-Gas Pretreatment Radioactivity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level or air in-leakage, in the nominal steady-state fission gas release from the primary coolant.
- c. The provisons of Specification 4.0.4 are not applicable.

*When the main condenser air ejector is in operation.

REACTIVITY CONTROL SYSTEMS

BASES

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillances are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3. Additional pertinent analysis is also contained in Amendment 17 to the Reference 4 Topical Report.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density over the range of power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. RBM OPERABILITY is required when the limiting condition described in Specification 3.1.4.3 exists.

REACTIVITY CONTROL SYSTEMS

BASES

STANDBY LIQUID CONTROL SYSTEM (Continued)

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume assures that the solution is available for use.

Replacement of the explosive charges in the valves will assure that these valves will not fail because of deterioration of the charges.

The Standby Liquid Control System also has a post-DBA LOCA safety function to buffer Suppression Pool pH in order to maintain bulk pH above 7.0. The buffering of Suppression Pool pH is necessary to prevent iodine re-evolution to satisfy the methodology for Alternative Source Term. Manual initiation is used, and the minimum amount of total boron required for Suppression Pool pH buffering is 240 lbs. Given that at least 185 lbs of Boron-10 is maintained in the tank, the total boron in the tank will be greater than 240 lbs for the range of enrichments from 29% to 62%.

ACTION Statement (a) applies only to OPERATIONAL CONDITIONS 1 and 2 because a single pump can satisfy both the reactor control function and the post-DBA LOCA function to control Suppression Pool pH since boron injection is not required until 13 hours post-LOCA. ACTION Statement (b) applies to OPERATIONAL CONDITIONS 1, 2 and 3 to address the post-LOCA safety function of the SLC system.

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1. C. J. Paone, R. C. Stirn and J. A. Woolley, "Rod Drop Accident Analysis for Large BWR's," G. E. Topical Report NEDO-10527, March 1972.
 2. C. J. Paone, R. C. Stirn, and R. M. Young, Supplement 1 to NEDO-10527, July 1972.
 3. J. M. Haun, C. J. Paone, and R. C. Stirn, Addendum 2, "Exposed Cores," Supplement 2 to NEDO-10527, January 1973.
 4. Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel".
 5. "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Limerick Generating Station Units 1 and 2," NEDC-32193P, Revision 2, October 1993.

3/4.3 INSTRUMENTATION\

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

- a. Preserve the integrity of the fuel cladding.
- b. Preserve the integrity of the reactor coolant system.
- c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and
- d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The APRM system is divided into four APRM channels and four 2-Out-Of-4 Voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed.

The system meets the intent of IEEE-279 for nuclear power plant protection systems. Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System" and NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function." The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The APRM Functions include five Functions accomplished by the four APRM channels (Functions 2.a, 2.b, 2.c, 2.d, and 2.f) and one accomplished by the four 2-Out-Of-4 Voter channels (Function 2.e). Two of the five Functions accomplished by the APRM channels are based on neutron flux only (Functions 2.a and 2.c), one Function is based on neutron flux and recirculation drive flow (Function 2.b) and one is based on equipment status (Function 2.d). The fifth Function accomplished by the APRM channels is the Oscillation Power Range Monitor (OPRM) Upscale trip Function 2.f, which is based on detecting oscillatory characteristics in the neutron flux. The OPRM Upscale Function is also dependent on average neutron flux (Simulated Thermal Power) and recirculation drive flow, which are used to automatically enable the output trip.

The Two-Out-Of-Four Logic Module includes 2-Out-Of-4 Voter hardware and the APRM Interface hardware. The 2-Out-Of-4 Voter Function 2.e votes APRM Functions 2.a, 2.b, 2.c, and 2.d independently of Function 2.f. This voting is accomplished by the 2-Out-Of-4 Voter hardware in the Two-Out-Of-Four Logic Module. The voter includes separate outputs to RPS for the two independently voted sets of Functions, each of which is redundant (four total outputs). The analysis in Reference 2 took credit for this redundancy in the justification of the 12-hour allowed out-of-service time for

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION (continued)

core flow relationship. The drive flow / core flow relationship is established once per refuel cycle, while operating within 10% of rated core flow and within 10% of RATED THERMAL POWER. Plant operational experience has shown that this flow correlation methodology is consistent with the guidance and intent in Reference 8. Changes throughout the cycle in the drive flow / core flow relationship due to the changing thermal hydraulic operating conditions of the core are accounted for in the margins included in the bases or analyses used to establish the setpoints for the APRM Simulated Thermal Power - Upscale Function and the OPRM Upscale Function.

As noted in Table 3.3.1-2, Note "*", the redundant outputs from the 2-Out-Of-4 Voter channel are considered part of the same channel, but the OPRM and APRM outputs are considered to be separate channels, so N = 8 to determine the interval between tests for application of Specification 4.3.1.3 (REACTOR PROTECTION SYSTEM RESPONSE TIME). The note further requires that testing of OPRM and APRM outputs shall be alternated.

Each test of an OPRM or APRM output tests each of the redundant outputs from the 2-Out-Of-4 Voter channel for that function, and each of the corresponding relays in the RPS. Consequently, each of the RPS relays is tested every fourth cycle. This testing frequency is twice the frequency justified by References 2 and 3.

Automatic reactor trip upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A.C. Thadani, NRC, dated May 15, 1991).

The measurement of response time at the frequencies specified in the Surveillance Frequency Control Program provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable except for the APRM Simulated Thermal Power - Upscale and Neutron Flux - Upscale trip functions and the OPRM Upscale trip function (Table 3.3.1-2, Items 2.b, 2.c, and 2.f). Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times. Response time testing for the sensors as noted in Table 3.3.1-2 is not required based on the analysis in NEDO-32291-A. Response time testing for the remaining channel components is required as noted. For the digital electronic portions of the APRM functions, performance characteristics that determine response time are checked by a combination of automatic self-test, calibration activities, and response time tests of the 2-Out-Of-4 Voter (Table 3.3.1-2, Item 2.e).

INSTRUMENTATION

BASES

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30851P, Supplement 2, "Technical Specification Improvement Analysis for BWR Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (SER) (letter to D.N. Grace from C.E. Rossi dated January 6, 1989) and NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," as approved by the NRC and documented in the NRC SER (letter to S.D. Floyd from C.E. Rossi dated June 18, 1990).

Automatic closure of the MSIVs upon receipt of a high-high radiation signal from the Main Steam Line Radiation Monitoring System was removed as the result of an analysis performed by General Electric in NEDO-31400A. The NRC approved the results of this analysis as documented in the SER (letter to George J. Beck, BWR Owner's Group from A.C. Thadani, NRC, dated May 15, 1991).

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10-second diesel startup and the 3 second load center loading delay. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions.

Response time testing for sensors are not required based on the analysis in NEDO-32291-A. Response time testing of the remaining channel components is required as noted in Table 3.3.2-3.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. Primary containment isolation valves that are actuated by the isolation signals specified in Technical Specification Table 3.3.2-1 are identified in Technical Requirements Manual Table 3.6.3-1.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

INSTRUMENTATION

BASES

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (Continued)

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with NEDC-30936P, Parts 1 and 2, "Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)," as approved by the NRC and documented in the SER (letter to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1) and letter to D. N. Grace from C. E. Rossi dated December 9, 1988 (Part 2)).

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power for energizing various components such as pump motors, motor operated valves, and the associated control components. If the loss of power instrumentation detects that voltage levels are too low, the buses are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources. The loss of power relays in each channel have sufficient overlapping detection characteristics and functionality to permit operation subject to the conditions in Action Statement 37. Bases 3/4.8.1, 3/4.8.2, and 3/4.8.3 provide discussion regarding parametric bounds for determining operability of the offsite sources. Those Bases assume that the loss of power relays are operable. With an inoperable 127Z-11X0X relay, the grid voltage is monitored to 230kV (for the 101 Safeguard Bus Source) or 525kV (for the 201 Safeguard Bus Source) to increase the margin for the operation of the 127Z-11X0X relay.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

INSTRUMENTATION

BASES

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been specified in accordance with recommendations made by GE in their letter to the BWR Owner's Group dated August 7, 1989, SUBJECT: "Clarification of Technical Specification changes given in ECCS Actuation Instrumentation Analysis."

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage time have been determined in accordance with NEDC-30851P, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," as approved by the NRC and documented in the SER (letter to D. N. Grace from C. E. Rossi dated September 22, 1988).

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

INSTRUMENTATION

BASES

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variable following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

The surveillance interval for the Main Control Room Normal Fresh Air Supply Radiation Monitor is determined in accordance with the Surveillance Frequency Control Program.

3/4.3.7.2 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE UFSAR.

3/4.3.7.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50, Appendix A. The Unit 1 RHR transfer switches are included only due to their potential impact on the RHRSW system, which is common to both units.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

INSTRUMENTATION

BASES

3/4.3.7.7 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

There are three toxic gas detection subsystems. The high toxic chemical concentration alarm in the Main Control Room annunciates when two of the three subsystems detect a high toxic gas concentration. An Operate/Inop keylock switch is provided for each subsystem which allows an individual subsystem to be placed in the tripped condition. Placing the keylock switch in the INOP position initiates one of the two inputs required to initiate the alarm in the Main Control Room.

Surveillance intervals are determined in accordance with the Surveillance Frequency Control Program and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R.D. Binz, IV, from C.E. Rossi dated July 21, 1992).

3/4.3.7.9 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE TRM.

REACTOR COOLANT SYSTEM

BASES

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 12 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown. The safety/relief valves will be removed and either set pressure tested or replaced with spares which have been previously set pressure tested and stored in accordance with manufacturers recommendations at the frequency specified in the Surveillance Frequency Control Program.

REACTOR COOLANT SYSTEM

BASES

ACTIONS (Continued)

- C. With the required primary containment air cooler condensate flow rate monitoring system inoperable, SR 4.4.3.1.a must be performed every 8 hours to provide periodic information of activity in the primary containment at a more frequent interval than the routine frequency of every 12 hours. The 8 hour interval provides periodic information that is adequate to detect leakage and recognizes that other forms of leakage detection are available. The required ACTION has been clarified to state that the additional surveillance requirement is not applicable if the required primary containment atmosphere gaseous radioactivity monitoring system is also inoperable. Consistent with SR 4.0.3, surveillances are not required to be performed on inoperable equipment. In this case, ACTION Statement A. and E. requirements apply.
- D. With the primary containment pressure and temperature monitoring system inoperable, operation may continue for up to 30 days given the system's indirect capability to detect RCS leakage. However, other more limiting Tech Spec requirements associated with the primary containment pressure/temperature monitoring system will still apply.
- E. With both the primary containment atmosphere gaseous radioactivity monitor and the primary containment air cooler condensate flow rate monitor inoperable, the only means of detecting leakage is the drywell floor drain sump monitor and the drywell pressure/temperature instrumentation. This condition does not provide the required diverse means of leakage detection. The required ACTION is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period. While the primary containment atmosphere gaseous radioactivity monitor is INOPERABLE, Primary containment atmospheric grab samples will be taken and analyzed every 12 hours since ACTION Statement A. requirements also apply.
- F. If any required ACTION of Conditions A, B, C, D or E cannot be met within the associated Completion Time, the plant must be brought to an OPERATIONAL CONDITION in which the LCO does not apply. To achieve this status, the plant must be brought to at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the next 24 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the ACTIONS in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 4.4.3.1.a

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmospheric monitoring system. The check gives reasonable confidence that the channel is operating properly.

REACTOR COOLANT SYSTEM

BASES

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.4.3.1.b

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string.

SR 4.4.3.1.c

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment.

SR 4.4.3.1.d

This SR provides a routine check of primary containment pressure and temperature for indirect evidence of RCS leakage.

REFERENCES

1. LGS UFSAR, Section 5.2.5.1.
2. Regulatory Guide 1.45, May 1973.
3. LGS UFSAR, Section 5.2.5.2.1.3
4. LGS UFSAR, Section 5.2.5.2.1.5
5. LGS UFSAR, Section 5.2.5.2.1.4
6. LGS UFSAR, Section 5.2.5.2.1.1(2)
7. GEAP-5620, April 1968.
8. NUREG-75/067, October 1975.
9. LGS UFSAR, Section 5.2.5.6.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. The limit of 2 gpm increase in UNIDENTIFIED LEAKAGE over a 24-hour period and the monitoring of drywell floor drain sump and drywell equipment drain tank flow rate at least once every eight (8) hours conforms with NRC staff positions specified in NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," as revised by NRC Safety Evaluation dated March 6, 1990. The ACTION requirement for the 2 gpm increase in UNIDENTIFIED LEAKAGE limit ensures that such leakage is identified or a plant shutdown is initiated to allow further investigation and corrective action. Once identified, reactor operation may continue dependent upon the impact on total leakage.

EMERGENCY CORE COOLING SYSTEM

BASES

ECCS - OPERATING and SHUTDOWN (Continued)

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the CS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system. The HPCI system, and one LPCI subsystem, and/or one CSS subsystem out-of-service period of 8 hours ensures that sufficient ECCS, comprised of a minimum of one CSS subsystem, three LPCI subsystems, and all of the ADS will be available to 1) provide for safe shutdown of the facility, and 2) mitigate and control accident conditions within the facility. A Note prohibits the application of Specification 3.0.4.b to an inoperable HPCI subsystem. There is an increased risk associated with entering an OPERATIONAL CONDITION or other specified condition in the Applicability with an inoperable HPCI subsystem and the provisions of Specification 3.0.4.b, which allow entry into an OPERATIONAL CONDITION or other specified condition in the Applicability with the Limiting Condition for Operation not met after performance of a risk assessment addressing inoperable systems and components, should not be applied in this circumstance.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety/relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls five selected safety-relief valves. The safety analysis assumes all five are operable. The allowed out-of-service time for one valve for up to fourteen days is determined in a similar manner to other ECCS sub-system out-of-service time allowances.

Verification that ADS accumulator gas supply header pressure is ≥ 90 psig ensures adequate gas pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator at least two valve actuations can occur with the drywell at 70% of design pressure. The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. This minimum required pressure of ≥ 90 psig is provided by the PCIG supply.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum calculated pressure in the event of a LOCA. A visual inspection in accordance with the Primary Containment Leakage Rate Testing Program is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the calculated containment peak pressure does not exceed the design pressure of 55 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 5.0 psid. The limit of - 1.0 to + 2.0 psig for initial containment pressure will limit the total pressure to ≤ 44 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during steam line break conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. Limiting the use of the drywell and suppression chamber purge system to specific criteria is imposed to protect the integrity of the SGTS filters. Analysis indicates that should a LOCA occur while this pathway is being utilized, the associated pressure surge through the (18 or 24") purge lines will adversely affect the integrity of SGTS. This condition is not imposed on the 1 and 2 inch valves used for pressure control since a surge through these lines does not threaten the operability of SGTS.

Surveillance requirement 4.6.1.8 ensures that the primary containment purge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. The SR is modified by a Note stating that primary containment purge valves are only required to be closed in OPERATIONAL CONDITIONS 1, 2 and 3. The SR is also modified by a Note stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, deinerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. The 18 or 24 inch purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time.

CONTAINMENT SYSTEMS

BASES

DEPRESSURIZATION SYSTEMS (Continued)

The drywell-to-suppression chamber bypass test at a differential pressure of at least 4.0 psi verifies the overall bypass leakage area for simulated LOCA conditions is less than the specified limit. For those outages where the drywell-to-suppression chamber bypass leakage test is not conducted, the VB leakage test verifies that the VB leakage area is less than the bypass limit, with a 76% margin to the bypass limit to accommodate the remaining potential leakage area through the passive structural components. Previous drywell-to-suppression chamber bypass test data indicates that the bypass leakage through the passive structural components will be much less than the 76% margin. The VB leakage limit, combined with the negligible passive structural leakage area, ensures that the drywell-to-suppression chamber bypass leakage limit is met for those outages for which the drywell-to-suppression chamber bypass test is not scheduled.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A of 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The scram discharge volume vent and drain valves serve a dual function, one of which is primary containment isolation. Since the other safety functions of the scram discharge volume vent and drain valves would not be available if the normal PCIV actions were taken, actions are provided to direct the user to the scram discharge volume vent and drain operability requirements contained in Specification 3.1.3.1. However, since the scram discharge volume vent and drain valves are PCIVs, the Surveillance Requirements of Specification 4.6.3 still apply to these valves.

The opening of a containment isolation valve that was locked or sealed closed to satisfy Technical Specification 3.6.3 Action statements, may be reopened on an intermittent basis under administrative controls. These controls consist of stationing a dedicated individual at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

Primary containment isolation valves governed by this Technical Specification are identified in Table 3.6.3-1 of the TRM.

This Surveillance Requirement requires a demonstration that a representative sample of reactor instrument line excess flow check valves (EFCVs) is OPERABLE by verifying that the valve actuates to the isolation position on a simulated instrument line break signal. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested in accordance with the Surveillance Frequency Control Program. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes, and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. This Surveillance Requirement provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during a postulated instrument line break event. Furthermore, any EFCV failures will be evaluated to determine if additional testing in the test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint. For some EFCVs, this Surveillance can be performed with the reactor at power.

CONTAINMENT SYSTEMS

BASES

3/4.6.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Reactor Enclosure and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the reactor enclosure secondary containment with the standby gas treatment system in accordance with the Surveillance Frequency Control Program, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the reactor enclosure recirculation system and the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY and Control Room radiation doses associated with containment leakage. The operation of these systems and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analysis. Provisions have been made to continuously purge the filter plenums with instrument air when the filters are not in use to prevent buildup of moisture on the adsorbers and the HEPA filters.

As a result of the Alternative Source Term (AST) project, secondary containment integrity of the refueling area is not required during certain conditions when handling irradiated fuel, during CORE ALTERATIONS, or during operations with a potential for draining the reactor vessel with the vessel head removed and fuel in the vessel. The control room dose analysis for the fuel handling accident (FHA) is based on unfiltered releases from the South Stack and therefore, does not require the standby gas treatment system to be aligned to the refueling area whenever irradiated fuel is being handled, during CORE ALTERATIONS or operations are being conducted that have the potential to drain the reactor vessel. The OPERABILITY of the standby gas treatment system however, ensures that release pathways do not result in a control room dose higher than determined in the refueling accident dose analysis. This is accomplished by assuring that the Standby Gas Treatment System is OPERABLE if releases from the refueling area are made through the North Stack following a FHA.

Although the safety analyses assumes that the reactor enclosure secondary containment draw down time will take 930 seconds, these surveillance requirements specify a draw down time of 916 seconds. This 14 second difference is due to the diesel generator starting and sequence loading delays which is not part of this surveillance requirement.

The reactor enclosure secondary containment draw down time analyses assumes a starting point of 0.25 inch of vacuum water gauge and worst case SGTS dirty filter flow rate of 2800 cfm. The surveillance requirements satisfy this assumption by starting the drawdown from ambient conditions and connecting the adjacent reactor enclosure and refueling area to the SGTS to split the exhaust flow between the three zones and verifying a minimum flow rate of 2800 cfm from the test zone. This simulates the worst case flow alignment and verifies adequate flow is available to drawdown the test zone within the required time. The Technical Specification Surveillance Requirement 4.6.5.3.b.3 is intended to be a multi-zone air balance verification without isolating any test zone.

The SGTS is common to Unit 1 and 2 and consists of two independent subsystems. The power supplies for the common portions of the subsystems are from Unit 1 safeguard busses, therefore the inoperability of these Unit 1 supplies are addressed in the SGTS ACTION statements in order to ensure adequate onsite power sources to SGTS for its Unit 2 function during a loss of offsite power event. The allowable out of service times are consistent with those in the Unit 1 Technical Specifications for SGTS and AC electrical power supply out of service condition combinations.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Supplies, March 10, 1971, Regulatory Guide 1.137 "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979 and Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 except for paragraphs C.2.a(3), C.2.c(1), C.2.c(2), C.2.d(3) and C.2.d(4), and the periodic testing will be performed in accordance with the Surveillance Frequency Control Program. The exceptions to Regulatory Guide 1.108 allow for gradual loading of diesel generators during testing and decreased surveillance test frequencies (in response to Generic Letter 84-15). The single largest post-accident load on each diesel generator is the RHR pump.

The Surveillance Requirement for removal of accumulated water from the fuel oil storage tanks is for preventive maintenance. The presence of water does not necessarily represent failure of the Surveillance Requirement, provided the accumulated water is removed during performance of the Surveillance. Accumulated water in the fuel oil storage tanks constitutes a collection of water at a level that can be consistently and reliably measured. The minimum level at which accumulated water can be consistently and reliably measured in the fuel oil storage tank sump is 0.25 inches. Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of accumulated water from the fuel storage tanks once every (31) days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137.

The surveillance requirements for demonstrating the OPERABILITY of the units batteries are in accordance with the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

Verifying battery float current while on float charge (4.8.2.1.a.1) is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450-1995.

This Surveillance Requirement states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of 4.8.2.1.a.2. When this float voltage is not maintained the Actions of LCO 3.8.2.1, Action b., are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limits are established based on the float voltage range and is not directly applicable when this voltage is not maintained.

3/4.8 ELECTRICAL POWER SYSTEMS

BASES

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Verifying, per 4.8.2.1.a.2, battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the minimum float voltage established by the battery manufacturer (2.20 Vpc, average, or 132 V at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years).

Surveillance Requirements 4.8.2.1.b.1 and 4.8.2.1.c require verification that the cell float voltages are equal to or greater than 2.07 V.

The limit specified in 4.8.2.1.b.2 for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability.

Surveillance Requirement 4.8.2.1.b.3 verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., 60 degrees Fahrenheit). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provide the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity.

Surveillance Requirement 4.8.2.1.d.1 verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32, the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

Surveillance Requirement 4.8.2.1.d.1 requires that each battery charger be capable of supplying the amps listed for the specified charger at the minimum established float voltage for 4 hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. This time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least 2 hours.

A battery service test, per 4.8.2.1.d.2, is a special test of the battery's capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in the UFSAR.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

i. Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications," of the following:

- a. Actions to restore battery cells with float voltage < 2.13 volts, and
- b. Actions to equalize and test battery cells that have been discovered with electrolyte level below the minimum established design limit.

j. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The Program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 0.
- c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.