LIMITING CONDITION FOR OPERATION

3.7.6.2 The following spray and sprinkler systems shall be OPERABLE:

Fire Zone	Description
	Reactor Enclosure Hatchway Water Curtains:
	1. EL 253'
	2. EL 283'
	3. EL 313'
	Fire Area Separation Water Curtains:
48A	1. Area 602, EL 313'
45A	2. Area 402, EL 253'
44	3. Area 304, EL 217' (2 curtains)
22	Cable Spreading Room, Room 450, EL 254'
27	Control Structure Fan Room, EL 304'
27	CREFAS System Filters, EL 304'
28B	SGTS Filters, Compartment 624 and SGTS Access Area 625, EL 332'
33	RCIC Pump Room, Room 108, EL 177'
34	HPCI Pump Room, Room 109, EL 177'
41	RECW Area, EL 201'
42A	Safeguard System Access Area 200, EL 201'
44	Safeguard System Access Area 304, EL 217' (Partial) (2 systems)
45A	CRD Hydraulic Equipment Area 402, Reactor Enclosure, EL 253' (Partial)
45B	Neutron Monitoring System Area 406, EL 253' (Partial)
47A	General Equipment Area 500 and Corridor 506, Reactor Enclosure, EL 283' (Partial)
51A & B	Reactor Enclosure Recirculation System Filters, EL 331'
79,80,81,82	Diesel Generator Cells (4 cells)

APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE.

ACTION:

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

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

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the core spray, high pressure coolant injection, reactor core isolation cooling, residual heat removal, post-accident sampling system, safeguard piping fill system, control rod drive scram discharge system, and containment air monitor systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. Reactor Vessel Pressure	2	1	1,2	80
2. Reactor Vessel Water Level	2	1	1,2	80
3. Suppression Chamber Water Level	2	1	1,2	80
4. Suppression Chamber Water Temperature	8,6 locations	6, 1/location	1,2	80
5. Suppression Chamber Air Temperature	1	1	1,2	80
6. Drywell Pressure	2	1	1,2	80
7. Drywell Air Temperature	1	1	1,2	80
8. Drywell Oxygen Concentration Analyzer	2	1	1,2	80
9. Drywell Hydrogen Concentration Analyzer	2	1	1,2	80
10. Safety/Relief Valve Position Indicators	1/valve	1/valve	1,2	80
11. Primary Containment Post-LOCA Radiation Monitors	4	2	1,2,3	81
12. North Stack Wide Range Accident Monitor**	3*	3*	1,2,3	81
13. Neutron Flux	2	1	1,2	80

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1. Low, Low-Level 2	≥ -38 inches*	≥ -45 inches
2. Low, Low, Low, Level 1	≥ -129 inches*	≥ -136 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. North Stack Effluent Radiation - High	≤ 2.1 uCi/cc	≤ 4.0 uCi/cc
d. Refueling Area Ventilation Exhaust Duct - Radiation - High	≤ 2.0 mR/h	≤ 2.2 mR/h
e. Reactor Enclosure Ventilation Exhaust Duct - Radiation - High	≤ 1.35 mR/h	≤ 1.5 mR/h
f. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	≥ 0.1 inches of H2O	≥ 0.0 inches of H2O
g. Outside Atmosphere To Refueling Area Δ Pressure - Low	≥ 0.1 inches of H2O	≥ 0.0 inches of H2O
h. Drywell Pressure - High/ Reactor Pressure - Low	≤ 1.68 psig/ ≥ 455 psig (decreasing)	≤ 1.88 psig/ ≥ 435 psig (decreasing)
i. Primary Containment Instrument Gas to Drywell Δ Pressure-Low	≥ 2.0 psig	≥ 1.9 psig
j. Manual Initiation	N.A.	N.A.

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
7. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low, Low-Level 2	≥ -38 inches*	≥ -45 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Refueling Area Ventilation Exhaust Duct Radiation - High	≤ 2.0 mR/h	≤ 2.2 mR/h
d. Reactor Enclosure Ventilation Exhaust Duct Radiation - High	≤ 1.35 mR/h	≤ 1.5 mR/h
e. Outside Atmosphere To Reactor Enclosure Δ Pressure - Low	≥ 0.1 inches of H2O	≥ 0.0 inches of H2O
f. Outside Atmosphere to Refueling Area Δ Pressure - Low	≥ 0.1 inches of H2O	≥ 0.0 inches of H2O
g. Manual Initiation	N.A.	N.A.

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

**The low setpoints are for the RWCU Heat Exchanger Rooms; the high setpoints are for the pump rooms.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level	
1) Low, Low - Level 2	≤ 13(a)**
2) Low, Low, Low - Level 1	≤ 1.0*/≤ 13(a)**
b. Main Steam Line Radiation - High(b)	≤ 1.0*/≤ 13(a)**
c. Main Steam Line Pressure - Low	≤ 1.0*/≤ 13(a)**
d. Main Steam Line Flow - High	≤ 0.5*/≤ 13(a)**
e. Condenser Vacuum - Low	N.A.
f. Main Steam Line Tunnel Temperature - High	N.A.
g. Turbine Enclosure - Main Steam Line Tunnel Temperature - High	N.A.
h. Manual Initiation	N.A.
2. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	
a. Reactor Vessel Water Level Low - Level 3	≤ 13(a)
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	N.A.
c. Manual Initiation	N.A.
3. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCS Δ Flow - High	≤ 13##
b. RWCS Area Temperature - High	N.A.
c. RWCS Area Ventilation Δ Temperature - High	N.A.
d. SLCS Initiation	N.A.
e. Reactor Vessel Water Level - Low, Low-Level 2	≤ 13(a)
f. Manual Initiation	N.A.

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 curves C and C' within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor flux wire specimens shall be removed at the first refueling outage and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4 4.6-1. The results of these fluence determinations shall be used to adjust the curves of Figure 3.4.6.1-1, as required.

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

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100 degrees Fahrenheit, at least once per 12 hours.

   2. $\leq$ 90 degrees Fahrenheit, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 4 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

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

a. At least once per 31 days:
   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 $\geq$ 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 $\geq$ 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 1000 psig plus head and line losses when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. At least once per 18 months:
   1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

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*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.

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. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.
   2. At least once per 18 months:
      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) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig** and observing that either:
         1) The control valve or bypass valve position responds accordingly, or
         2) There is a corresponding change in the measured steam flow.
      c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying an alarm setpoint of 90 $\pm$ 2 psig on decreasing pressure.

**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 or ADS operability is not successfully demonstrated within the 12 hour period, reduce reactor steam dome pressure to less than 200 psig or 100 psig, respectively.

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 at least once per 12 hours.

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 pressure of 44.02 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations of drywell and suppression chamber internal pressure ensure that the containment peak pressure of 44.02 psig 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 44.02 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 degrees Fahrenheit 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 and pressure control. The 90 hours per 365 day limit on purge valve operation 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 limit is not imposed, however, on the subject valves when pressure control is being performed through the 2-inch bypass line, since a pressure surge through this line does not threaten the OPERABILITY of SGTS.

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Drywell average air temperature shall not exceed 135 degrees Fahrenheit.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the drywell average air temperature greater than 135 degrees Fahrenheit, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

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

	Approximate Elevation	Number of Installed Sensors*
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.

SUPPRESSION CHAMBER

LIMITING CONDITION FOR OPERATION

3.6.2.1 The suppression chamber shall be OPERABLE with:

a. The pool water:
   1. Volume* between 122,120 cubic feet and 134,600 cubic feet, equivalent to a level between 22' 0" and 24' 3", and a
   2. Maximum average temperature of 95 degrees Fahrenheit except that the maximum average temperature may be permitted to increase to:
      a) 105 degrees Fahrenheit during testing which adds heat to the suppression chamber.
      b) 110 degrees Fahrenheit with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.
      c) 120 degrees Fahrenheit with the main steam line isolation valves closed following a scram.

b. Drywell-to-suppression chamber bypass leakage less than or equal to 10% of the acceptable $A/\sqrt{K}$ design value of 0.0500 square feet.

c. At least eight suppression pool water temperature instrumentation indicators, one in each of the eight locations.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

b. With the suppression chamber average water temperature greater than 95 degrees Fahrenheit, restore the average temperature to less than or equal to 95 degrees Fahrenheit within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:
   1. With the suppression chamber average water temperature greater than 105 degrees Fahrenheit during testing which adds heat to the suppression chamber, stop all testing which adds heat to the suppression chamber and restore the average temperature to less than 95 degrees Fahrenheit within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   2. With the suppression chamber average water temperature greater than:
      a) 95 degrees Fahrenheit for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
      b) 110 degrees Fahrenheit, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

*Includes the volume inside the pedestal.

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: OPERATIONAL CONDITION*.

ACTION:

Without REFUELING AREA SECONDARY CONTAINMENT INTEGRITY, 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.

SURVEILLANCE REQUIREMENTS

4.6.5.1.2 REFUELING AREA SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the refueling area secondary containment is greater than or equal to 0.25 inch of vacuum water gauge.

b. Verifying at least once per 31 days 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, or deactivated automatic dampers/valves secured in position.

c. At least once per 18 months:

   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.

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*Required 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.

STANDBY GAS TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent standby gas treatment subsystems shall be OPERABLE.

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

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. In Operational Condition *, 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 in Operational Condition *, 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. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal absorbers and verifying that the subsystem operates with the heaters OPERABLE.

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*Required 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.

TABLE 3.6.3-1 (Continued)

PART A - PRIMARY CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	FUNCTION	INBOARD ISOLATION BARRIER	OUTBOARD ISOLATION BARRIER	MAX ISOL. TIME.IF APP. (SEC)(26)	ISOL. SIGNAL(S), IF APP. (20)	NOTES	FCID
028B	DRYWELL H2/O2 SAMPLE	SV57-133		5	B,H,R,S	11	57
			SV57-143	5	B,H,R,S	11	
			SV57-195	5	B,H,R,S	11	
030S-1	DRYWELL PRESSURE INSTRUMENTATION		HV42-147A	45		10	42
035B	TIP PURGE	59-1056(CK) (DOUBLE "O" RING)		NA			59
			HV59-131	7	B,H,S	16	
035C-G	TIP DRIVES	XV59-141A-E (DOUBLE "O" RING)		NA	B,H	11,16,21	59
			XV59-140A-E	NA		11,16	
037A-D	CRD INSERT LINES	BALL CHECK		NA		12	47
			HCU	NA		12	
038A-D	CRD WITHDRAW LINES SDV VENTS & DRAINS		HCU	NA		12	47
			XV47-1F010	25		30	
			XV47-1F180	30		30	
			XV47-1F011	25		30	
			XV47-1F101	30		30	
039A(B)	DRYWELL SPRAY	HV51-1F021A(B)		160		4,11	51
			HV51-1F016A(B)	160		11	
040E	DRYWELL PRESSURE INSTRUMENTATION		HV42-147D	45		10	42
040F-2	CONTAINMENT INSTRUMENT GAS-SUCTION	HV59-101		45	C,H,S	5	59
			HV59-102	7	C,H,S		

NOTES (Continued)

15. Check valve used instead of flow orifice.
16. Penetration is sealed by a flange with double O-ring seals. These seals are leakage rate tested by pressurizing between the O-rings. Both the TIP Purge Supply (Penetration 35B) and the TIP Drive Tubes (Penetration 35C-G) are welded to their respective flanges. Leakage through these seals is included in the Type C leakage rate total for this penetration. The ball valves (XV-141A-E) are Type C tested. It is not practicable to leak test the shear valves (XV-140A-E) because squib firing is required for closure. Shear valves (XV-140A-G) are normally open.
17. Instrument line isolation provisions consist of an excess flow check valve. Because the instrument line is connected to a closed cooling water system inside containment, no flow orifice is provided. The excess flow check valves are subject to operability testing, but no Type C test is performed nor required. The line does not isolate during a LOCA and can leak only if the line or instrument should rupture. Leaktightness of the line is verified during the integrated leak rate test (Type A test).
18. In addition to double "O" ring seals, this penetration is tested by pressurizing volume between doors per Specification 4.6.1.3.
19. The RHR system safety pressure relief valves will be exempted from the initial LLRT. The relief valves in these lines will be exposed to containment pressure during the initial ILRT and all subsequent ILRTs. In addition, modifications will be performed at the first refueling to facilitate local testing or removal and bench testing of the relief valves during subsequent LLRTs. Those relief valves which are flanged to facilitate removal will be equipped with double O-ring seal assemblies on the flange closest to primary containment by the end of the first refueling outage. These seals will be leak rate tested by pressurizing between the O-rings, and the results added into the Type C total for this penetration.
20. See Specification 3.3.2, Table 3.3.2-1, for a description of the PCRVICS isolation signal(s) that initiate closure of each automatic isolation valve. In addition, the following non-PCRVICS isolation signals also initiate closure of selected valves:

EA	Main steam line high pressure, high steam line leakage flow, low MSIV-LCS dilution air flow
LFHP	With HPCI pumps running, opens on low flow in associated pipe, closes when flow is above setpoint
LFRC	With RCIC pump running, opens on low flow in associated pipe, closes when flow is above setpoint
LFCH	With CSS pump running, opens on low flow in associated pipe, closes when flow is above setpoint
LFCC	Steam supply valve fully closed or RCIC turbine stop valve fully closed

All power operated isolation valves may be opened or closed remote manually.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

b. At least once per 18 months by removing the explosive squib from the explosive valve, such that each explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which as 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.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   4. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with ACTION a.2 or a.3 above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable, operation may continue and the provisions of Specifications 3.0.3 and 3.0.4 are not applicable provided that within 4 hours either:

   1. The inoperable valve is returned to OPERABLE status, or

   2. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under adminstrative control.

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.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:
   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
   2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
   3. Verifying that the pump flow controller is in the correct position.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

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

2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15, -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 the reactor steam pressure to less than 150 psig.

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and
b. Four separate and independent diesel generators, each with:
   1. A separate day tank containing a minimum of 200 gallons of fuel,
   2. A separate fuel storage system containing a minimum of 33,500 gallons of fuel, and
   3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a.and 4.8.1.1.2a.4., for one diesel generator at a time, within 24 hours and at least once per 7 days thereafter; restore the inoperable diesel generator to OPERABLE status within 92 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. With two diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4., for one diesel generator at a time, within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators 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. Also see action e of 3.8.1.1.
c. With three diesel generators of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a. and 4.8.1.1.2a.4., for one diesel generator at a time, within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
d. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter. Restore at least two offsite circuits and at

ANNUAL REPORTS (Continued)

dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. Documentation of all challenges to safety/relief valves; and

c. Any other unit unique reports required on an annual basis.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate), with preoperational studies, operational controls and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of all radiological environmental samples and of all environmental radiation measurements taken during the report period pursuant to the locations specified in the tables and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps.**

* A single submittal may be made for a multiple unit station.
**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

3/4.1.2 REACTIVITY ANOMALIES

LIMITING CONDITON FOR OPERATION

3.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% Δ k/k.

APPLICABILITY: OPERATIONAL CONDITION 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% Δ k/k:

a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

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

SURVEILLANCE REQUIREMENTS

4.1.2 The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% Δ k/k:

a. During the first startup following CORE ALTERATIONS, and

b. At least once per 31 effective full power days during POWER OPERATION.

c. The provisions of Specification 4.0.4 are not applicable.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R	1, 2, 5(i)
b. Float Switch	N.A.	M	R	1, 2, 5(i)
9. Turbine Stop Valve - Closure	N.A.	M	R	1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.	M	R	1
11. Reactor Mode Switch Shutdown Position	N.A.	R	N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.	M	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) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This 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. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow). 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.
(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.

ACTION STATEMENTS

ACTION 100-With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

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

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 101-With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma). Beta is analyzed at a limit of detection of at least 1N7 microcurie/mL. Gamma is analyzed at a limit of detection of at least 5N7 microcurie/mL.

ACTION 102-With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves generated in situ may be used to estimate flow.

TABLE 3.3.7.9-1 (Continued)
FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION				TOTAL NUMBER OF INSTRUMENTS		
FIRE ZONE	STRUCTURE	ELEV.	AREA	HEAT (x/y)	SMOKE (x/y)	FLAME (x/y)
25	Control	289'	Auxiliary Equipment Room 542	0/112 (PGCC Floor) 0/13 (Non-PGCC Floor)	57/0 (Ceiling) 56/0 (PGCC Floor) 14/0 (Non-PGCC Floor) 32/0 (Terminal Cabinets)	NA
26	Control	289'	Remote Shutdown Panel Area 540	0/4 (Non PGCC Floor)	3/0 (Ceiling Level) 2/0 (Non-PGCC Floor)	NA
27	Control	304'	Control Structure Fan Room 619	0/23 4/0 (inside plenum)	10/0	NA
28A	Control	332'	SGTS Access Area 625 (SGTS Room Ventilation Exhaust)	4/0 (inside plenum)	NA	NA
28B	Control	332'	SGTS Filter Compartment 624	4/0 (inside plenum)	NA	NA
28C	Control	332'	Control Room Fresh Air Intake Plenum	NA	3/0	NA
31	Unit 1 Reactor	177'	RHR Heat Exchanger & Pump Room 103 (B&D)	NA	6/0	NA
32	Unit 1 Reactor	177'	RHR Heat Exchanger & Pump Room 102 (A&C)	NA	5/0	NA
33	Unit 1 Reactor	177'	RCIC Pump Room 108	0/3	2/0	NA
34	Unit 1 Reactor	177'	HPCI Pump Room 109	0/4	3/0	NA
35	Unit 1 Reactor	177'	'A' Core Spray Pump Room 110	NA	2/0	NA