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Bases 4.0:

This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operations are met and will be performed during the periods when the Limiting Conditions for Operation are applicable.

A tolerance for performing surveillance activities beyond the nominal interval is provided to allow operational flexibility because of scheduling and performance considerations. Each surveillance test is completed within plus 25% of each scheduled date. Scheduled dates are based on dividing each calendar year into four 13-week "surveillance" quarters consisting of 3 4-week "surveillance" months and one "catch-up" week. This method of scheduling permits certain tests always to be scheduled on certain days of the week.

The specification ensures that surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into a plant condition for which the Limiting Condition for Operation is applicable. Under the terms of this specification, for example, during-initial plant startup or following extended plant outage, the surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment to Operable status.

"Affected equipment" refers to the specific equipment on which a surveillance is being performed. If there is an LCO that corresponds to the specific equipment that has failed the surveillance, then that LCO shall be entered. If there is no corresponding LCO, then the effect of inoperability of the specific equipment that has failed the surveillance shall be evaluated (i.e., by applying the definition of operability) and actions taken as appropriate (e.g., to comply with the technical specifications).

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
<p>3.4 <u>STANDBY LIQUID CONTROL SYSTEM</u></p> <p><u>Applicability:</u> Applies to the operating status of the standby liquid control system.</p> <p><u>Objective:</u> To assure the availability of an independent reactivity control mechanism.</p> <p><u>Specification:</u></p> <p>A. System Operation</p> <ol style="list-style-type: none"> 1. The standby liquid control system shall be operable at all times when fuel is in the reactor and the reactor is not shut down by control rods, except as specified in 3.4.A.2. 2. From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible only during the following 7 days provided that the redundant component is operable. 	<p>4.4 <u>STANDBY LIQUID CONTROL SYSTEM</u></p> <p><u>Applicability:</u> Applies to the periodic testing requirements for the standby liquid control system.</p> <p><u>Objective:</u> To verify the operability of the standby liquid control system.</p> <p><u>Specification:</u></p> <p>A. The operability of the standby liquid control system shall be verified by performance of the following tests:</p> <ol style="list-style-type: none"> 1. <u>At least once per quarter</u> - Pump minimum flow rate of 24 gpm shall be verified against a system head of 1275 psig when tested in accordance with the Inservice Testing Program. Comparison of the measured pump flow rate against equation 2 of paragraph 3.4.B.1 shall be made to demonstrate operability of the system in accordance with the ATWS Design Basis. 2. <u>At least once during each operating cycle</u> - <ol style="list-style-type: none"> a. Manually initiate one of the two standby liquid control systems and pump demineralized water into the reactor vessel. This test checks explosion of the charge associated with the tested system, proper operation of the valves and pump capacity. Both systems shall be tested and inspected, including each explosion valve in the course of two operating cycles.

3.4/4.4

3.0 LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling systems.

Objective:

To insure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

Specification:

A. ECCS Systems

1. Except as specified in section 3.5.A.3, both Core Spray subsystems and the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System) shall be operable whenever irradiated fuel is in the reactor vessel and the reactor water temperature is greater than 212°F.
2. Except as specified in section 3.5.A.3, the High Pressure Coolant Injection (HPCI) System and the Automatic Depressurization System (ADS) shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel except during reactor vessel hydrostatic or leakage tests.

3.5/4.5

4.0 SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

Applicability:

Applies to the periodic testing of the emergency cooling systems.

Objective:

To verify the operability of the emergency cooling systems.

Specification:

A. ECCS Systems

1. Demonstrate the Core Spray Pumps develop a 2,800 gpm flow rate against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure, when tested in accordance with the Inservice Testing Program.
2. Demonstrate the LPCI Pumps develop a 3,870 gpm flow rate against a system head corresponding to two pumps delivering 7,740 gpm at a reactor pressure of 20 psi greater than containment pressure, when tested in accordance with the Inservice Testing Program.

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Amendment No. 77, 79, 83, 122

3.0 LIMITING CONDITIONS FOR OPERATION

3. One of the following conditions of inoperability may exist for the period specified:
 - a. One Core Spray subsystem may be inoperable for 7 days, or
 - b. One RHR pump may be inoperable for 30 days, or
 - c. One low pressure pump or valve (Core Spray or RHR) may be inoperable with an ADS valve inoperable for 7 days, or
 - d. One of the two LPCI injection paths may be inoperable for 7 days, or
 - e. Two RHR pumps may be inoperable for 7 days, or
 - f. Both of the LPCI injection paths may be inoperable for 72 hours, or
 - g. HPCI may be inoperable for 14 days, provided RCIC is operable, or
 - h. One ADS valve may be inoperable for 14 days, or
 - i. Two or more ADS valves may be inoperable for 12 hours.
4. If the requirements or conditions of 3.5.A.1, 2 or 3 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be placed in a condition in which the affected equipment is not required to be operable within 24 hours.

3.5/4.5

4.0 SURVEILLANCE REQUIREMENTS

3. NOTE: Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.
 - a. Demonstrate, quarterly, with reactor pressure ≤ 1120 psig and ≥ 950 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure, when tested in accordance with the Inservice Testing Program.
 - b. Demonstrate, once per operating cycle, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 2700 gpm against a system head corresponding to reactor pressure.
4. Perform the following tests:

<u>Item</u>	<u>Frequency</u>
ADS Valve Operability	Each Operating Cycle
NOTE: Safety/relief valve operability is verified by cycling the valve and observing a compensating change in turbine bypass or control valve position.	
ADS Inhibit Switch Operability	Each Operating Cycle
Perform a simulated automatic actuation test (including HPCI transfer to the suppression pool and automatic restart on subsequent low reactor water level)	Each Operating Cycle

102

Amendment No. 77, 79, 104, 122

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

B. RHR Intertie Return Line Isolation Valves

- 1. Both RHR Intertie Return Line Isolation Valves shall be operable whenever the mode switch is in RUN.

To be considered operable, each valve must be capable of automatic closure on a LPCI initiation signal or be in the closed position.

Flow shall not be established in the RHR intertie line with the reactor in the Run Mode.

- 2. If one valve is inoperable, either:
 - a. Close the inoperable valve, or
 - b. Close the other Return Line Isolation valve and the RHR Suction Line Isolation valve.
- 3. If the requirements of 3.5.B.1 and 2 cannot be met, the reactor shall be taken out of the RUN mode within 24 hours.

- 5. Perform the following test on the Core Spray Δp Instrumentation:

Check	Once/day
Test	Once/month
Calibrate	Once/3 months

3.5/4.5

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

C. Containment Spray/Cooling System

1. Except as specified in 3.5.C.2 below, both Containment Spray/Cooling Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. A containment/spray cooling subsystem consists of the following equipment powered from one division:
 - 1 RHR Service Water Pump
 - 1 RHR Heat Exchanger
 - 1 RHR Pump
 - Valves and piping necessary for:
 - Torus Cooling
 - Drywell Spray
2. One Containment Spray/Cooling Subsystem may be inoperable for 7 days.
3. If the requirements of 3.5.C.1 or 2 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

* For allowed out of service times for the RHR pumps see Section 3.5.A.

C. Containment Spray/Cooling System

1. Demonstrate the operability of the drywell spray headers and nozzles with an air test during each 10 year period.

3.5/4.5

104
Amendment No. ~~27, 77, 79, 95, 102~~, 122

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

D. RCIC

1. Except as specified in 3.5.D.2 and 3 below, the Reactor Core Isolation Cooling System (RCIC) shall be operable whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 150 psig, except during reactor vessel hydrostatic or leakage tests.
2. RCIC may be inoperable for 14 days, provided HPCI is operable.
3. The controls for the automatic transfer of the pump suction may be inoperable for 30 days, if the pump suction is aligned to the suppression pool.
4. If the requirements or conditions of 3.5.D.1, 2 or 3 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be placed in a condition in which the affected equipment is not required to be operable within 24 hours.

D. RCIC

1. NOTE: Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.
 - a. Demonstrate, quarterly, with reactor pressure ≤ 1120 psig and ≥ 950 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure, when tested in accordance with the Inservice Testing Program.
 - b. Demonstrate, once per operating cycle, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 400 gpm against a system head corresponding to reactor pressure.
2. Perform a simulated automatic actuation test (including transfer to suppression pool and automatic restart on subsequent low reactor water level) each refueling outage.

3.5/4.5

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Amendment No. ~~27, 77, 79~~, 122

Bases 3.5/4.5 (Continued):

The surveillance requirements provide adequate assurance that the LPCI 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 requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which Core Spray system operation or LPCI mode of the RHR system operation maintains core cooling.

The flow tests for the HPCI System are performed at two different pressure ranges such that the system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 990 psig to perform SR 4.5.A.3.a and ≤ 165 psig to perform SR 4.5.A.3.b. Adequate steam flow is represented by total steam flow $\geq 10^6$ lb/hr. Reactor startup, and pressure increase to ≤ 165 psig, is allowed prior to performing the low pressure surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the surveillance test is short. Therefore, pressure may be raised above 150 psig, but ≤ 165 psig to perform this surveillance without entering an LCO for the HPCI System. The reactor pressure is allowed to be increased to normal operating pressure once the low pressure test has been satisfactorily completed since there would be no indication or reason to believe that HPCI is inoperable.

Sufficient time is needed after adequate pressure and flow are achieved to perform these tests. Therefore, SR 4.5.A.3.a and SR 4.5.A.3.b are modified by a note which states that the surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

With the HPCI system inoperable, adequate core cooling is assured by the operability of the redundant and diversified automatic depressurization system and both the Core Spray 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 surveillance requirements provide adequate assurance that the HPCI system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Bases 3.5/4.5 (Continued):

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 150 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 three selected safety-relief valves although the safety analysis only takes credit for two valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

B. RHR Intertie Line

An intertie line is provided to connect the RHR suction line with the two RHR loop return lines. This four-inch line is equipped with three isolation valves. The purpose of this line is to reduce the potential for water hammer in the recirculation and RHR systems. The isolation valves are opened during a cooldown to establish recirculation flow through the RHR suction line and return lines, thereby ensuring a uniform cooldown of this piping. The RHR loop return line isolation valves receive a closure signal on LPCI initiation. In the event of an inoperable return line isolation valve, there is a potential for some of the LPCI flow to be diverted to the broken loop during a loss of coolant accident. Surveillance requirements have been established to periodically cycle the RHR intertie line isolation valves. In the event of an inoperable RHR loop return line isolation valve, either the inoperable valve is closed or the other two isolation valves are closed to prevent diversion of LPCI flow. The RHR intertie line flow is not permitted in the Run Mode to eliminate 1) the need to compensate for the small change in jet pump drive flow or 2) a reduction in core flow during a loss of coolant accident.

C. Containment Spray/Cooling Systems

Two containment spray/cooling subsystems of the RHR system are provided to remove heat energy from the containment and control torus and drywell pressure in the event of a loss of coolant accident. A containment spray/cooling subsystem consists of 2 RHR service water pumps, a RHR heat exchanger, 2 RHR pumps, and valves and piping necessary for Torus Cooling and Drywell Spray. Torus Spray is not considered part of a containment spray/cooling subsystem. Placing a containment spray/cooling subsystem into operation following a loss of coolant accident is a manual operation.

The most degraded condition for long term containment heat removal following the design basis loss of coolant accident results from the loss of one diesel generator. Under these conditions, only one RHR pump and one RHR service water pump in the redundant division can be used for containment spray/cooling. The containment temperature and pressure have been analyzed under these conditions assuming service water and initial suppression pool temperature are both 90°F. Acceptable margins to containment design conditions have been demonstrated. Therefore the containment spray/cooling system is more than ample to provide the required heat removal capability. Refer to USAR Sections 5.2.3.3, 6.2.3.2.3, and 8.4.1.3.

Bases 3.5/4.5 (Continued):

During normal plant operation, the containment spray/cooling system provides cooling of the suppression pool water to maintain temperature within the limits specified in Specification 3.7.A.1.

The surveillance requirements and testing in accordance with the IST Program provide adequate assurance that the containment spray/cooling system will be operable when required.

D. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

The flow tests for the RCIC System are performed at two different pressure ranges such that the system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 4.5.D.1.a and ≤ 165 psig to perform SR 4.5.D.1.b. Adequate steam flow is represented by total steam flow $\geq 10^6$ lb/hr. Reactor startup, and pressure increase to ≤ 165 psig, is allowed prior to performing the low pressure surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the surveillance test is short. Therefore, pressure may be raised above 150 psig, but ≤ 165 psig to perform this surveillance without entering an LCO for the RCIC System. The reactor pressure is allowed to be increased to normal operating pressure once the low pressure test has been satisfactorily completed since there would be no indication or reason to believe that RCIC is inoperable.

Sufficient time is needed after adequate pressure and flow are achieved to perform these tests. Therefore, SR 4.5.D.1.a and SR 4.5.D.1.b are modified by a note which states that the surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Bases 3.5/4.5 (Continued):

E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

F. Recirculation System

The reactor is designed such that thermal hydraulic oscillations are prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal-hydraulic instability, a power-flow exclusion region, to be avoided during normal operation, is calculated using the approved methodology as stated in specification 6.7.A.7. Since the exclusion region may change each fuel cycle the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in this region and to immediately exit upon an entry. Entries into the exclusion region are not part of normal operation. An entry may occur as the result of an abnormal event such as a single recirculation pump trip. In these events, operation in the exclusion region may be needed to prevent equipment damage, but actual time spent inside the exclusion region is minimized. Though operator action can prevent the occurrence and protect the reactor from an instability, the APRM flow biased scram function will suppress oscillations prior to exceeding the fuel safety limit.

Power distribution controls are established to ensure the reactor is operated within the bounds of the stability analysis. With these controls in place, there is confidence that an oscillation will not occur outside of the stability exclusion region. Without these controls, it is theoretically possible to operate the reactor in such a manner as to cause an oscillation outside of the exclusion region. A nominal 5% power-flow buffer region outside of the exclusion region is provided to establish a stability margin to the analytically defined exclusion region. The buffer region may be entered only when the power distribution controls are in place.

Continuous operation with one recirculation loop was analyzed and the adjustments specified in specification 3.5.F.3 were determined by NEDO-24271, June 1980, "Monticello Nuclear Generating Plant Single Loop Operation;" NEDC-30492, April 1984, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant;" and NEDC-32456P, July 1996. Specification 3.6.A.2 governs the restart of the pump in an idle recirculation loop. Adherence to this specification limits the probability of excessive flux transients and/or thermal stresses.

Bases 3.5/4.5 (Continued):

The surveillance requirements provide adequate assurance that the LPCI 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 requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The high pressure coolant injection (HPCI) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCI system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which Core Spray system operation or LPCI mode of the RHR system operation maintains core cooling.

The flow tests for the HPCI System are performed at two different pressure ranges such that the system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ 950 psig to perform SR 4.5.A.3.a and ≤ 165 psig to perform SR 4.5.A.3.b. Adequate steam flow is represented by total steam flow $\geq 10^6$ lb/hr. Reactor startup, and pressure increase to ≤ 165 psig, is allowed prior to performing the low pressure surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the surveillance test is short. Therefore pressure may be raised above 150 psig, but ≤ 165 psig to perform this surveillance without entering an LCO for the HPCI System. The reactor pressure is allowed to be increased to normal operating pressure once the low pressure test has been satisfactorily completed since there would be no indication or reason to believe that HPCI is inoperable.

Sufficient time is needed after adequate pressure and flow are achieved to perform these tests. Therefore, SR 4.5.A.3.a and SR 4.5.A.3.b are modified by a note which states that the surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

With the HPCI system inoperable, adequate core cooling is assured by the operability of the redundant and diversified automatic depressurization system and both the Core Spray 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 surveillance requirements provide adequate assurance that the HPCI system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

3.5/4.5 BASES

3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specification 3.2.H. and 3.5.A, respectively).
2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

3.6/4.6

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1.
 - a. Safety/relief valves shall be tested or replaced each refueling outage in accordance with the Inservice Testing Program. The nominal self-actuation setpoints are specified in Section 2.4.B.
 - b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.
 - c. The integrity of the safety/relief valve bellows shall be continuously monitored.
 - d. The operability of the bellows monitoring system shall be demonstrated each operating cycle.
2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

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Amendment No. 30, 62, 76, 92, 93, 114, 122

3.0 LIMITING CONDITIONS FOR OPERATION

H. Snubbers

1. Except as permitted below, all safety related snubbers shall be operable whenever the supported system is required to be Operable.
2. With one or more snubbers made or found to be inoperable for any reason when Operability is required, within 72 hours:
 - a. Replace or restore the inoperable snubbers to Operable status and perform an engineering evaluation or inspection of the supported components, or
 - b. Determine through engineering evaluation that the as-found condition of the snubber had no adverse effect on the supported components and that they would retain their structural integrity in the event of design basis seismic event, or
 - c. Declare the supported system inoperable and take the action required by the Technical Specifications for inoperability of that system.

3.6/4.6

4.0 SURVEILLANCE REQUIREMENTS

H. Snubbers

The following surveillance requirements apply to all safety related snubbers.

1. Visual inspections:

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible or accessible) may be inspected independently according to the schedule determined by Table 4.6-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.6-1. The initial inspection interval for new types of snubbers shall be established at 18 months +25%.

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Amendment No. ~~9, 39, 45, 82~~, 122

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

3. Functional testing of snubbers shall be conducted at least once per 18 months +25% during cold shutdown. Ten percent of the total number of each brand of snubber shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria in Specification 4.6.H.4 below, an additional ten percent of that brand shall be functionally tested until no more failures are found or all snubbers of that brand have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of the snubbers.

In addition to the regular sample and specified re-samples, snubbers which failed the previous functional test shall be retested during the next test period if they were reinstalled as a safety-related snubber. If a spare snubber has been installed in place of a failed safety related snubber, it shall be tested during the next period.

If any snubber selected for functional testing either fails to lockup or fails to move (i.e. frozen in place) the cause shall be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

3.6/4.6

131
Amendment No. 9-39, 122

Table 4.6.1
SNUBBER VISUAL INSPECTION INTERVAL
Number of Unacceptable Snubbers

Population or Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25

- Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and that decision shall be used as the basis upon which to determine the next inspection interval for that category.
- Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.
- Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.
- Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.
- Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.
- Note 6: All inspection intervals up to and including 48 months may be adjusted a maximum of plus 25%.

Bases 3.6/4.6 (Continued):

To provide assurance of snubber functional reliability, a representative sample of 10% of the installed snubbers will be functionally tested during plant shutdowns at intervals of no more than 18 months +25%. Observed failures of these sample snubbers will require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3.0 LIMITING CONDITIONS FOR OPERATION

reactor core, operations with a potential for reducing the shutdown margin below that specified in specification 3.3.A, and handling of irradiated fuel or the fuel cask in the secondary containment are to be immediately suspended if secondary containment integrity is not maintained.

D. Primary Containment Automatic Isolation Valves

1. During reactor power operating conditions, all Primary Containment automatic isolation valves and all primary system instrument line flow check valves shall be operable except as specified in 3.7.D.2.

3.7/4.7

4.0 SURVEILLANCE REQUIREMENTS

D. Primary Containment Automatic Isolation Valves

1. The primary containment automatic isolation valve surveillance shall be performed as follows:
 - a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.
 - b. At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation.
 - c. All normally open power-operated isolation valves shall be tested in accordance with the Inservice Testing Program. Main Steam isolation valves shall be tested (one at a time) with the reactor power less than 75% of rated.

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Bases 3.14/4.14:

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Learned Task Force Status Report and Short Term Recommendations".

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the site to areas at or beyond the site boundary, conforming to 10 CFR 50, Appendix I.
9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the site to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190; and
11. Limitations on venting and purging of the containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable.

The provisions of Specifications 4.0.B, 4.0.D and 4.0.E are applicable to the Radioactive Effluent Controls Program surveillance frequency.

6.8.E and 6.8.F - RESERVED

G. Inservice Testing Program

This program provides controls for inservice testing of Quality Group A, B, and C pumps and valves which shall be performed in accordance with the requirements of ASME Code Class 1, 2, and 3 pumps and valves, respectively.

1. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

ASME Boiler and Pressure Vessel Code and Applicable Addenda Terminology for Inservice Testing Activities	Required Frequencies for Performing Inservice Testing Activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Biquarterly	At least once per 46 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

2. The provisions of Surveillance Requirement 4.0.B are applicable to the Frequencies for performing inservice testing activities;
3. The provisions of Surveillance Requirement 4.0.D and 4.0.E are applicable to inservice testing activities; and
4. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

6.8.H - RESERVED

I. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Offgas Treatment System, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The quantity of radioactivity after 12 hours holdup contained in each gas storage tank shall be limited to $\leq 22,000$ curies of noble gases (considered as dose equivalent Xe-133). The quantity of liquid radioactive material contained in each outside temporary tank shall be limited to ≤ 10 curies, excluding tritium and dissolved or entrained noble gases.

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the Offgas Treatment System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
3. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of Specifications 4.0.B, 4.0.D and 4.0.E are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

6.8.J - RESERVED

K. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
2. Changes to Bases may be made without prior NRC approval provided the changes do not involve either of the following:
 - a. a change in the TS incorporated in the license; or
 - b. a change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.
4. Proposed changes to the Bases that involve changes as described in a. or b. of Specification 6.8.K.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).