NMC
Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

Operated by Nuclear Management Company LLC

May 2, 2001

10 CFR Part 50
Section 50.90

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request
Relocation of ASME Inservice Testing Requirements to a Licensee Program

Reference 1: NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 1, April 7, 1995.

Reference 2: NUREG-1482, "Guidelines for Inservice Testing Programs at Nuclear Power Plants," April, 1995.

Reference 3: NRC Generic Letter 89-04, Supplement 1: "Guidance on Developing Acceptable Inservice Testing Programs," April 4, 1995.

Reference 4: NRC Letter to Northern States Power Company, "Monticello Nuclear Generating Plant – Technical Specification Interpretation of Surveillance Intervals Required To Be Met For Monticello (TAC NOS. M98821 and MA4277)," January 12, 1999.

Reference 5: Technical Specification Task Force (TSTF)-279, "Remove 'including applicable supports' from Inservice Testing Program." Approved by NRC on July 16, 1998.

Attached is a request for change to the Technical Specifications (TS) of the Operating License for the Monticello Nuclear Generating Plant. This request is submitted pursuant to 10 CFR 50.90.

A047

The proposed amendment would relocate requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Inservice Testing (IST) Program currently contained in Technical Specification 4.15.B to the Technical Specification Administrative Control Section 6.8, Programs and Manuals. The proposed amendment provides conforming changes to several Surveillance Requirements to change the reference from Specification 4.15.B to the Inservice Testing Program. Technical Specifications Surveillance Requirements 4.5.A.3 and 4.5.D.1 are rewritten to be more consistent with the wording in NUREG-1433 (Reference 1). The wording of the Inservice Testing Program in Technical Specification Administrative Control Section 6.8 has been changed to reflect the incorporation of Technical Specification Task Force Initiative (TSTF)-279 (Reference 5), approved by the NRC on July 16, 1998, and to more closely reflect the existing wording in the Monticello Technical Specifications. Additionally, TS Surveillance Requirements for TS 4.6.H.1, 4.6.H.3 and Table 4.6.1 will be revised to change the inspection and functional testing interval extensions reference from +/- 25% to +25% (Reference 4).

Upon Nuclear Regulatory Commission (NRC) approval of the requested changes, the IST requirements will be defined in the Monticello IST program in accordance with 10 CFR 50.55a. The Monticello Operating License includes provisions to meet 10 CFR 50, which includes 10 CFR 50.55a for IST. The proposed license change will make the Monticello Technical Specifications more consistent with NUREG-1433 (Reference 1), NUREG-1482 (Reference 2), and was prepared using the guidance provided in NRC Generic Letter 89-04, Supplement 1 (Reference 3). Additionally, TS regarding surveillance requirements for inspection and functional testing interval extensions will be revised per the guidance in Reference 4.

Exhibit A contains a description of the proposed changes, the reasons for requesting the change, a supporting Safety Evaluation, a Determination of No Significant Hazards, and an Environmental Assessment. Exhibit B contains current Monticello Technical Specification pages marked up to show the proposed changes. Exhibit C contains the revised Monticello Technical Specification pages.

Several changes are on the docket which may administratively effect the changes proposed herein. NMC will submit revised pages, as necessary, to reflect approval of other changes currently pending.

The Monticello Operations Committee has reviewed this application. A copy of this submittal, along with the evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91.

NMC respectfully request a 45-day implementation period for this revision.

If you have any questions regarding this License Amendment Request please contact Doug Neve, Licensing Manager (Interim), at 763-295-1353.

By Byron D. Day

Byron D. Day
Plant Manager
Monticello Nuclear Generating Plant

Subscribed to and sworn before me this 2nd day of May, 2001

Samuel I. Shirey
Notary

SAMUEL I. SHIREY
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

Attachments:	Exhibit A –	Evaluation of Proposed Change to the Monticello Technical Specifications
	Exhibit B -	Current Monticello Technical Specifications Pages Marked up With Proposed Changes
	Exhibit C -	Revised Monticello Technical Specifications Pages

Cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
Minnesota Department of Commerce
J. Silberg, Esq.

Exhibit A

License Amendment Request
<u>Relocation of ASME Inservice Testing Requirements to a Licensee Controlled Program</u>

Evaluation of Proposed Changes to the Monticello Technical Specifications

Pursuant to 10 CFR Part 50, Section 50.90, Nuclear Management Company, LLC, hereby proposes the following changes to Appendix A of Facility Operating License DPR-22, Technical Specifications and associated Bases for the Monticello Nuclear Generating Plant.

Background

The methodology used to create these proposed changes was to remove the existing Inservice Testing Requirements from the Monticello Technical Specifications and to relocate them to a Licensee controlled program with minimal impact to the other existing Technical Specifications.

Details of the Inservice Testing Program (IST) in the Technical Specifications (TS) are proposed to be relocated to a licensee controlled IST program. The IST program is required by 10 CFR 50.55a to be performed in accordance with American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI. Compliance with 10 CFR 50.55a is required by the Monticello Operating License. The Monticello IST program implements the applicable provisions of ASME Code, Section XI. Changes to the plant controlled IST program are controlled by the provisions of 10 CFR 50.59 and 10 CFR 50.55a.

With the current version of the TS, whenever it is determined that a specific Code requirement cannot be satisfied, compliance with the TS requires that the affected component(s) or system(s) be declared inoperable. In NUREG-1482, Section 6, (Reference 1) the Nuclear Regulatory Commission (NRC) recommended that licensees revise their TS to incorporate the revised standard technical specifications for IST programs. Guidance on appropriate changes to Technical Specifications is contained in Reference 1 and Reference 2.

Additionally, TS pages regarding surveillance requirements for inspection and functional testing interval extensions will be revised to change the reference from +/- 25% to +25% (Reference 3).

Proposed Changes and Reasons for Changes

The purpose of this amendment request is to relocate the Inservice Testing requirements contained in Monticello TS 4.15.B to a licensee controlled program. Upon NRC approval of the requested changes, the IST requirements currently defined in TS would be embodied within the Monticello IST program. The Monticello Operating License includes provisions to meet 10 CFR 50, including 10 CFR 50.55a for IST. The proposed license change is consistent with NUREG-1433 (Reference 2), in that Technical Specification Administrative Controls Section 6.8. is being created for the Inservice Testing Program. Surveillance

Requirements 4.5.A.3 and 4.5.D.1 are rewritten for clarification. The reference to "including applicable supports" is being deleted from the description of the "Inservice Testing Program" based on the NRC approval of TSTF-279 (Reference 5). These changes are being proposed pursuant to the guidance provided in NRC Generic Letter 89-04, Supplement 1 (Reference 4). Additionally, TS regarding surveillance requirements for inspection and functional testing interval extensions will be revised to change the reference from +/- 25% to +25% (Reference 3), consistent with Monticello TS surveillance requirement 4.0.B.

The proposed changes to Monticello Technical Specifications, are described below. Specific wording changes are shown in Exhibits B and C, although as previously stated there are docketed changes which may administratively effect the changes proposed herein.

The following changes are proposed:

1. Table of Contents: The Table of Contents is revised to reflect the deletion of Inservice Testing requirements from TS 3.15 and 4.15.

   Justification: The Table of Contents are revised to reflect the requested relocation of the ASME Code, Section XI, Inservice Testing requirements to the plant controlled program.

2. Specification 4.4.A.1, 4.5.A.1 and 4.5.A.2: Delete references of Specification 4.15.B and replace with a reference to the Inservice Testing Program.

   Justification: These changes are to provide consistency with the revised Technical Specifications.

3. Specification 4.5.A.3 and 4.5.D.1: Rewrite TS Surveillance Requirements for HPCI and RCIC for clarification and to more consistently reflect the wording in NUREG-1433.

   Justification: The rewording of the Surveillance Requirements now make them more consistent with the wording in NUREG-1433. The frequencies are specified as quarterly and once per operating cycle, consistent with other similar Monticello TS. The addition of the "NOTE" from NUREG-1433 allows sufficient time to ensure adequate pressure and flow are achieved before performing these tests, and the test pressures are consistent with current surveillance requirements. Reactor startup, and pressure increase to less than or equal 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.

4. Specification 4.5.A.4, 4.5.B, 4.5.C.2 and 4.5.D.2: Delete Valve Operability Surveillance Requirements.

   Justification: Valve operability testing will be relocated to the licensee controlled IST program and the procedures implementing the IST program. Any changes to these requirements will be controlled by the provisions of 10 CFR 50.59 and 10 CFR 50.55a. These changes are consistent with NUREG-1433.

5. Specification 4.5.C.1: Delete surveillance requirement for RHR Service Water pumps.

   Justification: RHR Service Water pump testing requirements will be relocated to the licensee controlled IST program and the procedures implementing the IST program. This surveillance requirement can be relocated to the IST program because RHR Service Water is not an Emergency Core Cooling System (ECCS) directly required to meet the criteria of 10 CFR Part 50, Appendix K. This change is consistent with NUREG-1433.

6. Specification 4.6.E.1.a and 4.7.D.1.c: Delete reference of Specification 4.15.B and replace with a reference to the Inservice Testing Program.

   Justification: These changes are to provide consistency with the revised TS.

7. Specification 4.6.H.1, 4.6.H.3 and Table 4.6.1: Revise TS surveillance requirements for inspection and functional testing interval extensions of Snubbers from +/- 25% to +25%.

   Justification: Letter from NRC to Northern States Power Company (Reference 3).

8. Specification 3.15 and 4.15, Inservice Testing: Delete Title, Specification, Applicability, Objective, and Specification.

   Justification: The requirements are to be removed from the TS and relocated to a licensee controlled IST program. Additionally, consistent with NUREG-1433, "Standard Technical Specifications, General Electric Plant, BWR/4," Revision 1, an Administrative Controls Technical Specification Section 6.8, has been created for the Inservice Testing Program.

9. TS 6.8.G: Added a new requirement to the Administrative Control Section of TS defining the requirements for implementing the Inservice Testing Program for Pumps and Valves. It also updates the wording of the Inservice Testing Program to delete the reference to "including applicable supports." Additionally, the Table which correlates frequencies has been omitted to more closely match the existing Monticello TS.

   Justification: This requires implementation of a pump and valve Inservice Testing Program in compliance with 10 CFR 50.55a and clarifies the compliance requirements of the applicable sections of the ASME Boiler and Pressure Vessel Code per the guidance in NUREG-1482, with the wording from NUREG-1433, Revision 1. The wording "including applicable supports" has been deleted from the IST Program per the NRC approval of TSTF-279 (Reference 5). The Table in NUREG-1433 that correlates frequencies is not included in the proposed IST Program, because the Monticello TS frequencies more closely correspond to the wording in the ASME, Boiler and Pressure Vessel Code.

Monticello Surveillance Requirements 4.0.B, 4.0.D and 4.0.E are referenced since they are the equivalent of NUREG-1433, Surveillance Requirements 3.0.2 and 3.0.3.

The proposed changes identified above involve changes to the Monticello Technical Specifications. Except as noted, these changes are consistent with guidance provided by the NRC in NUREG-1482, "Guidance for Inservice Testing At Nuclear Power Plants," NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4" and NRC Generic Letter 89-04, Supplement 1: "Guidance on Developing Acceptable Inservice Testing Programs." These changes do not affect plant design, method of operation, or the scope or intent of the pump and valve inservice testing program. Similar changes have been implemented for plants converting to the improved Standard Technical Specifications (STS). Additionally, TS surveillance requirements for inspection and functional testing interval extensions will be revised to change the reference from +/- 25% to +25% (Reference 3).

Appropriate changes to the Bases are also included.

Safety Evaluation

The changes proposed above relocate TS requirements for the Inservice Testing Program to a licensee controlled program. Also, consistent with NUREG-1433, this change creates Administrative Controls Section 6.8.G, "Inservice Testing Program," in the Monticello Technical Specifications. Per the guidance provided in NUREG-1482 RHR Service Water pump testing requirements are being relocated to the IST program because RHR Service Water is not an Emergency Core Cooling System (ECCS) directly required to meet the criteria of 10 CFR Part 50, Appendix K. Additionally, these proposed changes also revise surveillance interval extensions for TS surveillance requirements 4.6.H.1, 4.6.H.3 and Table 4.6.1 from +/-25% to +25% (Reference 3).

As previously stated, the relocated requirements are duplicated in 10 CFR 50.55a. Therefore, it is not necessary to retain the provisions in the TS. No reduction in any previous commitments to 10 CFR 50.55a or the ASME Code is proposed as a result of the relocation. Changes to the plant controlled IST program will be controlled by the provisions of 10 CFR 50.59 and 10 CFR 50.55a. Adopting the recommendations of NUREG-1482, NUREG-1433, and TSTF-279, will allow Monticello to fully comply with the prescribed requirements of 10 CFR 50.55a and the plant Technical Specifications without placing impractical administrative requirements on the NRC or the plant staff. Also, revising surveillance interval extensions for TS 4.6.H.1, 4.6.H.3 and Table 4.6.1 from +/-25% to +25%, will correct the Monticello Technical Specifications as committed to by Monticello plant staff in a conference call with the NRC, and reaffirmed in an NRC letter from E. G. Adensam to R. O. Anderson, dated January 12,1999 (Reference 3).

Exhibit A

Determination of No Significant Hazards Consideration

Changes are proposed to the Monticello Technical Specifications (TS) for the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI required Inservice Testing (IST). The proposed license change is submitted using the guidance provided in NUREG-1482 (Reference 1) and is consistent with NUREG-1433 (Reference 2), in that Technical Specification Administrative Controls Section 6.8. is being created for IST. Additionally, surveillance interval extensions will be revised from the previous +/-25% to +25%, per the guidance from Reference 3. The proposed changes have been evaluated to determine whether they constitute a no significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92.

This analysis is provided below:

1. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The requested changes are administrative in nature in that they relocate IST requirements from the Monticello TS to a licensee controlled IST program, rewrite TS Surveillance Requirements 4.5.A.3 and 4.5.D.1 for clarification using the wording from NUREG-1433 and revise TS surveillance requirements for inspection and functional testing interval extensions. The requested changes will not revise previous commitments to 10 CFR 50.55a of ASME Code, Section XI, IST requirements.

The proposed changes do not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Since the accident analyses remain bounding, their radiological consequences are not adversely affected.

Therefore, the probability or consequences of an accident previously evaluated are not affected.

2. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The requested changes are administrative in nature in that they relocate IST requirements from the Monticello TS to the licensee controlled IST program, rewrite TS Surveillance Requirements 4.5.A.3 and 4.5.D.1 for clarification using the wording from NUREG-1433 and revise TS surveillance requirements for inspection and functional testing interval extensions. The requested changes will not revise previous commitments to 10 CFR 50.55a or ASME Code, Section XI, IST requirements.

The proposed changes do not involve changes to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes.

Therefore, the possibility of a new or different kind of accident from any accident previously analyzed is not created.

3. The proposed amendment will not involve a significant reduction in the margin of safety.

The requested changes are administrative in nature in that they relocate IST requirements from the Monticello TS to the licensee controlled IST program, rewrite TS Surveillance Requirements 4.5.A.3 and 4.5.D.1 for clarification using the wording from NUREG-1433 and revise TS surveillance requirements for inspection and functional testing interval extensions. The requested changes will not revise previous commitments to 10 CFR 50.55a or ASME Code, Section XI, IST requirements. Program requirements will remain to ensure that Code requirements are met.

Therefore, a significant reduction in the margin of safety is not involved.

Based on the above evaluation, and pursuant to 10 CFR 50.91, NMC has determined that the operation of the Monticello Nuclear Generating Plant in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR 50.92.

Environmental Assessment

Nuclear Management Company has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.

Exhibit A

References

1. NUREG-1482, " Guidelines for Inservice Testing Programs at Nuclear Power Plants," April, 1995

2. NUREG-1433, "Standard Technical Specification, General Electric Plants, BWR/4," Revision 1, April 7, 1995

3. NRC Letter to Northern States Power Company, "Monticello Nuclear Generating Plant – Technical Specification Interpretation of Surveillance Intervals Required To Be Met For Monticello," January 12, 1999.

4. NRC Generic Letter 89-04, supplement 1: "Guidance on Developing Acceptable Inservice Testing Programs," April 4, 1995

5. Technical Specification Task Force (TSTF)-279, "Remove 'including applicable supports' from Inservice Testing Program." Approved by NRC on July 16, 1998.

Exhibit B

License Amendment Request

Relocation of ASME Inservice Testing Requirements to a Licensee Controlled Program

Current Monticello Operating License and Monticello Technical Specification Pages Marked Up With Proposed Changes

This Exhibit consist of current Monticello Operating License and Technical Specification pages marked up with the proposed changes. The pages included in the exhibit are listed below:

Pages

Monticello Technical Specifications pages

iv
25b
93
101
102
103
104
105
111
113
127
129
131
132a
155
170
229e
229f
229ff
229g

TABLE OF CONTENTS (Cont'd)

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. ~~The plant uses a fixed surveillance program that prevents repetitive addition of the allowable 25% extension.~~ Each surveillance test is completed within plus ~~or minus~~ 25% of each scheduled ~~fixed~~ 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).

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. System Operation

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.

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

Specification:

A. The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once per quarter -

   Pump minimum flow rate of 24 gpm shall be verified against a system head of 1275 psig when tested ~~pursuant to Specification 4.15.B.~~ 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. At least once during each operating cycle -

   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.

3.0 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.

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 ~~pursuant to Specification 4.15.B.~~ 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 ~~pursuant to Specification 4.15.B.~~ in accordance with the Inservice Testing Program.

3. ~~Demonstrate the HPCI Pump develops a 2700 gpm flow rate against a reactor pressure range of 1120 psig to 150 psig, when tested pursuant to Specification 4.15.B.~~ Insert Attached

INSERT

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 $\leq$ 1120 psig and $\geq$ 950 psig, the HPCI pump can develop a flow rate $\geq$ 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 $\leq$ 165 psig, the HPCI pump can develop a flow rate $\geq$ 2700 gpm against a system head corresponding to reactor pressure.

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.

4.0 SURVEILLANCE REQUIREMENTS

4. Perform the following tests:

Item	Frequency	
~~Motor Operated Valve Operability~~	~~Pursuant to Specification 4.15.B~~	DELETE
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	

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

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

3.0 LIMITING CONDITIONS FOR OPERATION

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.

4.0 SURVEILLANCE REQUIREMENTS

~~B. RHR Intertie Return Line Isolation Valves~~

~~Test the RHR Intertie Line Isolation valves in accordance with Specification 4.15.B.~~

DELETE

3.0 LIMITING CONDITIONS FOR OPERATION

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.

4.0 SURVEILLANCE REQUIREMENTS

C. Containment Spray/Cooling System

DELETE

1. ~~Demonstrate the RHR Service Water pumps develop 3,500 gpm flow rate against a 500 ft head when tested pursuant to Specification 4.15.B.~~

2. ~~Test the valves in accordance with Specification 4.15.B.~~

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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
D. RCIC	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.	Insert Attached 1. ~~Demonstrate the RCIC Pump develops a 400 gpm flow rate against a reactor pressure range of 1120 to 150 psig, when tested pursuant to Specification 4.15.B.~~ ~~in accordance with the Inservice Testing Program.~~
2. RCIC may be inoperable for 14 days, provided HPCI is operable.	2. ~~Test the motor operated valves pursuant to Specification 4.15.B.~~ DELETE
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.	3. Perform a simulated automatic actuation test (including transfer to suppression pool and automatic restart on subsequent low reactor water level) each refueling outage.
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.	

INSERT

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 $\leq$ 1120 psig and $\geq$ 950 psig, the RCIC pump can develop a flow rate $\geq$ 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 $\leq$ 165 psig, the RCIC pump can develop a flow rate $\geq$ 400 gpm against a system head corresponding to reactor pressure.

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.

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~~The capacity of the system is selected to provide the required core cooling. The HPCI pump is designed to deliver greater than or equal to 3000 gpm (safety analyses assumed 2700 gpm) at reactor pressures between 1120 and 150 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.~~

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.

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.

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The flow tests for the HPCI System are performed at two different pressure ranges such that 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 $\geq$ 950 psig to perform SR 4.5.A.3.a and $\leq$ 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 $\leq$ 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 $\leq$ 165 psig to perform this surveillance without entering a 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.

and testing in accordance with the IST program

Bases 3.5/4.5 (Continued):

The surveillance requirements provide adequate assurance that the containment spray/cooling system will be operable when required. ~~The head and flow requirements specified for the RHR service water pumps provide assurance that the minimum required service water flow can be supplied to the RHR heat exchangers for the most degraded condition for long-term containment heat removal following the design basis loss of coolant accident.~~

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.

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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.

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.

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The flow tests for the RCIC System are performed at two different pressure ranges such that 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 $\geq$ 950 psig to perform SR 4.5.D.1.a and $\leq$ 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 $\leq$ 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 $\leq$ 165 psig to perform this surveillance without entering a 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.

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.

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage ~~pursuant to Specification 4.15.B.~~ 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.

in accordance with the Inservice Testing Program.

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.

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%.
   +

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.

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 ~~or minus~~ 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.

4.0 SURVEILLANCE REQUIREMENTS

D. Primary Containment Automatic Isolation Valves

1. The primary containment automatic isolation valve surveillance shall be performed as follows: g

   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 ~~pursuant to Specification 4.15.B.~~ 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. g

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".

(next page is 229h)

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.15 INSERVICE TESTING	**4.15 INSERVICE TESTING**
Applicability:	Applicability:
Applies to safety-related pumps and valves.	Applies to the periodic testing of safety-related pumps and valves.
Objective:	Objective:
To assure the integrity and operational readiness of safety-related pumps and valves.	To verify the integrity and operational readiness of safety-related pumps and valves.
Specification:	Specification:
A. (Deleted)	A. (Deleted)

DELETE

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	B. <u>Inservice Testing</u> 1. Inservice Testing of Quality Group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2 and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(a)(g)(6)(i), or where alternate testing is justified in accordance with Generic Letter 89-04. 2. Nothing in the ASME Boiler and Pressure Vessel code shall be construed to supersede the requirements of any Technical Specification.

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Bases 3.15/4.15:

A program of inservice testing of Quality Group A, B, and C pumps and valves is in effect at the Monticello plant that conforms to the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code or where alternate testing is justified in accordance with Generic Letter 89-04. If a Code required inspection is impractical for the Monticello facility, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

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Exhibit C

License Amendment Request

Relocation of ASME Inservice Testing Requirements to a Licensee Controlled Program

Revised Monticello Operating License and Technical Specification Pages

This Exhibit consist of revised Monticello Operating License and Technical Specification pages that incorporate the proposed changes. The pages included in the exhibit are listed below:

Pages

Monticello Technical Specifications pages

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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

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the standby liquid control system.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. System Operation

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.

4.0 SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirements for the standby liquid control system.

Objective:

To verify the operability of the standby liquid control system.

Specification:

A. The operability of the standby liquid control system shall be verified by performance of the following tests:

1. At least once per quarter -

   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. At least once during each operating cycle -

   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.

3.0 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.

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.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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.

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:

Item	Frequency
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.

Item	Frequency
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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE REQUIREMENTS

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

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

3.0 LIMITING CONDITIONS FOR OPERATION

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.

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.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

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.

4.0 SURVEILLANCE REQUIREMENTS

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.

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.

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.

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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.

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 at least once every three months.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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.

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%.

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.

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.

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.

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.

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".

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6.8.B through 6.8.H - 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. The provisions of Surveillance Requirement 4.0.B are applicable to the Frequencies for performing inservice testing activities;

2. The provisions of Surveillance Requirement 4.0.D and 4.0.E are applicable to inservice testing activities; and

3. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.