Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

FEB - 5 2002

Docket No. 50-336
B18462

RE: 10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 2
License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and
Service Water Surveillance Requirements

Introduction

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2. DNC is proposing to change Technical Specifications 3.6.3.1, "Containment Systems - Containment Isolation Valves;" 3.7.3.1, "Plant Systems - Reactor Building Closed Cooling Water System;" and 3.7.4.1, "Plant Systems - Service Water System." The Bases for these Technical Specifications will be modified to address the proposed changes.

The majority of the proposed changes will revise the surveillance requirements associated with the Containment Isolation Valves (CIVs), Reactor Building Closed Cooling Water (RBCCW) System, and Service Water (SW) System. The proposed changes will remove redundant testing requirements that are already addressed by the Inservice Testing (IST) Program, which is required pursuant to Technical Specification 4.0.5, and will use Technical Specification 4.0.5 to control the specific acceptance criteria and frequency of test performance. Additional proposed changes will remove the post maintenance testing requirements associated with the CIVs, revise the wording of the RBCCW and SW Systems Limiting Conditions for Operation (LCOs), and increase the allowed outage times for the RBCCW and SW Systems.

Attachment 1 provides a discussion of the proposed changes and the Safety Summary. Attachment 2 provides the Significant Hazards Consideration. Attachment 3 provides the marked-up version of the appropriate pages of the current Technical Specifications. Attachment 4 provides the retyped pages of the Technical Specifications.

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Environmental Considerations

DNC has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.22. DNC has determined that the proposed changes meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that the changes are being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to use of a facility component located within the restricted area, as defined by 10 CFR 20, or that changes an inspection or a surveillance requirement, and that the amendment request meets the following specific criteria.

(i) The proposed changes involve no Significant Hazards Consideration.

As demonstrated in Attachment 2, the proposed changes do not involve a Significant Hazards Consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released off site.

The proposed changes will revise the Technical Specification LCO, action, and surveillance requirements associated with the CIVs and the RBCCW and SW Systems. However, the operability requirements for these components and systems will remain the same. The proposed changes are consistent with the design basis of the plant. The proposed changes will not result in an increase in power level, will not increase the production of radioactive waste and byproducts, and will not alter the flowpath or method of disposal of radioactive waste or byproducts. Therefore, the proposed changes will not increase the type and amounts of effluents that may be released off site.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed changes will revise the Technical Specification LCO, action, and surveillance requirements associated with the CIVs and the RBCCW and SW Systems. However, the operability requirements for these components and systems will remain the same. The proposed changes will not result in changes in the configuration of the facility. There will be no change in the level of controls or methodology used for processing radioactive effluents or the handling of solid radioactive waste. There will be no change to the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.

<u>Conclusions</u>

The proposed changes have been evaluated and we have concluded the proposed changes are safe. The proposed changes do not involve an adverse impact on public health and safety (see the Safety Summary provided in Attachment 1) and do not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2).

<u>Site Operations Review Committee and Nuclear Safety Assessment Board</u>

The Site Operations Review Committee and Nuclear Safety Assessment Board have reviewed and concurred with the determinations.

<u>Schedule</u>

We request issuance of this amendment for Millstone Unit No. 2 by January 31, 2003, with the amendment to be implemented within 90 days of issuance.

<u>Additional Conditions</u>

We request the following additional condition apply to the proposed License Amendment.

> For surveillance requirements that are new in this amendment, the first performance is due at the end of the first surveillance interval that begins on the date of implementation of this amendment.

<u>State Notification</u>

In accordance with 10 CFR 50.91(b), a copy of this License Amendment Request is being provided to the State of Connecticut.

There are no regulatory commitments contained within this letter.

If you should have any questions on the above, please contact Mr. Ravi Joshi at (860) 440-2080.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

Sworn to and subscribed before me

this 5th day of February, 2002

Notary Public

My Commission expires SANDRA J. ANTON
NOTARY PUBLIC
COMMISSION EXPIRES
MAY 31, 2005

Attachments (4)

cc: H. J. Miller, Region I Administrator
J. Harrison, NRC Project Manager, Millstone Unit No. 2
NRC Senior Resident Inspector, Millstone Unit No. 2

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

Attachment 1

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and Service Water Surveillance Requirements
Discussion of Proposed Changes and Safety Summary

License Basis Document Change Request 2-14-01 Containment Isolation, Reactor Building Closed Cooling Water, and Service Water Surveillance Requirements

Discussion of Proposed Changes and Safety Summary

Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License DPR-65 by incorporating the attached proposed changes into the Technical Specifications of Millstone Unit No. 2. DNC is proposing to change Technical Specifications 3.6.3.1, "Containment Systems - Containment Isolation Valves;" 3.7.3.1, "Plant Systems - Reactor Building Closed Cooling Water System;" and 3.7.4.1, "Plant Systems - Service Water System." The Bases for these Technical Specifications will be modified to address the proposed changes.

The majority of the proposed changes will revise the surveillance requirements associated with the Containment Isolation Valves (CIVs), Reactor Building Closed Cooling Water (RBCCW) System, and Service Water (SW) System. The proposed changes will remove redundant testing requirements that are already addressed by the Inservice Testing (IST) Program, which is required pursuant to Technical Specification 4.0.5, and use Technical Specification 4.0.5 to control the specific acceptance criteria and frequency of test performance. Additional proposed changes will remove the post maintenance testing requirements associated with the CIVs, revise the wording of the RBCCW and SW Systems Limiting Conditions for Operation (LCOs), and increase the allowed outage time (AOT) for the RBCCW and SW Systems.

Millstone Unit No. 2 Inservice Testing Program

The Millstone Unit No. 2 IST Program covers ASME Code Class 1, 2, and 3 pumps and valves. This program contains the test requirements for each component, approved alternatives to the test requirements where implemented, and special comments or conditions associated with each component. Some safety significant non-ASME Class 1, 2 or 3 components have been included in the IST Program as augmented testing.

The Third Ten-Year IST Interval, which began on April 1, 1999, was developed in accordance with the requirements of ASME/ANSI OM-1987[1] and Addenda OMa-1988, which is referenced from 10 CFR 50.55a and ASME Section XI, 1989 edition. The guidelines of NUREG-1482,[2] that provide acceptable alternative methods of inservice testing have been adopted, where noted.

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(1) ASME/ANSI OM-1987, "Operation and Maintenance of Nuclear Power Plants, Inservice Testing of Valves in Light Water Reactor Power Plants," dated 1987.

(2) NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," April 1995.

Components that provide a specific function in shutting down the reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an analyzed accident are called "safety-related components" in this document. Millstone Unit No. 2 was licensed for hot shutdown. However, in recognition of the concern for the ability to provide long term cooling post-accident, components required to bring the reactor to cold shutdown, and maintain the reactor at cold shutdown have been included in the program as augmented tests.

The Millstone Unit No. 2 IST Program covers the following safety related systems.

- Auxiliary Feedwater System
- Chemical and Volume Control System (Charging, Boric Acid)
- Chilled Water System (Vital)
- Containment Spray System
- Containment Ventilation System (CIVs Only)
- Diesel Generator (Non ASME Code Augmented Testing)
- Enclosure Building Filtration System (CIVs Only)
- Fire Protection System (CIVs Only)
- Gaseous Radwaste System (CIVs Only)
- Instrument Air System (CIVs Only)
- Liquid Radwaste System (CIVs Only)
- Main Steam System
- Primary Makeup Water System (CIVs Only)
- Reactor Building Closed Cooling Water System
- Reactor Coolant System
- High Pressure Safety Injection System
- Low Pressure Safety Injection System
- Service Water System
- Spent Fuel Pool Cooling System
- Station Air (CIVs Only)

The criteria for valves included in Millstone Unit No. 2 IST Program are:

- Active and passive valves which are required to perform a specific function in shutting down the reactor to a safe shutdown condition.
- Active and passive valves which are required to perform a specific function in maintaining the safe shutdown condition.
- Active and passive valves which are required to perform a specific function in mitigating the consequences of an accident.
- Pressure relief devices that protect systems or portions of systems which perform a required function in shutting down the reactor to a safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident.

- If repositioning of a manual valve is credited in the safety analysis, the valve is included in the IST Program and tested in accordance with ASME/ANSI OM-1987. Passive manual valves are not included in the IST Program testing unless they have remote position indication or require leak rate testing.

The criteria for pumps included in Millstone Unit No. 2 IST Program are:

- All ASME Code Class 1, 2, or 3 pumps provided with an emergency power source which are required to perform a specific function in shutting down the reactor to a safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident.
- Pumps which are provided with an emergency power source solely for operating convenience are excluded.

The IST Program (Technical Specification 4.0.5) will control the component acceptance criteria and determine the frequency of test performance. The acceptance criteria (e.g., valve stroke time, pump differential pressure, pump flowrate) is based on the assumed component operation. Performance of the required testing will verify proper component operation, and should be able to detect component degradation. The frequency of test performance may change based on equipment performance.

The use of the IST Program to control pump and valve testing is consistent with current industry practices and published guidelines. Many of the surveillance requirements contained in NUREG-1432[3] illustrate the use of the IST Program to control the acceptance criteria and frequency of test performance. The surveillance requirements contained in NUREG-1432 refer to the IST Program instead of Specification 4.0.5. NUREG-1432 has replaced Specification 4.0.5 with a program contained in Section 5 (Technical Specification 5.5.8, "Inservice Testing Program"). However, since the Millstone Unit No. 2 Technical Specifications still contains Technical Specification 4.0.5, the proposed CIV surveillance requirement will refer to "Specification 4.0.5."

<u>Technical Specification Changes</u>

Each proposed Technical Specification change will be discussed. Table 1, located at the end of this attachment, summarizes the proposed changes to the surveillance requirements.

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[3] NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," Revision 2, April 2001.

Technical Specification 3.6.3.1

1. The following changes will be made to SR 4.6.3.1.1.

    a. The number of this SR will be changed from SR 4.6.3.1.1 to SR 4.6.3.1 as a result of the changes to be discussed. This will not result in any technical change to the current requirements.

    b. The word "containment" will be added to the proposed SR 4.6.3.1 to specify the type of isolation valves this SR addresses. This will clarify the intent of the SR. It will not result in any technical change to the current requirements.

    c. The phrase "testable during power operation" will not be retained in the proposed SR 4.6.3.1. The proposed testing will be done in accordance with the IST Program (SR 4.6.3.1.a). This program provides guidance to determine valve testing frequency based on the ability to test valves during power operation. This will not adversely impact test performance as discussed below.

2. The requirements of SR 4.6.3.1.1.a.1 will be relocated to the proposed SR 4.6.3.1.a. This will result in the following changes to the current requirements.

    a. The proposed frequency of "when tested pursuant to Specification 4.0.5" will not result in a change in test frequency from 92 days. The IST Program, which covers safety related valves, determines the frequency of valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of valves, is consistent with standard industry practices and guidelines.

    b. SR 4.6.3.1.a will not specify that the valves are stroked through one complete cycle. However, the proposed SR will still require the valve isolation times to be checked. Since this will require the valves to be stroked, there will be no change in test performance.

    c. SR 4.6.3.1.a will specify that the stroke time of "automatic" power operated CIVs is to be verified. This will reduce the number of CIVs tested by this requirement since not all power operated valves are automatically operated. However, the IST Program (Technical Specification 4.0.5) will require the non-automatic power operated CIVs to be tested since they are classified as safety related valves. As a result, the number of power operated CIVs tested is not expected to change.

3. SR 4.6.3.1.1.a.2 will be deleted. The requirement to cycle manual valves will be addressed by the IST Program, which covers safety related valves. The IST Program will determine which safety related valves need to be cycled, and at what frequency. The number of valves tested is expected to decrease as a result of this change because not all manually operated valves are required to change position to mitigate design basis events or support safe shutdown conditions. Manual valves that are not required to change position are classified as passive valves by the IST Program and are not required to be cycled. The IST Program determines the frequency of safety related valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of testable valves, is consistent with standard industry practices and guidelines. The expected reduction in the number of valves tested is a less restrictive change.

4. SR 4.6.3.1.1.b will be deleted. This requirement, which requires valve operability to be verified prior to returning the valve to service following maintenance, repair, or replacement is not necessary. Post maintenance testing after completion of valve work, which is controlled by plant procedures, will specify this verification if the associated work could adversely affect valve operation (e.g., affect valve stroke time). This verification is necessary prior to considering the valve operable after completion of maintenance activities that could affect valve operation. This requirement can be adequately controlled in plant procedures and the associated procedure change process. This approach is consistent with standard industry practices and guidelines. This is a less restrictive change.

5. The requirements of SRs 4.6.3.1.2.a and 4.6.3.1.2.b will be combined into the proposed SR 4.6.3.1.b. This surveillance requirement will require verification that all automatic containment isolation valves actuate to the isolation position following an actual or simulated actuation signal. This will encompass containment isolation valves that close on a containment isolation actuation signal, and containment purge valves (containment isolation valves) that close on a containment high radiation actuation signal. In addition, verification that the containment purge valves automatically close on high containment radiation is also required by SR 4.9.4.2. This will not change test performance.

6. The requirements of SR 4.6.3.1.2.c will be relocated to the proposed SR 4.6.3.1.a. This will result in the following changes to the current requirements.

    a. The proposed frequency of "when tested pursuant to Specification 4.0.5" will not result in a change in test frequency from a cold shutdown or refueling mode interval. The IST Program, which covers safety related valves, determines the frequency of valve testing based on the ability to test valves during plant operation. The valves covered by the current SR 4.6.3.1.2.c are not capable of being tested during plant operation. As a result, the IST Program will specify testing these valves at a cold shutdown or refueling interval frequency which is equivalent to the current requirement. This approach, to use the IST Program to control the cycling of valves, is consistent with standard industry practices and guidelines.

b. SR 4.6.3.1.a will not specify that the valves are stroked through one complete cycle. However, the proposed SR will still require the valve isolation times to be checked. Since this will require the valves to be stroked, there will be no change in test performance.

c. SR 4.6.3.1.a will specify that the stroke time of "automatic" power operated CIVs is to be verified. This will reduce the number of CIVs tested by this requirement since not all power operated valves are automatically operated. However, the IST Program (Technical Specification 4.0.5) will require the non-automatic power operated CIVs to be tested since they are classified as safety related valves. As a result, the number of power operated CIVs tested is not expected to change.

7. SR 4.6.3.1.2.d will be deleted. The requirement to cycle each manual valve not locked, sealed, or otherwise secured in position, will be addressed by the IST Program, which covers safety related valves. The IST Program will determine which safety related valves need to be cycled, and at what frequency. The number of valves tested is expected to decrease as a result of this change because not all manually operated valves are required to change position to mitigate design basis events or support safe shutdown conditions. Manual valves that are not required to change position are classified as passive valves by the IST Program and are not required to be cycled. The IST Program determines the frequency of safety related valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of testable valves, is consistent with standard industry practices and guidelines. The expected reduction in the number of valves tested is a less restrictive change.

Technical Specification 3.7.3.1

1. The word "independent" will be removed from the LCO. The level of independence between the RBCCW System loops (trains) is a design feature of the RBCCW System. The Millstone Unit No. 2 Final Safety Analysis Report (FSAR) describes the approved degree of separation and level of independence between RBCCW loops. If the approved degree of separation and level of independence are not maintained, the RBCCW loops may not be operable as discussed in the Bases for this specification. Therefore, it is not necessary to include a requirement for the RBCCW loops to be independent in Technical Specifications. This will not change the requirement for two RBCCW loops to be operable.

2. The action requirement AOT to restore an inoperable RBCCW loop will be increased from 48 hours to 72 hours. A 72 hour AOT is a standard time to restore safety significant equipment (e.g., one inoperable EDG, Technical Specification 3.8.1.1). It is consistent with standard industry guidelines contained in NUREG-1432 (Technical Specification 3.7.7, Component Cooling Water System). This is a less restrictive change.

3. The requirements of SRs 4.7.3.1.a.1, 4.7.3.1.a.2, and 4.7.3.1.a.3 will be deleted. These three monthly surveillance requirements check the RBCCW pumps for proper operation and for potential degradation. However, since both loops and the associated pumps are normally operating, any potential malfunction or pump degradation should be readily apparent. Therefore, it is not necessary to include a monthly or quarterly surveillance requirement in Technical Specifications to check for proper RBCCW pump operation or potential degradation. The IST Program will still monitor the RBCCW pumps for proper operation on a regular interval. The IST Program specifies a minimum test performance interval of 92 days, which may become more frequent based on equipment performance. Automatic operation of the RBCCW pumps will be verified by proposed SR 4.7.3.1.c. This approach, to not include a surveillance requirement in Technical Specifications that requires monthly or quarterly testing of the RBCCW pumps, is consistent with standard industry practices and guidelines, including NUREG-0212[(4)] and NUREG-1432 (Technical Specification 3.7.7). This is a less restrictive change.

4. SR 4.7.3.1.a.4, which verifies each RBCCW loop is aligned to receive power from a separate operable emergency bus, will be deleted. This is redundant to the definition of operable (Definition 1.6) and can be removed without affecting any operability requirements. For a component to be operable it must have its normal and emergency power supply, except as provided by Technical Specification 3.0.5. In addition, it is not necessary to specify separate emergency busses. The degree of separation and the level of independence between the RBCCW loops and the associated emergency busses is a design feature of the RBCCW System and the emergency power distribution system. The Millstone Unit No. 2 FSAR describes the approved degree of separation and level of independence between RBCCW loops and emergency busses. If the approved degree of separation and level of independence are not maintained, the RBCCW loops may not be operable. Therefore, it is not necessary to include a check that each RBCCW loop is aligned to separate emergency busses. Since this surveillance requirement is redundant to the current definition of operable, its deletion will not result in a technical change.

[(4)] NUREG-0212, "Standard Technical Specifications Combustion Engineering PWRs," Revision 2, Fall 1980.

5. The requirements of SR 4.7.3.1.a.5 will be relocated to the proposed SR 4.7.3.1.a. This surveillance requirement will require verification that all RBCCW valves in the flow path servicing safety related equipment that are not locked, sealed, or otherwise secured in position are in the correct position. This will encompass manual, remote, and automatically operated RBCCW valves. Relocation of this requirement will not change test performance.

6. SR 4.7.3.1.a.6 will be deleted. The requirement to cycle all automatically operated valves will be addressed by the IST Program, which covers safety related valves. The IST Program will determine what safety related valves need to be cycled, and at what frequency. The number of valves tested is expected to decrease as a result of this change because not all automatically operated valves are required to change position to mitigate design basis events or support safe shutdown conditions. Automatic valves that are not required to change position are classified as passive valves by the IST Program and are not required to be cycled. The IST Program determines the frequency of safety related valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of testable valves, is consistent with standard industry practices and guidelines. The expected reduction in the number of valves tested is a less restrictive change.

7. SR 4.7.3.1.b will be deleted. The requirement to cycle all power operated valves will be addressed by the IST Program, which covers safety related valves. The IST Program will determine what safety related valves need to be cycled, and at what frequency. The number of valves tested is expected to decrease as a result of this change because not all power operated valves are required to change position to mitigate design basis events or support safe shutdown conditions. Power operated valves that are not required to change position are classified as passive valves by the IST Program and are not required to be cycled. The IST Program determines the frequency of safety related valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of testable valves, is consistent with standard industry practices and guidelines. The expected reduction in the number of valves tested is a less restrictive change.

8. A new surveillance requirement, SR 4.7.3.1.b, will be added. This surveillance requirement will require verification that all automatic valves associated with the RBCCW System actuate to the correct position following an actual or simulated actuation signal. The 18 month frequency is consistent with similar surveillance requirements such as SR 4.7.1.2.c.1 for Auxiliary Feedwater (AFW) automatic valves. This is a more restrictive change.

9. A new surveillance requirement, SR 4.7.3.1.c, will be added. This surveillance requirement will require verification that each RBCCW pump starts automatically on an actual or simulated actuation signal. The 18 month frequency is consistent with similar surveillance requirements such as SR 4.7.1.2.c.2 for the AFW pumps. This is a more restrictive change.

10. Performance of surveillance testing on a staggered test basis will not be retained. Based on the definition of staggered test basis in the Millstone Unit No. 2 Technical Specifications, the current frequency of every 31 days on a staggered test basis requires an RBCCW loop to be tested every 15 days (31 days divided by number of loops). There is no benefit to specifying performance of SR 4.7.3.1.a on a staggered test basis (i.e., one loop every 15 days) since the position of the RBCCW System (both loops) valves is required to be verified every 31 days. Therefore, the frequency of individual valve position verification will remain at 31 days. This is consistent with standard industry practices and guidelines.

Technical Specification 3.7.4.1

1. The word "independent" will be removed from the LCO. The level of independence between the SW System loops (trains) is a design feature of the SW System. The Millstone Unit No. 2 FSAR describes the approved degree of separation and level of independence between SW loops. If the approved degree of separation and level of independence are not maintained, the SW loops may not be operable as discussed in the Bases for this specification. Therefore, it is not necessary to include a requirement for the SW loops to be independent in Technical Specifications. This will not change the requirement for two SW loops to be operable.

2. The action requirement AOT to restore an inoperable SW loop will be increased from 48 hours to 72 hours. A 72 hour AOT is a standard time to restore safety significant equipment (e.g., one inoperable EDG, Technical Specification 3.8.1.1). It is consistent with standard industry guidelines contained in NUREG-1432 (Technical Specification 3.7.8, Service Water System). This is a less restrictive change.

3. The requirements of SRs 4.7.4.1.a.1, 4.7.4.1.a.2, and 4.7.4.1.a.3 will be deleted. These three monthly surveillance requirements check the SW pumps for proper operation and for potential degradation. However, since both loops and the associated pumps are normally operating, any potential malfunction or pump degradation should be readily apparent. Therefore, it is not necessary to include a monthly or quarterly surveillance requirement in Technical Specifications to check for proper SW pump operation or potential degradation. The IST Program will still monitor the SW pumps for proper operation on a regular interval. The IST Program specifies a minimum test performance interval of 92 days, which

may become more frequent based on equipment performance. Automatic operation of the SW pumps will be verified by proposed SR 4.7.4.1.c. This approach, to not include a surveillance requirement in Technical Specifications that requires monthly or quarterly testing of the SW pumps, is consistent with standard industry practices and guidelines, including NUREG-0212 and NUREG-1432 (Technical Specification 3.7.8). This is a less restrictive change.

4. SR 4.7.4.1.a.4, which verifies each SW loop is aligned to receive power from a separate operable emergency bus, will be deleted. This is redundant wording to the definition of operable (Definition 1.6) and can be removed without affecting any operability requirements. For a component to be operable it must have its normal and emergency power supply, except as provided by Technical Specification 3.0.5. In addition, it is not necessary to specify separate emergency busses. The degree of separation and the level of independence between the SW loops and the associated emergency busses is a design feature of the SW System and the emergency power distribution system. The Millstone Unit No. 2 FSAR describes the approved degree of separation and level of independence between SW loops and emergency busses. If the approved degree of separation and level of independence are not maintained, the SW loops may not be operable. Therefore, it is not necessary to include a check that each SW loop is aligned to separate emergency busses. Since this surveillance requirement is redundant to the current definition of operable, its deletion will not result in a technical change.

5. The requirements of SR 4.7.4.1.a.5 will be relocated to the proposed SR 4.7.4.1.a. This surveillance requirement will require verification that all SW valves in the flow path servicing safety related equipment that are not locked, sealed, or otherwise secured in position are in the correct position. This will encompass manual, remote, and automatically operated SW valves. Relocation of this requirement will not change test performance.

6. SR 4.7.4.1.a.6 will be deleted. The requirement to cycle all automatically operated valves will be addressed by the IST Program, which covers safety related valves. The IST Program will determine which safety related valves need to be cycled, and at what frequency. The number of valves tested is expected to decrease as a result of this change because not all automatically operated valves are required to change position to mitigate design basis events or support safe shutdown conditions. Automatic valves that are not required to change position are classified as passive valves by the IST Program and are not required to be cycled. The IST Program determines the frequency of safety related valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of testable valves, is consistent with standard industry practices and guidelines. The expected reduction in the number of valves tested is a less restrictive change.

7. SR 4.7.4.1.b will be deleted. The requirement to cycle all power operated valves will be addressed by the IST Program, which covers safety related valves. The IST Program will determine which safety related valves need to be cycled, and at what frequency. The number of valves tested is expected to decrease as a result of this change because not all power operated valves are required to change position to mitigate design basis events or support safe shutdown conditions. Power operated valves that are not required to change position are classified as passive valves by the IST Program and are not required to be cycled. The IST Program determines the frequency of safety related valve testing based on the ability to test valves during plant operation. Valves testable at power will be tested every 92 days. Valves not capable of testing during plant operation will be tested at a cold shutdown or refueling interval frequency. This approach, to use the IST Program to control the cycling of testable valves, is consistent with standard industry practices and guidelines. The expected reduction in the number of valves tested is a less restrictive change.

8. The footnote (*) associated with SR 4.7.4.1.b will be deleted. This footnote is no longer necessary since the associated surveillance requirement will be deleted, and it is no longer valid based on the time limitation of September 30, 1994. This is a non-technical change.

9. A new surveillance requirement, SR 4.7.4.1.b, will be added. This surveillance requirement will require verification that all automatic valves associated with the SW System actuate to the correct position following an actual or simulated actuation signal. The 18 month frequency is consistent with similar surveillance requirements such as SR 4.7.1.2.c.1 for AFW automatic valves. This is a more restrictive change.

10. A new surveillance requirement, SR 4.7.4.1.c, will be added. This surveillance requirement will require verification that each SW pump starts automatically on an actual or simulated actuation signal. The 18 month frequency is consistent with similar surveillance requirements such as SR 4.7.1.2.c.2 for the AFW pumps. This is a more restrictive change.

11. Performance of surveillance testing on a staggered test basis will not be retained. Based on the definition of staggered test basis in the Millstone Unit No. 2 Technical Specifications, the current frequency of every 31 days on a staggered test basis requires an SW loop to be tested every 15 days (31 days divided by number of loops). There is no benefit to specifying performance of SR 4.7.4.1.a on a staggered test basis (i.e., one loop every 15 days) since the position of the SW System (both loops) valves is required to be verified every 31 days. Therefore, the frequency of individual valve position verification will remain at 31 days. This is consistent with standard industry practices and guidelines.

Technical Specification Bases

The Bases for Technical Specifications 3.6.3.1, 3.7.3.1, and 3.7.4.1 will be expanded to describe the associated surveillance requirements and to include technical information that was not included in the proposed surveillance requirements. This approach, to include additional information that describes the surveillance requirements in the associated Bases, is consistent with NUREG-1432.

The Bases for Technical Specifications 3.7.3.1 and 3.7.4.1 currently contain descriptions of the pump surveillance requirements. These descriptions discuss how instrument uncertainty is applied to the pump acceptance criteria values currently contained in the surveillance requirements. Since the proposed changes will remove the pump acceptance criteria values from the surveillance requirements, it is no longer necessary to include a discussion of instrument uncertainty in the associated Bases. This information will be removed from the associated Bases. The IST Program will address instrument uncertainty in the pump acceptance criteria.

Safety Summary

The proposed changes will remove redundant testing requirements that are already addressed by the IST Program, which is required pursuant to Technical Specification 4.0.5. The IST Program addresses safety related components that are used to shut down the reactor to the safe shutdown condition, maintain the safe shutdown condition, and mitigate the consequences of the design basis accidents. In addition, the IST Program has been expanded to include components required to bring the reactor to a cold shutdown condition, maintain the reactor at cold shutdown, and provide long term post accident core cooling. Any changes to these requirements will be evaluated in accordance with 10 CFR 50.55a(f).

The proposed changes will use the IST Program (Technical Specification 4.0.5) to control the component acceptance criteria and determine the frequency of test performance. The acceptance criteria (e.g., valve stroke time, pump differential pressure, pump flowrate) is based on the component operation assumed in the associated safety analysis. The IST Program provides sufficient control of the acceptance criteria to ensure the associated components will perform as assumed in the safety analysis. The frequency of test performance will change from monthly to quarterly for numerous components, unless equipment performance indicates more frequent testing is required. In addition, the IST Program will be used to control test performance (e.g., how the pumps are started, how long the pumps are required to operate). This approach, to allow the IST Program to specify the acceptance criteria (based on design basis requirements), determine the test frequency, and control the testing process is consistent with standard industry practices and guidelines. This is illustrated in NUREG-1432 where many of the surveillance requirements use the IST Program to control the acceptance criteria and frequency of test performance.

The requirement to perform RBCCW and SW valve position surveillance testing on a staggered test basis will not be retained. Based on the definition of staggered test basis in the Millstone Unit No. 2 Technical Specifications, the current frequency of every 31 days on a staggered test basis requires a loop to be tested every 15 days (31 days divided by number of loops). There is no benefit to specifying performance on a staggered test basis (i.e., one loop every 15 days) since the position of system (both loops) valves is required to be verified every 31 days. Removal of this requirement will not result in a change in testing frequency. Elimination of testing on a staggered test basis for these components is consistent with standard industry practices and guidelines.

The proposed changes to the surveillance requirements that address 31 day, 92 day, or 18 month surveillance intervals for cycling system valves will result in a reduction in the number of valves tested. The reduction in valve population is the result of using the IST Program. This program only addresses safety related components, while the current requirements specify all system valves. As a result, not all valves will be tested on a regular basis. This is acceptable because the IST Program includes the valves required to change position for accident mitigation and safe shutdown of the unit. The valves excluded do not perform any safety related function or are not required to change position to perform a safety function. The IST Program will establish a quarterly frequency for testable valves and a cold shutdown or refueling interval (18 month) for valves not testable at power. The change in frequency is acceptable because it is consistent with current industry standards that are based on engineering judgment and operating experience, which have demonstrated no adverse impact on plant safety. The IST Program will monitor future component operation to determine if a more frequent surveillance interval is necessary. This is a less restrictive change. (SRs 4.6.3.1.1.a, 4.6.3.1.2.c, 4.6.3.1.2.d, 4.7.3.1.a.6, 4.7.3.1.b, 4.7.4.1.a.6, and 4.7.4.1.b)

The proposed Technical Specification change will remove SR 4.6.3.1.1.b, which is associated with post maintenance testing of the containment isolation valves. Post maintenance testing of a component following maintenance activities is already required to the extent necessary to ensure that the maintenance activity has not adversely affected component operability. It is implicit in the definition of operability and as such does not need to be restated separately in the surveillance requirement section of any Technical Specification.

The determination of the appropriate post maintenance testing will be based on the work performed. If the maintenance activities include work that could adversely affect component operation, the post maintenance testing will include the performance of the appropriate Technical Specification surveillance requirements prior to considering the component operable. The Technical Specification surveillance requirements are designed to verify operability, but their performance for post maintenance testing may not be necessary. By allowing flexibility in determining the appropriate testing, based on the work performed, unnecessary post maintenance testing can be avoided. This approach is consistent with standard industry practices and guidelines.

The requirement to verify RBCCW and SW valve position every 31 days will be retained. The proposed requirements will specify the types of valves to check, consistent with standard industry terminology (NUREG-1432). This will not result in any change to the valve population covered by the proposed surveillance requirements. (SRs 4.7.3.1.a.5 and 4.7.4.1.a.5)

Requirements to verify that the RBCCW and SW pumps and automatic valves actuate properly on an actual or simulated actuation signal every 18 months will be added. These more restrictive changes will provide additional assurance the pumps and valves will function as assumed for accident mitigation. The proposed frequency is consistent with similar Millstone Unit No. 2 surveillance requirements and with current industry standards (NUREG-1432) that are based on engineering judgment and operating experience, which have demonstrated no adverse impact on plant safety.

The proposed deletion of the SRs that address monthly testing of the RBCCW and SW System pumps will not adversely affect system operation. These systems and the associated pumps are normally in operation. As a result, any potential malfunction or pump degradation should be readily apparent. Pump operation will be verified quarterly by the IST Program. Therefore, it is not necessary to include monthly or quarterly surveillance requirements in Technical Specifications to check for proper pump operation and potential degradation. This approach is consistent with standard industry practices and guidelines (NUREG-0212 and NUREG-1432) that are based on engineering judgment and operating experience, which have demonstrated no adverse impact on plant safety. The IST Program will monitor pump operation to determine if a more frequent surveillance interval is necessary.

The proposed changes in the RBCCW and SW allowed outage times from 48 hours to 72 hours are consistent with generic industry standards (NUREG-0212 and NUREG-1432). As specified in Regulatory Guide (RG) 1.177,(1) Licensee initiated Technical Specification changes (surveillance frequencies and allowed outage times) that are consistent with currently approved staff positions (e.g., NUREG-1432) do not require the submittal of risk information in support of the proposed changes. DNC has performed a qualitative evaluation of the proposed changes and determined the changes will not adversely impact plant safety. (Technical Specifications 3.7.3.1 and 3.7.4.1)

Removal of the phrase "independent" from the LCOs for Technical Specifications 3.7.3.1 and 3.7.4.1 will not affect the requirement to have two operable loops. The degree of separation and the level of independence of the associated systems (RBCCW and SW) is a design feature. The Millstone Unit No. 2 Final Safety Analysis Report (FSAR) describes the approved degree of separation and level of independence between RBCCW and SW loops. If the approved degree of separation and level of independence are not maintained, the affected loops may not be operable as discussed in the Bases for these specifications. Therefore, it is not necessary to include a requirement to be independent in Technical Specifications.

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(1) Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998.

Removal of the requirement to verify each RBCCW and SW loop is aligned to receive power from a separate operable emergency bus will not affect loop operability. This is redundant to the Millstone Unit No. 2 Technical Specification definition of operable which requires normal and emergency power supplies, except as provided by Technical Specification 3.0.5. In addition, it is not necessary to specify separate emergency busses. The degree of separation and the level of independence between the respective subsystems and the associated emergency busses is a design feature. The Millstone Unit No. 2 FSAR describes the approved degree of separation and level of independence between the respective subsystems and emergency busses. If the approved degree of separation and level of independence are not maintained, the respective loops may not be operable. Therefore, it is not necessary to include a check that each loop is aligned to separate emergency busses.

The editorial changes proposed (e.g., combining requirements, deleting an expired footnote, renumbering a requirement) will not result in any technical changes to the associated requirements.

The current Bases for the Technical Specifications affected by the proposed changes contain descriptions of the pump surveillance requirements. These descriptions discuss how instrument uncertainty is applied to the pump acceptance criteria values currently contained in the surveillance requirements. Since the proposed changes will remove the pump acceptance criteria values from the surveillance requirements, it is no longer necessary to include a discussion of instrument uncertainty in the associated Bases. This information will be removed from the associated Bases. The IST Program will address instrument uncertainty in the pump acceptance criteria.

The Bases for these Technical Specifications will be expanded to describe the associated surveillance requirements and to include technical information that was not included in the proposed surveillance requirements. The additional guidance added to the Bases will ensure the requirements of the applicable Technical Specifications are applied correctly. This approach, to include additional information that describes the surveillance requirements in the associated Bases, is consistent with NUREG-1432.

The proposed changes to the Technical Specifications and Bases will not adversely affect the availability or operation of the equipment used to mitigate the design basis events. There will be no adverse effect on plant operation. The plant response to the design basis events will not change. The proposed changes are consistent with industry guidance contained in NUREG-0212, NUREG-1432, and NUREG-1482. The risk of a plant transient due to surveillance testing, personnel radiation exposure, and equipment degradation will be reduced as a result of the proposed changes. In addition, a review of the previous three years of Millstone Unit No. 2 surveillance test data for the pumps and valves affected by the proposed changes indicates that equipment performance issues were promptly corrected, and that the equipment is reliable. Therefore, there will be no adverse impact on public health and safety. Thus, the proposed changes are safe.

TABLE 1
SURVEILLANCE REQUIREMENT MATRIX

Technical Specification	Current SR	Proposed SR
3.6.3.1	4.6.3.1.1.a.1	4.6.3.1.a
	4.6.3.1.1.a.2	Deleted
	4.6.3.1.1.b	Deleted
	4.6.3.1.2.a	4.6.3.1.b
	4.6.3.1.2.b	4.6.3.1.b
	4.6.3.1.2.c	4.6.3.1.a
	4.6.3.1.2.d	Deleted
3.7.3.1	4.7.3.1.a.1	Deleted
	4.7.3.1.a.2	Deleted
	4.7.3.1.a.3	Deleted
	4.7.3.1.a.4	Deleted
	4.7.3.1.a.5	4.7.3.1.a
	4.7.3.1.a.6	Deleted
	4.7.3.1.b	Deleted
		4.7.3.1.b Added
		4.7.3.1.c Added
3.7.4.1	4.7.4.1.a.1	Deleted
	4.7.4.1.a.2	Deleted
	4.7.4.1.a.3	Deleted
	4.7.4.1.a.4	Deleted
	4.7.4.1.a.5	4.7.4.1.a
	4.7.4.1.a.6	Deleted
	4.7.4.1.b	Deleted
		4.7.4.1.b Added
		4.7.4.1.c Added

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and
Service Water Surveillance Requirements
Significant Hazards Consideration

License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and Service Water Surveillance Requirements
Significant Hazards Consideration

Description of License Amendment Request

Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to revise the Millstone Unit No. 2 Technical Specifications as described in this License Amendment Request. The majority of the proposed changes will revise the surveillance requirements associated with the Containment Isolation Valves (CIVs), Reactor Building Closed Cooling Water (RBCCW) System, and Service Water (SW) System. The proposed changes will remove redundant testing requirements that are already addressed by the Inservice Testing (IST) Program, which is required pursuant to Technical Specification 4.0.5, and use Technical Specification 4.0.5 to control the specific acceptance criteria and frequency of test performance. Additional proposed changes will remove the post maintenance testing requirements associated with the CIVs, revise the wording of the RBCCW and SW Systems Limiting Conditions for Operation (LCOs), and increase the allowed outage times for the RBCCW and SW Systems. Refer to Attachment 1 of this submittal for a detailed discussion of the proposed changes.

Significant Hazards Consideration

In accordance with 10 CFR 50.92, DNC has reviewed the proposed changes and has concluded that they do not involve a Significant Hazards Consideration (SHC). The basis for this conclusion is that the three criteria of 10 CFR 50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed Technical Specification changes associated with the limiting condition for operation requirements, surveillance requirements, and allowed outage times will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The ability of the equipment associated with the proposed changes to mitigate the design basis accidents will not be affected. The proposed changes to the limiting condition for operation requirements will not affect the equipment operability requirements. The proposed surveillance requirements are adequate to ensure proper operation of the associated accident mitigation equipment. Proper operation of the containment isolation valves will still be verified, as appropriate, following maintenance activities. The proposed allowed outage times are reasonable and consistent with standard industry guidelines to ensure the accident mitigation equipment will be restored in a timely manner. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 2 Final Safety Analysis Report, and the consequences of those events will not be affected. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

The additional proposed changes to the Technical Specifications (e.g., combining requirements, deleting an expired footnote, and renumbering a requirement) will not result in any technical changes to the current requirements. Therefore, these additional proposed changes will not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes to the Technical Specifications do not impact any system or component that could cause an accident. The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed changes will not alter the way any structure, system, or component functions, and will not alter the manner in which the plant is operated. There will be no effect on plant operation or accident mitigation equipment. The response of the plant and the operators following an accident will not be different. In addition, the proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed.

3. Involve a significant reduction in a margin of safety.

   The proposed Technical Specification changes associated with the limiting condition for operation requirements, surveillance requirements, and allowed outage times will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The equipment associated with the proposed Technical Specification changes will continue to be able to mitigate the design basis accidents as assumed in the safety analysis. The proposed surveillance requirements are adequate to ensure proper operation of the affected accident mitigation equipment. The proposed allowed outage times are reasonable and consistent with standard industry guidelines to ensure the accident mitigation equipment will be restored in a timely manner. In addition, the proposed changes will not affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings. The proposed Technical Specification changes, in conjunction with existing administrative controls (e.g., IST Program), will provide adequate control measures to ensure the accident mitigation functions are maintained. Therefore, the proposed changes will not result in a reduction in a margin of safety.

The additional proposed administrative changes to the Technical Specifications (e.g., combining requirements, deleting an expired footnote, and renumbering a requirement) will not result in any technical changes to the current requirements. Therefore, these additional changes will not result in a reduction in a margin of safety.

Attachment 3

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and
Service Water Surveillance Requirements
Marked Up Pages

License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and Service Water Surveillance Requirements
Marked Up Pages

The following Technical Specification pages have been proposed to be changed.

Technical Specification Section Number	Title(s) of Section(s)	Page and Revision Numbers
3/4.6.3.1	Containment Systems Containment Isolation Valves	3/4 6-15 Amend. 210 3/4 6-16 Amend. 210
3/4.7.3.1	Plant Systems Reactor Building Closed Cooling Water System	3/4 7-11 Amend. 236
3/4.7.4.1	Plant Systems Service Water System	3/4 7-12 Amend. 236
3/4.6	Containment Systems Containment Isolation Valves Bases	B 3/4 6-3e Amend. 216
3/4.7.3	Plant Systems Reactor Building Closed Cooling Water System Bases	B 3/4 7-3c Amend. 238 and PTSCR 2-18-01
3/4.7.4	Plant Systems Service Water System Bases	B 3/4 7-4a Amend. 236 and PTSCR 2-18-01

November 19, 1997

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate the affected penetration(s) within 4 hours by use of a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a closed manual valve(s) or blind flange(s); or

d. Be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1~~.1~~ Each Containment isolation valve ~~testable during plant operation~~ shall be demonstrated OPERABLE:

~~a. At least once per 92 days by:~~

~~1. Exercising each power operated valve through one complete cycle of full travel and measuring the isolation time, and~~

~~2. Exercising each manual valve, except those that are closed, through one complete cycle of full travel.~~

~~b. Immediately prior to returning the valve to service after maintenance, repair or replacement work is performed on the~~

INSERT A

*Locked or sealed closed valves may be opened on an intermittent basis under administrative controls.

INSERT A - Page 3/4 6-15

a. By verifying the isolation time of each power operated automatic containment isolation valve when tested pursuant to Specification 4.0.5.

b. At least once per 18 months by verifying each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.

SURVEILLANCE REQUIREMENTS (Continued)

valve or its associated actuator, control or power circuit by performance of the applicable cycling test, above.

4.6.3.1.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a containment isolation test signal, each isolation valve actuates to its isolation position,

b. Verifying that on a Containment Radiation-High signal, all containment purge valves actuate to their isolation position,

c. Exercising each power operated valve not testable during plant operation, through one complete cycle of full travel and measuring its isolation time, and

d. Exercising each manual valve not locked, sealed or otherwise secured in position through at least one complete cycle of full travel.

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3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 Two ~~independent~~ reactor building closed cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one reactor building closed cooling water loop inoperable, restore the inoperable loop to OPERABLE status within ~~48~~ 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 Each reactor building closed cooling water loop shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting (unless already operating) each pump from the control room,
2. Verifying that each pump develops at least 93% of the differential pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.
3. Verifying that each pump operates for at least 15 minutes,
4. Verifying that each loop is aligned to receive electrical power from separate OPERABLE emergency busses.
5. Verifying correct position of all valves servicing safety related equipment that are not locked, sealed or otherwise secured in position, and
6. Exercising all automatically operated valves servicing safety related equipment and testable during plant operation.

b. At least once per 18 months by exercising all power operated valves through one complete cycle of full travel.

INSERT B

INSERT B - Page 3/4 7-11

a. At least once per 31 days by verifying each reactor building closed cooling water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At least once per 18 months by verifying each reactor building closed cooling water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At least once per 18 months by verifying each reactor building closed cooling water pump starts automatically on an actual or simulated actuation signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two ~~independent~~ service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within ~~48~~ 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting (unless already operating) each pump from the control room,

2. Verifying that each pump develops at least 93% of the differential pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.

3. Verifying that each pump operates for at least 15 minutes,

4. Verifying that each loop is aligned to receive electrical power from separate OPERABLE emergency busses.

5. Verifying correct position of all valves servicing safety related equipment that are not locked, sealed or otherwise secured in position, and

6. Exercising all automatically operated valves servicing safety related equipment and testable during plant operation.

b. At least once per 18 months* by exercising all power operated valves through one complete cycle of full travel.

INSERT C

*Except that the surveillance requirement due no later than May 5, 1994, may be deferred until the next refueling outage, but no later than September 30, 1994, whichever is earlier.

INSERT C - Page 3/4 7-12

a. At least once per 31 days by verifying each service water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At least once per 18 months by verifying each service water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At least once per 18 months by verifying each service water pump starts automatically on an actual or simulated actuation signal.

3/4.6.2.1 CONTAINMENT SPRAY AND COOLING SYSTEMS (Continued)

pressure measurement instrument inaccuracies are already reflected in the Technical Specification acceptance criteria.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The Technical Requirements Manual contains the list of containment isolation valves (except the containment air lock and equipment hatch). Any changes to this list will be reviewed under 10CFR50.59 and approved by the committee(s) as described in the NUQAP Topical Report.

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment isolation valves are used to close all fluid (liquid and gas) penetrations not required for operation of the engineered safety feature systems, to prevent the leakage of radioactive materials to the environment. The fluid penetrations which may require isolation after an accident are categorized as Type P, O, or N. The penetration types are listed with the containment isolation valves in the Technical Requirements Manual.

Type P penetrations are lines that connect to the reactor coolant pressure boundary (Criterion 55 of 10CFR50, Appendix A). These lines are provided with two containment isolation valves, one inside containment, and one outside containment.

Type O penetrations are lines that are open to the containment internal atmosphere (Criterion 56 of 10CFR50, Appendix A). These lines are provided with two containment isolation valves, one inside containment, and one outside containment.

Type N penetrations are lines that neither connect to the reactor coolant pressure boundary nor are open to the containment internal atmosphere, but do form a closed system within the containment structure (Criterion 57 of 10CFR50, Appendix A). These lines are provided with single containment isolation valves outside containment. These valves are either remotely operated or locked closed manual valves.

Locked or sealed closed containment isolation valves may be opened on an intermittent basis provided appropriate administrative controls are established. The position of the NRC concerning acceptable administrative controls is contained in Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," and includes the following considerations:

(1) stationing an operator, who is in constant communication with the control room, at the valve controls,

(2) instructing this operator to close these valves in an accident situation, and

NO CHANGE FOR INFORMATION

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

(3) assuring that environmental conditions will not preclude access to close the valve and that this action will prevent the release of radioactivity outside the containment.

The appropriate administrative controls, based on the above considerations, to allow locked or sealed closed containment isolation valves to be opened are contained in the procedures that will be used to operate the valves. Entries should be placed in the Shift Manager Log when these valves are opened and closed. However, it is not necessary to log into any Technical Specification Action Statement for these valves, provided the appropriate administrative controls have been established.

If a locked or sealed closed containment isolation valve is opened while operating in accordance with Abnormal or Emergency Operating Procedures (AOPs and EOPs), it is not necessary to establish a dedicated operator. The AOPs and EOPs provide sufficient procedural control over the operation of the containment isolation valves.

Opening a locked or sealed closed containment isolation valve bypasses a plant design feature that prevents the release of radioactivity outside the containment. Therefore, this should not be done frequently, and the time the valve is opened should be minimized. As a general guideline, a locked or sealed closed containment isolation valve should not be opened longer than the time allowed to restore the valve to OPERABLE status, as stated in the action statement for LCO 3.6.3.1 "Containment Isolation Valves."

A discussion of the appropriate administrative controls for the containment isolation valves, that are expected to be opened during operation in MODES 1 through 4, is presented below.

Manual containment isolation valve 2-SI-463, safety injection tank (SIT) recirculation header stop valve, is opened to fill or drain the SITs and for Shutdown Cooling System (SDC) boron equalization. While 2-SI-463 is open, a dedicated operator, in continuous communication with the control room, is required.

When SDC is initiated, SDC suction isolation remotely operated valves 2-SI-652 and 2-SI-651 (inside containment isolation valve) and manual valve 2-SI-709 (outside containment isolation valve) are opened. 2-SI-651 is normally operated from the control room. While in Modes 1, 2 or 3, 2-SI-651 is closed with the closing and opening coils removed and stored to satisfy Appendix R requirements. It does not receive an automatic containment isolation closure signal, but is interlocked to prevent opening if Reactor Coolant System (RCS) pressure is greater than approximately 275 psia. When 2-SI-651 is opened from the control room, either one of the two required licensed (Reactor Operator) control room operators can be credited as the dedicated operator required for administrative control. It is not necessary to use a separate dedicated operator.

When valve 2-SI-709 is opened locally, a separate dedicated operator is not required to remain at the valve. 2-SI-709 is opened before 2-SI-651. Therefore, opening 2-SI-709 will not establish a connection between the RCS and the SDC System. Opening 2-SI-651 will connect the RCS and SDC System. If a problem then develops, 2-SI-651 can be closed from the control room.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

The administrative controls for valves 2-SI-651 and 2-SI-709 apply only during preparations for initiation of SDC, and during SDC operations. They are acceptable because RCS pressure and temperature are significantly below normal operating pressure and temperature when 2-SI-651 and 2-SI-709 are opened, and these valves are not opened until shortly before SDC flow is initiated. The penetration flowpath can be isolated from the control room by closing either 2-SI-652 or 2-SI-651, and the manipulation of these valves, during this evolution, is controlled by plant procedures.

The pressurizer auxiliary spray valve, 2-CH-517, can be used as an alternate method to decrease pressurizer pressure, or for boron precipitation control following a loss of coolant accident. When this valve is opened from the control room, either one of the two required licensed (Reactor Operator) control room operators can be credited as the dedicated operator required for administrative control. It is not necessary to use a separate dedicated operator.

The exception for 2-CH-517 is acceptable because the fluid that passes through this valve will be collected in the Pressurizer (reverse flow from the Pressurizer to the charging system is prevented by check valve 2-CH-431), and the penetration associated with 2-CH-517 is open during accident conditions to allow flow from the charging pumps. Also, this valve is normally operated from the control room, under the supervision of the licensed control room operators, in accordance with plant procedures.

A dedicated operator is not required when opening remotely operated valves associated with Type N fluid penetrations (Criterion 57 of 10CFR50, Appendix A). Operating these valves from the control room is sufficient. The main steam isolation valves (2-MS-64A and 64B), atmospheric steam dump valves (2-MS-190A and 190B), and the containment air recirculation cooler RBCCW discharge valves (2-RB-28.2A-D) are examples of remotely operated containment isolation valves associated with Type N fluid penetrations.

MSIV bypass valves 2-MS-65A and 65B are remotely operated MOVs, but while in MODE 1, they are closed with power to the valve motors removed via lockable disconnect switches located at their respective MCC to satisfy Appendix "R" requirements.

Local operation of the atmospheric steam dump valves (2-MS-190A and 190B), or other remotely operated valves associated with Type N fluid penetrations, will require a dedicated operator in constant communication with the control room, except when operating in accordance with AOPs or EOPs. Even though these valves can not be classified as locked or sealed closed, the use of a dedicated operator will satisfy administrative control requirements. Local operation of these valves with a dedicated operator is equivalent to the operation of other manual (locked or sealed closed) containment isolation valves with a dedicated operator.

The main steam supplies to the turbine driven auxiliary feedwater pump (2-MS-201 and 2-MS-202) are remotely operated valves associated with Type N fluid penetrations. These valves are maintained open during power operation. 2-MS-201 is maintained energized, so it can be closed from the control room, if necessary, for containment isolation. However, 2-MS-202 is deenergized

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

open by removing power to the valve's motor via a lockable disconnect switch to satisfy Appendix R requirements. Therefore, 2-MS-202 cannot be closed immediately from the control room, if necessary, for containment isolation. The disconect switch key to power for 2-MS-202 is stored in the Unit 2 control room, and can be used to re-power the valve at the MCC; this will allow the valve to be closed from the control room. It is not necessary to maintain a dedicated operator at 2-MS-202 because this valve is already in the required accident position. Also, the steam that passes through this valve should not contain any radioactivity. The steam generators provide the barrier between the containment and the atmosphere. Therefore, it would take an additional structural failure for radioactivity to be released to the environment through this valve.

Steam generator chemical addition valves, 2-FW-15A and 2-FW-15B, are opened to add chemicals to the steam generators using the Auxiliary Feedwater System (AFW). When either 2-FW-15A or 2-FW-15B is opened, a dedicated operator, in continuous communication with the control room, is required. Operation of these valves is expected during plant startup and shutdown.

The bypasses around the main steam supplies to the turbine driven auxiliary feedwater pump (2-MS-201 and 2-MS-202), 2-MS-458 and 2-MS-459, are opened to drain water from the steam supply lines. When either 2-MS-458 or 2-MS-459 is opened, a dedicated operator, in continuous communication with the control room, is required. Operation of these valves is expected during plant startup.

The containment station air header isolation, 2-SA-19, is opened to supply station air to containment. When 2-SA-19 is opened, a dedicated operator, in continuous communication with the control room, is required. Operation of this valve is only expected for maintenance activities inside containment.

The backup air supply master stop, 2-IA-566, is opened to supply backup air to 2-CH-517, 2-CH-518, 2-CH-519, 2-EB-88, and 2-EB-89. When 2-IA-566 is opened, a dedicated operator, in continuous communication with the control room, is required. Operation of this valve is only expected in response to a loss of the normal air supply to the valves listed.

The nitrogen header drain valve, 2-SI-045, is opened to depressurize the containment side of the nitrogen supply header stop valve, 2-SI-312. When 2-SI-045 is opened, a dedicated operator, in continuous communication with the control room, is required. Operation of this valve is only expected after using the high pressure nitrogen system to raise SIT nitrogen pressure.

The containment waste gas header test connection isolation valve, 2-GR-63, is opened to sample the primary drain tank for oxygen and nitrogen. When 2-GR-63 is opened, a dedicated operator, in continuous communication with the control room, is required. Operation of this valve is expected during plant startup and shutdown.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

The upstream vent valves for the steam generator atmospheric dump valves, 2-MS-369 and 2-MS-371, are opened during steam generator safety valve set point testing to allow steam header pressure instrumentation to be placed in service. When either 2-MS-369 or 2-MS-371 is opened, a dedicated operator in continuous communication with the control room is required.

The determination of the appropriate administrative controls for these containment isolation valves included an evaluation of the expected environmental conditions. This evaluation has concluded environmental conditions will not preclude access to close the valve, and this action will prevent the release of radioactivity outside of containment through the respective penetration.

The containment purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Such a demonstration would require justification of the mechanical operability of the purge valves and consideration of the appropriateness of the electrical override circuits. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. The containment purge supply and exhaust isolation valves are sealed closed by removing power from the valves. This is accomplished by pulling the control power fuses for each of the valves. The associated fuse blocks are then locked. This is consistent with the guidance contained in NUREG-0737 Item II.E.4.2 and Standard Review Plan 6.2.4, "Containment Isolation System," Item II.f.

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Surveillance Requirement 4.6.3.1.a verifies the isolation time of each power operated automatic containment isolation valve is within limits to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and surveillance frequency are in accordance with the Inservice Testing Program.

Surveillance Requirement 4.6.3.1.b demonstrate that each automatic containment isolation valve actuates to the isolation position on an actual or simulated containment isolation signal [containment isolation actuation signal (CIAS) or containment high radiation actuation signal (containment purge valves only)]. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillance was performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

a feedwater isolation signal since the steam line break accident analysis credits them in prevention of feed line volume flashing in some cases. Feedwater pumps are assumed to trip immediately with an MSI signal.

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

The atmospheric dump valve (ADV) lines provide a method to maintain the unit in HOT STANDBY, and to replace or supplement the condenser steam dump valves to cool the unit to Shutdown Cooling (SDC) entry conditions. Each ADV line contains an air operated ADV, and an upstream manual isolation valve. The manual isolation valves are normally open, and the ADVs closed. The ADVs, which are normally operated from the main control room, can be operated locally using a manual handwheel.

An ADV line is OPERABLE if local manual operation of the associated valves can be used to perform a controlled release of steam to the atmosphere. This is consistent with the LOCA analysis which credits local manual operation of the ADV lines for accident mitigation.

3/4.7.1.8 STEAM GENERATOR BLOWDOWN ISOLATION VALVES

The steam generator blowdown isolation valves will isolate steam generator blowdown on low steam generator water level. An auxiliary feedwater actuation signal will also be generated at this steam generator water level. Isolation of steam generator blowdown will conserve steam generator water inventory following a loss of main feedwater. The steam generator blowdown isolation valves will also close automatically upon receipt of a containment isolation signal or a high radiation signal (steam generator blowdown or condenser air ejector discharge).

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°F and are sufficient to prevent brittle fracture.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

The OPERABILITY of the Reactor Building Closed Cooling Water (RBCCW) System ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

The RBCCW loops are redundant of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a design basis accident, one RBCCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two RBCCW loops must be OPERABLE, and independent to the extent necessary to ensure that a single failure will not result in the unavailability

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM (Continued)

of both RBCCW loops. At least one RBCCW loop will operate assuming the worst single active failure occurs following a design basis accident coincident with a loss of offsite power, or the worst single passive failure occurs during post-loss of coolant accident long term cooling. System design is assumed to mitigate the single active failure. System design or operator action is assumed to mitigate the passive failure.

The RBCCW System has numerous cross connection points between the redundant loops, with manual valve isolation capability. When these valves are opened, the two system loops are no longer independent. The loss of independence will result in one large RBCCW loop. This may adversely impact the ability of the RBCCW System to mitigate the design basis events if a single failure, active or passive, occurs. Opening the manual cross-connection valves during normal operation should be evaluated to ensure system stability, minimum component cooling flow requirements, and the ability to mitigate the design basis events coincident with a single failure are maintained. Continuous operation with cross-connection valves open is acceptable if the configuration has been evaluated and protection against a single failure can be demonstrated. (Several system configurations that have been evaluated and determined acceptable for continuous plant operation are identified below). If opening a cross-connection valve will result in a plant configuration that does not provide adequate protection against a single failure, the following guidance applies. If only the manual cross-connect valves have been opened, and the RBCCW System is in a normal configuration otherwise, with all system equipment OPERABLE, one RBCCW loop should be considered inoperable and the ACTION requirements of Technical Specification 3.7.3.1 applied. If the RBCCW System is not in a normal configuration otherwise and/or not all equipment is OPERABLE, both RBCCW loops should be considered inoperable and the ACTION requirements of Technical Specification 3.0.3 applied.

The loss of loop independence is equivalent to the situation where one loop is inoperable. If one loop is inoperable, the remaining OPERABLE loop will be able to meet all design basis accident functions, assuming an additional single failure does not occur. If the loops are not independent, the remaining single large OPERABLE loop will be able to meet all design basis accident functions, assuming a single failure does not occur. Operation in a plant configuration where protection against a single failure can not be shown is acceptable provided the time period in that configuration is limited to less than the Technical Specification specified allowed outage time. It is acceptable to operate in the off normal plant configurations identified in the ACTION requirements for the time periods specified due to the low probability of occurrence of a design basis event concurrent with a single failure during this limited time period. The allowed outage time for one inoperable RBCCW loop provides an appropriate limit for continued operation with only one OPERABLE RBCCW loop, and can be applied to a plant configuration where only loop independence has been compromised. The loop determined to be inoperable should be the loop that results in the most adverse plant configuration with respect to the availability of accident mitigation equipment. Restoration of loop independence within the time constraints of the allowed outage time is required, or a plant shutdown is necessary.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM (Continued)

It is acceptable to operate with the RBCCW pump minimum flow valves (2-RB 107A, 2-RB-107B, 2-RB-107C), RBCCW pump sample valves (2-RB-56A, 2-RB-56B, and 2-RB-56C), and the RBCCW pump radiation monitor stop valves (2-RB-39, 2-RB-41, and 2-RB-43) open. An active single failure will not adversely impact both RBCCW loops with these valves open. In addition, protection against a passive single failure after the initiation of post - loss of coolant accident long term cooling is achieved by manually closing these accessible valves, as directed by the emergency operating procedures. In addition, operation with RBCCW chemical addition valves (2-RB-50A and 2-RB-50B) open during chemical addition evolutions is acceptable since these normally closed valves are opened to add chemicals to the RBCCW and then closed as directed by normal operating procedures. Therefore, operation with these valves open does not affect OPERABILITY of the RBCCW loops.

~~The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the reactor building closed cooling water pumps differential pressure test, Surveillance Requirement 4.7.3.1.a.2, a substantial flow test, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the reactor building closed cooling water pumps was developed assuming a 7% degraded pump from the actual pump curves. Flow measurement instrument inaccuracy for the reactor building closed cooling water pumps have been accounted for in the design basis hydraulic analysis. Pressure measurement instrument inaccuracy for the reactor building closed cooling water pumps is accounted for in the acceptance criteria contained in the surveillance procedure.~~

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3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the Service Water (SW) System ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

The SW loops are redundant of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a design basis accident, one SW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two SW loops must be OPERABLE, and independent to the extent necessary to ensure that a single failure will not result in the unavailability of both SW loops. At least one SW loop will operate assuming the worst single active failure occurs following a design basis accident coincident with a loss of offsite power, or the worst single passive failure occurs post - loss of coolant accident long term cooling. System design is assumed to mitigate the single active failure. System design or operator action is assumed to mitigate the passive failure.

<u>INSERT E - Page B 3/4 7-3c</u>

Surveillance Requirement 4.7.3.1.a verifies the correct alignment for manual, power operated, and automatic valves in the RBCCW System flow paths to provide assurance that the proper flow paths exist for RBCCW operation. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This frequency has been shown to be acceptable through operating experience.

Surveillance Requirements 4.7.3.1.b and 4.7.3.1.c demonstrate that each automatic RBCCW valve actuates to the required position on an actual or simulated actuation signal and that each RBCCW pump starts on receipt of an actual or simulated actuation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillances were performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

3/4.7.4 SERVICE WATER SYSTEM (Continued)

The SW System has numerous cross connection points between the redundant loops, with manual valve isolation capability. When these valves are opened, the two system loops are no longer independent. The loss of independence will result in one large SW loop. This may adversely impact the ability of the SW System to mitigate the design basis events if a single failure, active or passive, occurs. Opening the manual cross-connection valves during normal operation should be evaluated to ensure system stability, minimum component cooling flow requirements, and the ability to mitigate the design basis event coincident with a single failure are maintained. Continuous operation with cross-connection valves open is acceptable if the configuration has been evaluated and protection against a single failure can be demonstrated. (Several system configurations that have been evaluated and determined acceptable for continuous plant operation are identified below). If opening a cross-connection valve will result in a plant configuration that does not provide adequate protection against a single failure, the following guidance applies. If only the manual cross-connect valves have been opened, and the SW System is in a normal configuration otherwise, with all system equipment OPERABLE, one SW loop should be considered inoperable and the ACTION requirements of Technical Specification 3.7.4.1 applied. If the SW System is not in a normal configuration otherwise and/or not all equipment is OPERABLE, both SW loops should be considered inoperable and the ACTION requirements of Technical Specification 3.0.3 applied.

The loss of loop independence is equivalent to the situation where one loop is inoperable. If one loop is inoperable, the remaining OPERABLE loop will be able to meet all design basis accident functions, assuming an additional single failure does not occur. If the loops are not independent, the remaining single large OPERABLE loop will be able to meet all design basis accident functions, assuming a single failure does not occur. Operation in a plant configuration where protection against a single failure can not be shown is acceptable provided the time period in that configuration is limited to less then the Technical Specification specified allowed outage time. It is acceptable to operate in the off normal plant configurations identified in the ACTION requirements for the time periods specified due to the low probability of occurrence of a design basis event concurrent with a single failure during this limited time period. The allowed outage time for one inoperable SW loop provides an appropriate limit for continued operation with only one OPERABLE SW loop, and can be applied to a plant configuration where only loop independence has been compromised. The loop determined to be inoperable should be the loop that results in the most adverse plant configuration with respect to the availability of accident mitigation equipment. Restoration of loop independence within the time constraints of the allowed outage time is required, or a plant shutdown is necessary.

It is acceptable to operate with the SW header supply valves to sodium hypochlorite (2-SW-84A and 2-SW-84B) and the SW header supply valves to the north and south filters (2-SW-298 and 2-SW-299) open. Protection against a single failure (active or passive after the initiation of post - loss of coolant accident long term cooling) with these valves open is provided by the flow restricting orifices contained in these lines. Therefore, operation with these valves open does not affect OPERABILITY of the SW loops.

3/4.7.4 SERVICE WATER SYSTEM (Continued)

~~The Technical Specification Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analysis are met and that subsystem OPERABILITY is maintained. The purpose of the service water pumps differential pressure test, Surveillance Requirement 4.7.4.1.a.2, a substantial flow test, is to ensure that the pumps have not degraded to a point where the accident analysis would be adversely impacted. The surveillance requirement acceptance criteria for the service water pumps was developed assuming a 7% degraded pump from the actual pump curves. Flow and pressure measurement instrument inaccuracies for the service water pumps have been accounted for in the design basis hydraulic analysis. It is not necessary to account for flow and pressure measurement instrument inaccuracies in the acceptance criteria contained in the surveillance procedure.~~

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3/4.7.5 FLOOD LEVEL

The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure operator accessibility to the intake structure action to provide pump motor protection will be initiated when the water level reaches plant grade level.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm ± 10%.

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Surveillance Requirement 4.7.4.1.a verifies the correct alignment for manual, power operated, and automatic valves in the Service Water (SW) System flow paths to provide assurance that the proper flow paths exist for SW operation. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This frequency has been shown to be acceptable through operating experience.

Surveillance Requirements 4.7.4.1.b and 4.7.4.1.c demonstrate that each automatic SW valve actuates to the required position on an actual or simulated actuation signal and that each SW pump starts on receipt of an actual or simulated actuation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillances were performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

Docket No. 50-336
B18462

Attachment 4

Millstone Nuclear Power Station, Unit No. 2

License Basis Document Change Request 2-14-01
Containment Isolation, Reactor Building Closed Cooling Water, and
Service Water Surveillance Requirements
Retyped Pages

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate the affected penetration(s) within 4 hours by use of a deactivated automatic valve(s) secured in the isolation position(s), or

c. Isolate the affected penetration(s) within 4 hours by use of a closed manual valve(s) or blind flange(s); or

d. Be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each containment isolation valve shall be demonstrated OPERABLE:

a. By verifying the isolation time of each power operated automatic containment isolation valve when tested pursuant to Specification 4.0.5.

b. At least once per 18 months by verifying each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.

*Locked or sealed closed valves may be opened on an intermittent basis under administrative controls.

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3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 Two reactor building closed cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one reactor building closed cooling water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 Each reactor building closed cooling water loop shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying each reactor building closed cooling water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At least once per 18 months by verifying each reactor building closed cooling water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At least once per 18 months by verifying each reactor building closed cooling water pump starts automatically on an actual or simulated actuation signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying each service water manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At least once per 18 months by verifying each service water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

c. At least once per 18 months by verifying each service water pump starts automatically on an actual or simulated actuation signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

The upstream vent valves for the steam generator atmospheric dump valves, 2-MS-369 and 2-MS-371, are opened during steam generator safety valve set point testing to allow steam header pressure instrumentation to be placed in service. When either 2-MS-369 or 2-MS-371 is opened, a dedicated operator in continuous communication with the control room is required.

The determination of the appropriate administrative controls for these containment isolation valves included an evaluation of the expected environmental conditions. This evaluation has concluded environmental conditions will not preclude access to close the valve, and this action will prevent the release of radioactivity outside of containment through the respective penetration.

The containment purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Such a demonstration would require justification of the mechanical operability of the purge valves and consideration of the appropriateness of the electrical override circuits. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. The containment purge supply and exhaust isolation valves are sealed closed by removing power from the valves. This is accomplished by pulling the control power fuses for each of the valves. The associated fuse blocks are then locked. This is consistent with the guidance contained in NUREG-0737 Item II.E.4.2 and Standard Review Plan 6.2.4, "Containment Isolation System," Item II.f.

Surveillance Requirement 4.6.3.1.a verifies the isolation time of each power operated automatic containment isolation valve is within limits to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and surveillance frequency are in accordance with the Inservice Testing Program.

Surveillance Requirement 4.6.3.1.b demonstrate that each automatic containment isolation valve actuates to the isolation position on an actual or simulated containment isolation signal [containment isolation actuation signal (CIAS) or containment high radiation actuation signal (containment purge valves only)]. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillance was performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

3/4.7.3 REACTOR BUILDING CLOSED COOLING WATER SYSTEM (Continued)

It is acceptable to operate with the RBCCW pump minimum flow valves (2-RB 107A, 2-RB-107B, 2-RB-107C), RBCCW pump sample valves (2-RB-56A, 2-RB-56B, and 2-RB-56C), and the RBCCW pump radiation monitor stop valves (2-RB-39, 2-RB-41, and 2-RB-43) open. An active single failure will not adversely impact both RBCCW loops with these valves open. In addition, protection against a passive single failure after the initiation of post - loss of coolant accident long term cooling is achieved by manually closing these accessible valves, as directed by the emergency operating procedures. In addition, operation with RBCCW chemical addition valves (2-RB-50A and 2-RB-50B) open during chemical addition evolutions is acceptable since these normally closed valves are opened to add chemicals to the RBCCW and then closed as directed by normal operating procedures. Therefore, operation with these valves open does not affect OPERABILITY of the RBCCW loops.

Surveillance Requirement 4.7.3.1.a verifies the correct alignment for manual, power operated, and automatic valves in the RBCCW System flow paths to provide assurance that the proper flow paths exist for RBCCW operation. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This frequency has been shown to be acceptable through operating experience.

Surveillance Requirements 4.7.3.1.b and 4.7.3.1.c demonstrate that each automatic RBCCW valve actuates to the required position on an actual or simulated actuation signal and that each RBCCW pump starts on receipt of an actual or simulated actuation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillances were performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the Service Water (SW) System ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM (Continued)

The SW loops are redundant of each other to the degree that each has separate controls and power supplies and the operation of one does not depend on the other. In the event of a design basis accident, one SW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two SW loops must be OPERABLE, and independent to the extent necessary to ensure that a single failure will not result in the unavailability of both SW loops. At least one SW loop will operate assuming the worst single active failure occurs following a design basis accident coincident with a loss of offsite power, or the worst single passive failure occurs post - loss of coolant accident long term cooling. System design is assumed to mitigate the single active failure. System design or operator action is assumed to mitigate the passive failure.

The SW System has numerous cross connection points between the redundant loops, with manual valve isolation capability. When these valves are opened, the two system loops are no longer independent. The loss of independence will result in one large SW loop. This may adversely impact the ability of the SW System to mitigate the design basis events if a single failure, active or passive, occurs. Opening the manual cross-connection valves during normal operation should be evaluated to ensure system stability, minimum component cooling flow requirements, and the ability to mitigate the design basis event coincident with a single failure are maintained. Continuous operation with cross-connection valves open is acceptable if the configuration has been evaluated and protection against a single failure can be demonstrated. (Several system configurations that have been evaluated and determined acceptable for continuous plant operation are identified below). If opening a cross-connection valve will result in a plant configuration that does not provide adequate protection against a single failure, the following guidance applies. If only the manual cross-connect valves have been opened, and the SW System is in a normal configuration otherwise, with all system equipment OPERABLE, one SW loop should be considered inoperable and the ACTION requirements of Technical Specification 3.7.4.1 applied. If the SW System is not in a normal configuration otherwise and/or not all equipment is OPERABLE, both SW loops should be considered inoperable and the ACTION requirements of Technical Specification 3.0.3 applied.

The loss of loop independence is equivalent to the situation where one loop is inoperable. If one loop is inoperable, the remaining OPERABLE loop will be able to meet all design basis accident functions, assuming an additional single failure does not occur. If the loops are not independent, the remaining single large OPERABLE loop will be able to meet all design basis accident functions, assuming a single failure does not occur. Operation in a plant configuration where protection against a single failure can not be shown is acceptable provided the time period in that configuration is limited to less then the Technical Specification specified allowed outage time. It is acceptable to operate in the off normal plant configurations identified in the ACTION requirements for the time periods specified due to the low probability of occurrence of a design basis event concurrent with a single failure during this limited time period. The allowed outage time for one inoperable SW loop provides an appropriate limit for continued operation with only one OPERABLE SW loop, and can be applied to a plant configuration where only loop independence has been compromised. The loop

3/4.7.4 SERVICE WATER SYSTEM (Continued)

determined to be inoperable should be the loop that results in the most adverse plant configuration with respect to the availability of accident mitigation equipment. Restoration of loop independence within the time constraints of the allowed outage time is required, or a plant shutdown is necessary.

It is acceptable to operate with the SW header supply valves to sodium hypochlorite (2-SW-84A and 2-SW-84B) and the SW header supply valves to the north and south filters (2-SW-298 and 2-SW-299) open. Protection against a single failure (active or passive after the initiation of post - loss of coolant accident long term cooling) with these valves open is provided by the flow restricting orifices contained in these lines. Therefore, operation with these valves open does not affect OPERABILITY of the SW loops.

Surveillance Requirement 4.7.4.1.a verifies the correct alignment for manual, power operated, and automatic valves in the Service Water (SW) System flow paths to provide assurance that the proper flow paths exist for SW operation. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve automatically repositions within the proper stroke time. This surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day frequency is appropriate because the valves are operated under procedural control and an improper valve position would only affect a single train. This frequency has been shown to be acceptable through operating experience.

Surveillance Requirements 4.7.4.1.b and 4.7.4.1.c demonstrate that each automatic SW valve actuates to the required position on an actual or simulated actuation signal and that each SW pump starts on receipt of an actual or simulated actuation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month frequency is based on the need to perform these surveillances under the conditions that apply during a plant outage and the potential for unplanned transients if the surveillances were performed with the reactor at power. The 18 month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the Engineered Safety Feature Actuation System (ESFAS) testing, and equipment performance is monitored as part of the Inservice Testing Program.

3/4.7.5 FLOOD LEVEL

The service water pump motors are normally protected against water damage to an elevation of 22 feet. If the water level is exceeding plant grade level or if a severe storm is approaching the plant site, one service water pump motor will be protected against flooding to a minimum elevation of 28 feet to ensure that this pump will continue to be capable of removing decay heat from the reactor. In order to ensure operator accessibility to the intake structure action to provide pump motor protection will be initiated when the water level reaches plant grade level.

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions.

The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The LCO is modified by a footnote allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in constant communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated.

The control room radiological dose calculations use the conservative minimum acceptable flow of 2250 cfm based on the flowrate surveillance requirement of 2500 cfm $\pm$ 10%.