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

Dominion

OCT - 1 2001

Docket No. 50-423
B18484

RE: 10 CFR 50.90

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

Millstone Power Station, Unit No. 3
Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage

Introduction

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the Millstone Unit No. 3 Technical Specifications. DNC is proposing to change Technical Specification 3.4.6.2, "Reactor Coolant System - Operational Leakage." The Bases for this Technical Specification will also be modified to reflect this change.

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.

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 a surveillance requirement, and that the amendment request meets the following specific criteria.

A001

(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 Limiting Condition for Operation (LCO) and Surveillance Requirements (SRs) associated with verification of Reactor Coolant System Operational Leakage. However, the operability requirements for equipment associated with measuring Reactor Coolant System Operation Leakage 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 and SRs associated with verification of Reactor Coolant System Operational Leakage. However, the operability requirements for equipment associated with measuring Reactor Coolant System Operation Leakage 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.

Conclusions

The proposed changes were evaluated and we have concluded that they are safe. The proposed changes do not involve an 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).

Site Operations Review Committee and Nuclear Safety Assessment Board

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

Schedule

We request issuance of this amendment for Millstone Unit No. 3 prior to June 30, 2002, with the amendment to be implemented within 60 days of issuance. This will allow Millstone Unit No. 3 to use the proposed changes during the next refueling outage currently scheduled in early September of 2002.

State Notification

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, Vice President
Nuclear Technical Services - Millstone

Sworn to and subscribed before me

this 1st day of October, 2001

Notary Public

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

Attachments (4)

cc: See next page

cc: H. J. Miller, Region I Administrator
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
NRC Senior Resident Inspector, Millstone Unit No. 3

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

Docket No. 50-423
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Attachment 1

Millstone Power Station, Unit No. 3

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage
Discussion of Proposed Changes and Safety Summary

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage
Discussion of Proposed Changes and Safety Summary

Introduction

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed changes into the Millstone Unit No. 3 Technical Specifications. DNC is proposing to change Technical Specification 3.4.6.2, "Reactor Coolant System - Operational Leakage." The Bases for this Technical Specification will be modified to address the proposed changes. Each proposed change will be discussed.

Technical Specification Changes

1. An "*" will be added to Limiting Condition of Operation (LCO) 3.4.6.2.f and an associated footnote will be added to the bottom of the affected page. The existing LCO specifies 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System (RCS) pressure of 2250 $\pm$ 20 psia from any RCS Pressure Isolation Valve (PIV) specified in Table 3.4-1. LCO 3.4.6.2.f will be revised such that it does not apply to those valves in the Residual Heat Removal (RHR) System flow path when in, or during the transition to or from, the RHR mode of operation. This is a less restrictive change.

2. Surveillance Requirements (SRs) 4.4.6.2.1.a and 4.4.6.2.1.b will be removed, and the word "DELETED" will be added for each. Neither of these surveillances verify compliance with the leakage limits specified in LCO 3.4.6.2. This LCO has specific leakage values that cannot be verified by monitoring the containment sump level or the containment atmosphere particulate radioactivity. These instruments provide indication of leakage, but cannot provide a value of leakage with the required accuracy to ensure compliance with the LCO. The operability and SRs for these instruments are addressed by Technical Specification 3.4.6.1, "Leakage Detection Systems."

   These changes are similar to changes associated with Millstone Unit No. 2 License Amendment Number 215[(1)] in which RCS leakage SRs 4.4.6.2.a and 4.4.6.2.b were deleted. Additionally, these changes are consistent with NUREG-1431, "Standard Technical Specifications Westinghouse Plants." This is a less restrictive change.

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(1) D. McDonald letter to M. Bowling, "Millstone Nuclear Power Station, Unit No. 2 - Issuance of Amendment (TAC No. M99504)," Amendment No. 215, dated May 26, 1998.

3. SR 4.4.6.2.1.d will be revised such that verification of RCS water inventory balance is only performed during steady-state operation. The phrase "within 12 hours of achieving steady-state operation and" will be added to the SR following the discussion of water inventory balance. The phrase "thereafter during steady-state operation" will be added after the existing 72 hour requirement for performance of the water inventory balance.

   Verification of RCS leakage by performing a water inventory balance when steady-state conditions do not exist is difficult and does not always result in an accurate assessment of RCS leakage. Performance of RCS water inventory balance during steady-state operation will result in a more precise verification of the RCS leakage and compliance with the requirements of LCO 3.4.6.2. The revised SR will require verification of RCS water inventory balance within 12 hours of achieving steady-state operation and at least once per 72 hours thereafter. This is a less restrictive change.

4. SR 4.4.6.2.2.c will be removed, and the word "DELETED" will be added. This SR, which requires that prior to returning an RCS PIV to service following maintenance, repair or replacement work on a valve the operability of the valve will be verified by ensuring that valve leakage is within its limit is not necessary. Post maintenance testing after completion of valve work, which is controlled by plant procedures, would specify this verification if the associated work could adversely affect valve operability (e.g., affect valve leakage). This verification is necessary prior to considering the valve operable after completion of maintenance activities that could affect valve integrity or isolation capabilities. This testing will continue to be performed, as required, to ensure component operability following maintenance activities. The proposed change will provide flexibility in determining the appropriate post maintenance testing based on the work performed. This is a less restrictive change.

5. A "(1)" will be added to SRs 4.4.6.2.1 and 4.4.6.2.2 and to the existing footnote for these SRs. The existing footnote states that the provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or 4. Addition of an identifier for the existing footnote will minimize the potential for confusion with respect to the applicability of this footnote due to the addition of another footnote for SR 4.4.6.2.2 (item 6 below). This is not a technical change.

6. A "(2)" will be added to SR 4.4.6.2.2 and an associated footnote will be added to the bottom of the affected page. Footnote "(2)" will state "This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from, the shutdown cooling mode of operation." The proposed change will clarify that verification of RCS leakage for PIVs in the RHR flow path is not required when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. This is a less restrictive change.

7. Additional guidance will be added to the Bases to discuss the requirements of SR 4.4.6.2.1.d for performance of an RCS water inventory balance. The following discussion will be added to the Basis of Technical Specification 3.4.6.2: "Steady-state operation is defined as stable Reactor Coolant System temperature, reactor power level, pressurizer and makeup tank levels, and makeup and letdown flows. The 12 hour allowance time for Surveillance Requirement 4.4.6.2.1.d provides sufficient time to collect and process all necessary data after stable plant conditions are established."

   Additional guidance will be added to the Bases to discuss the requirements of SR 4.4.6.2.2 for RCS PIVs which must be leak-tested at elevated RCS pressures. The following discussion will be added to the Basis of Technical Specification 3.4.6.2: "Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions for performance of SR 4.4.6.2.2 for Reactor Coolant System (RCS) pressure isolation valves which can only be leak-tested at elevated RCS pressures. The requirements of Surveillance Requirement 4.4.6.2.2 to verify that a pressure isolation valve is OPERABLE shall be performed within 24 hours after the required RCS pressure has been met."

Safety Summary

The proposed changes to Millstone Unit No. 3 Technical Specification 3.4.6.2 do not pose a condition adverse to safety and do not create any adverse safety consequences. The rationale for this conclusion is provided below.

The proposed change to LCO 3.4.6.2.f to clarify that this LCO does not apply to those valves in the RHR flow path during shutdown cooling operations is consistent with current operating practices. These valves are maintained open during shutdown cooling operations providing heat removal for the reactor core. Verification of the leakage limits for those valves in the RHR flow path during shutdown cooling operations does not provide any benefit in maintaining RCS inventory or in minimizing the release of radioactive material from the RCS. Upon securing shutdown cooling operations the requirements of Technical Specification 3.4.6.2.f will again apply to valves in the RHR flow path. Therefore, the proposed change is safe.

The proposed changes to delete SRs 4.4.6.2.1.a and 4.4.6.2.1.b do not affect how compliance with the leakage limits contained in Technical Specification 3.4.6.2 is verified. The containment atmosphere radioactivity monitor and containment drain sump inventory and discharge provide early indication that RCS leakage exists, but do not provide the specific information (amount of leakage) necessary to verify operation within the leakage limits. Performance of an RCS water inventory balance (SR 4.4.6.2.1.d) will be used to verify compliance with the leakage limits. SRs 4.4.6.1.a and 4.4.6.1.b verify operability of the containment atmosphere radioactivity monitor and containment drain sump inventory and discharge monitoring equipment. Therefore, these changes do not reduce the operability requirements for any equipment used to monitor RCS leakage. Additionally, performance of SR 4.4.6.2.1.d, water inventory balance, provides the means necessary to measure RCS leakage and verify that the

RCS is being operated within its leakage limits. Therefore, the proposed changes are safe. These changes are consistent with NUREG-1431, "Standard Technical Specifications Westinghouse Plants."

The proposed changes to SR 4.4.6.2.1.d to perform an RCS water inventory balance only during steady-state operation will result in a more precise verification of RCS leakage and compliance with the requirements of LCO 3.4.6.2. The revised SR will require verification of RCS water inventory balance within 12 hours of achieving steady-state operation and at least once per 72 hours thereafter during steady-state operations. Performance of SR 4.4.6.2.1.d prior to achieving steady-state operation generally does not provide a definitive measurement of compliance with leakage limits. Delaying performance of this SR will not adversely affect the ability to verify compliance with the requirements of the LCO. Therefore, the proposed changes are safe.

The proposed Technical Specification change will remove SR 4.4.6.2.2.c, which is associated with post maintenance testing of RCS PIVs. 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 OPERABLE - OPERABILITY (Millstone Unit No. 3 Technical Specification 1.19) and as such does not need to be restated separately in the SR 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 SRs prior to considering the component operable. The Technical Specification SRs 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 proposed change to the Technical Specifications will not adversely affect the availability or operation of the equipment used to mitigate the design basis accidents. Proper operation of RCS PIVs will still be verified, as appropriate, following maintenance activities. There will be no adverse effect on plant operation. The plant response to the design basis accidents will not change. Therefore, there will be no adverse impact on public health and safety. Thus, the proposed change is safe.

The proposed change to add an identifier to the existing footnote for SRs 4.4.6.2.1 and 4.4.6.2.2 is not a technical change. The proposed change will not create, change, or delete any actions related to compliance with the associated specification. Therefore, the proposed change is safe.

The proposed change to add a new footnote for SR 4.4.6.2.2 clarifies that verification of RCS leakage for PIVs in the RHR flow path is not required when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. The RHR System is the primary means for removing heat from the reactor core in the shutdown cooling mode of operation. The proposed change will ensure that isolation of the RHR System from the

reactor core during system operation and the resultant loss of heat removal does not occur. Verification of RCS PIV leakage during the shutdown cooling mode of operation could result in an RCS PIV failing in the closed position, resulting in a loss of reactor core heat removal. Entry into Modes 3 and 4 is currently allowed to establish the necessary differential pressure and stable conditions to allow for performance of this surveillance. SR 4.4.6.2.2 will still require that leakage limits be verified for those RCS PIVs in the RHR flow path upon securing RHR flow during plant startup. Therefore, the proposed change is safe.

Conclusion

The proposed changes have no effect on how any of the associated systems or components function to mitigate the consequences of design basis accidents. In addition, the proposed changes will not result in any significant change in, or new approach to, plant operation. The proposed changes will not adversely affect public safety. Therefore, the proposed changes are safe.

Attachment 2

Millstone Power Station, Unit No. 3

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage
Significant Hazards Consideration

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage
Significant Hazards Consideration

Description of License Amendment Request

Dominion Nuclear Connecticut, Inc. (DNC) hereby proposes to revise the Millstone Unit No. 3 Technical Specifications as described in this License Amendment Request. The proposed changes are associated with Technical Specification 3.4.6.2, "Reactor Coolant System - Operational Leakage." A brief summary of the changes is provided below. Refer to Attachment 1 of this submittal for a detailed discussion of the proposed changes.

Technical Specification Changes

- A footnote will be added to Limiting Condition for Operator (LCO) 3.4.6.2.f such that this LCO does not apply to those valves in the Residual Heat Removal (RHR) System flow path when in, or during the transition to or from, the RHR mode of operation.

- Surveillance Requirements (SRs) 4.4.6.2.1.a and 4.4.6.2.1.b will be removed, and the word "DELETED" will be added for each.

- SR 4.4.6.2.1.d will be revised such that verification of Reactor Coolant System (RCS) water inventory balance is only performed during steady-state operation.

- SR 4.4.6.2.2.c will be removed and the word "DELETED" will be added. This SR requires that prior to returning an RCS Pressure Isolation Valve (PIV) to service following maintenance, repair or replacement work on a valve the operability of the valve will be verified by ensuring that valve leakage is within its limit is not necessary.

- An identifier will be added to the existing footnote for SRs 4.4.6.2.1 and 4.4.6.2.2 to minimize the potential for confusion with respect to the applicability of this footnote due to the addition of another footnote to this Technical Specification.

- An new footnote will be added to SR 4.4.6.2.2 which clarifies that verification of RCS leakage for PIVs in the RHR flow path is not required when the RHR System is aligned to the RCS in the shutdown cooling mode of operation.

Basis for No 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 do not:

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

   The proposed changes to Technical Specification 3.4.6.2 for RCS PIVs in the RHR flow path will not cause an accident to occur and will not result in any change in the operation of associated accident mitigation equipment. The ability of the RHR System to remove core decay heat will not be affected. The proposed changes will not affect the ability of the RCS or the RHR System to mitigate any design basis event. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 Final Safety Analysis Report (FSAR), and the consequences of the design basis accidents will remain the same. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

   The proposed changes to delete SRs 4.4.6.2.1.a and 4.4.6.2.1.b and revise SR 4.4.6.2.1.d will not cause an accident to occur and will not result in any change in the operation of associated accident mitigation equipment. The ability to measure RCS operational leakage will not be affected. The proposed changes will not affect the ability to mitigate any design basis event. The design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 FSAR, and the consequences of the design basis accidents will remain the same. Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

   The proposed change to remove SR 4.4.6.2.2.c to perform post maintenance testing of the RCS PIVs will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The proposed change will not revise the operability requirements (e.g., valve leakage limits) for the RCS PIVs. Proper operation of the RCS PIVs will still be verified, as appropriate, following maintenance activities. As a result, the design basis accidents will remain the same postulated events described in the Millstone Unit No. 3 FSAR, and the consequences of the design basis accidents will remain the same. Therefore, the proposed change 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 do not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. The proposed changes do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce the margin of safety since they have no impact on any accident analysis assumption. The proposed changes do not decrease the scope of equipment currently required to be operable or subject to surveillance testing, nor do the proposed changes affect any instrument setpoints or equipment safety functions. The effectiveness of Technical Specifications will be maintained since the changes will not alter the operation of any component or system, nor will the proposed changes affect any safety limits or safety system settings. Therefore, there is no reduction in a margin of safety.

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Attachment 3

Millstone Power Station, Unit No. 3

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage
Marked Up Pages

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage

List of Marked Up Pages

Technical Specification Section Number	Title of Section	Affected Page with Amendment Number
3.4.6.2	Reactor Coolant System - Operational Leakage	3/4 4-22, Original Issue 3/4 4-23, Amendment No. 174 B 3/4 4-5, Original Issue

 For Information Only 2/19/97

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. Either the Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. The Containment Drain Sump Level or Pumped Capacity Monitoring System

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With both the Containment Atmosphere Gaseous and Particulate Radioactivity Monitors INOPERABLE, operation may continue for up to 30 days provided the Containment Drain Sump Level or Pumped Capacity Monitoring System is OPERABLE and gaseous grab samples of the containment atmosphere are obtained at least once per 12 hours and analyzed for gross noble gas activity within the subsequent 2 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the Containment Drain Sump Level or Pumped Capacity Monitoring System INOPERABLE, operation may continue for up to 30 days provided either the Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System is OPERABLE; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the followng 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Drain Sump Level and Pumped Capacity Monitoring System-performance of CHANNEL CALIBRATION at least once each REFUELING INTERVAL.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2250 ± 20 psia, and

f.* 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*This requirement does not apply to Pressure Isolation Valves in the Residual Heat Removal Flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 (1) Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours; Deleted

b. Monitoring the containment drain sump inventory and discharge at least once per 12 hours; Deleted

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 $\pm$ 20 psia at least once per 31 days with the modulating valve fully open. ~~The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;~~

d. Performance of a Reactor Coolant System water inventory balance within 12 hours of achieving steady state operation and at least once per 72 hours thereafter during steady-state operation; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 (1)(2) Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once each REFUELING INTERVAL,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

(1) The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR flow path when in, or during the transition to or from the shutdown cooling mode of operation.

For Information Only

JAN 31 1986

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
3-SIL-V15	SI Tank 1A Discharge Isolation Valve
3-SIL-V17	SI Tank 1B Discharge Isolation Valve
3-SIL-V19	SI Tank 1C Discharge Isolation Valve
3-SIL-V21	SI Tank 1D Discharge Isolation Valve
3-SIL-V26	RHR/SI to RCS Loop 2, Hot Leg
3-SIL-V27	SIH to RCS Loop 2, Hot Leg
3-SIL-V28	RHR/SI to RCS Loop 4, Hot Leg
3-SIL-V29	SIH to RCS Loop 4, Hot Leg
3-SIL-V984	RHR/SI to RCS Loop 4, Cold Leg
3-SIL-V985	RHR/SI to RCS Loop 3, Cold Leg
3-SIL-V986	RHR/SI to RCS Loop 2, Cold Leg
3-SIL-V987	RHR/SI to RCS Loop 1, Cold Leg
3-SIH-V5	SIH to RCS Cold Legs
3-SIH-V110	SIH to RCS Loop 1, Hot Leg
3-SIH-V112	SIH to RCS Loop 3, Hot Leg
3-RCS-V26	SIH to RCS Loop 1, Hot Leg
3-RCS-V29	SIH to RCS Loop 1, Cold Leg
3-RCS-V30	SIL to RCS Loop 1, Cold Leg
3-RCS-V69	RHR/SI to RCS Loop 2, Hot Leg
3-RCS-V70	SIH to RCS Loop 2, Cold Leg
3-RCS-V71	SIL to RCS Loop 2, Cold Leg
3-RCS-V102	SIH to RCS Loop 3, Hot Leg
3-RCS-V106	SIH to RCS Loop 3, Cold Leg
3-RCS-V107	SIL to RCS Loop 3, Cold Leg
3-RCS-V142	RHR/SI to RCS Loop 4, Hot Leg
3-RCS-V145	SIH to RCS Loop 4, Cold Leg
3-RCS-V146	SIL to RCS Loop 4, Cold Leg
3-RHS-MV8701C	RCS Loop 1, Hot Leg to RHR
3-RHS-MV8702C	RCS Loop 4, Hot Leg to RHR
3-RHS-MV8701A	RCS Loop 1, Hot Leg to RHR
3-RHS-MV8702B	RCS Loop 4, Hot Leg to RHR

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2250 psia. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

BASES

OPERATIONAL LEAKAGE (Continued)

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested ~~periodically~~ to ensure low probability of gross failure.

Insert A

~~The~~ Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Insert B

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values

INSERT 'A' TO PAGE B 3/4 4-5

Steady-state operation is defined as stable Reactor Coolant System temperature, reactor power level, pressurizer and makeup tank levels, and makeup and letdown flows. The 12 hour allowance time for Surveillance Requirement 4.4.6.2.1.d provides sufficient time to collect and process all necessary data after stable plant conditions are established.

INSERT 'B' TO PAGE B 3/4 4-5

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions for performance of Surveillance Requirement 4.4.6.2.2 for Reactor Coolant System (RCS) pressure isolation valves which can only be leak-tested at elevated RCS pressures. The requirements of Surveillance Requirement 4.4.6.2.2 to verify that a pressure isolation valve is OPERABLE shall be performed within 24 hours after the required RCS pressure has been met.

Attachment 4

Millstone Power Station, Unit No. 3

Technical Specifications Change Request 3-11-01
Reactor Coolant System Operational Leakage
Retyped Pages

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2250 ± 20 psia, and

f.* 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*This requirement does not apply to Pressure Isolation Valves in the Residual Heat Removal Flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1[1] Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Deleted

b. Deleted

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2250 $\pm$ 20 psia at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance within 12 hours of achieving steady state operation and at least once per 72 hours thereafter during steady state operation; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2[1][2] Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once each REFUELING INTERVAL,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Deleted

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve, and

e. When tested pursuant to Specification 4.0.5.

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(1) The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

(2) This surveillance is not required to be performed on Reactor Coolant System Pressure Isolation Valves located in the RHR Flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

OPERATIONAL LEAKAGE (Continued)

The specified allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

Steady-state operation is defined as stable Reactor Coolant System temperature, reactor power level, pressurizer and makeup tank levels, and makeup and letdown flows. The 12 hour allowance time for Surveillance Requirement 4.4.6.2.1.d provides sufficient time to collect and process all necessary data after stable plant conditions are established.

The Surveillance Requirements for RCS pressure isolation valves provide assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions for performance of Surveillance Requirement 4.4.6.2.2 for Reactor Coolant System (RCS) pressure isolation valves which can only be leak-tested at elevated RCS pressures. The requirements of Surveillance Requirement 4.4.6.2.2 to verify that a pressure isolation valve is OPERABLE shall be performed within 24 hours after the required RCS pressure has been met.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values