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

Dominion

JUN 28 2001

Docket No. 50-423
B18422

RE: 10 CFR 50.90

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

Millstone Nuclear Power Station, Unit No. 3
Technical Specifications Change Request 3-6-01
Containment Isolation Valves

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed change into the Technical Specifications of Millstone Unit No. 3. DNC is proposing to change Technical Specification 3.6.3, "Containment Systems - Containment Isolation Valves."

The proposed change will remove the Technical Specification surveillance requirement associated with post maintenance testing of the containment isolation valves. 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 test, based on the work performed.

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

Environmental Considerations

DNC has evaluated the proposed change 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 change meets 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

A001

determination is based on the fact that the change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a surveillance requirement, and that the amendment request meets the following specific criteria.

(i) The proposed change involves no significant hazards consideration.

As demonstrated in Attachment 2, the proposed change does 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 change will remove a surveillance requirement associated with post maintenance testing. However, this testing will continue to be performed, as required, to ensure component operability following maintenance activities. The proposed change is consistent with the design basis of the plant. The proposed change 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 change 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 change will not result in changes in the configuration of the facility. The proposed change will remove a surveillance requirement associated with post maintenance testing. However, this testing will continue to be performed, as required, to ensure component operability following maintenance activities. The manner in which the maintenance is performed will not be affected by the proposed change. 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 change.

Conclusions

The proposed change does not involve a significant impact on public health and safety (see the Safety Summary provided in Attachment 1) and does not involve a Significant Hazards Consideration pursuant to the provisions of 10 CFR 50.92 (see the Significant Hazards Consideration provided in Attachment 2). In addition, we have concluded the proposed change is safe.

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 December 31, 2001, with the amendment to be implemented within 30 days of issuance.

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 28th day of June, 2001

Notary Public

My Commission expires

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

cc: See next page

Attachments (4)

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

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
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Docket No. 50-423
B18422

Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-6-01
Containment Isolation Valves
Discussion of Proposed Change and Safety Summary

Technical Specifications Change Request 3-6-01
Containment Isolation Valves
Discussion of Proposed Change and Safety Summary

Introduction

Dominion Nuclear Connecticut, Inc. (DNC), hereby proposes to amend Operating License NPF-49 by incorporating the attached proposed change into the Technical Specifications of Millstone Unit No. 3. DNC is proposing to change Technical Specification 3.6.3, "Containment Systems - Containment Isolation Valves."

The proposed change will remove the Technical Specification surveillance requirement associated with post maintenance testing of the containment isolation valves. 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 test, based on the work performed.

Technical Specification Change

Surveillance Requirement (SR) 4.6.3.1 will be removed, and the word "DELETED" will be added. This SR, which requires containment isolation valve operability to be verified by performing a timed valve stroke prior to returning the valve to service following any maintenance, repair, or replacement 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 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.

Safety Summary

The proposed Technical Specification change will remove SR 4.6.3.1, 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 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 the containment isolation valves 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.

Attachment 2

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-6-01
Containment Isolation Valves
Significant Hazards Consideration

Technical Specifications Change Request 3-6-01
Containment Isolation Valves
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. DNC is proposing to change Technical Specification 3.6.3, "Containment Systems - Containment Isolation Valves," by removing the surveillance requirement associated with post maintenance testing. This testing will continue to be performed, as required, to ensure component operability following maintenance activities. Refer to Attachment 1 of this submittal for a detailed discussion of the proposed change.

Significant Hazards Consideration

In accordance with 10 CFR 50.92, DNC has reviewed the proposed change and has concluded that it does 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 change does not involve an SHC because the change would not:

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

   The proposed Technical Specification change to remove the surveillance requirement to perform post maintenance testing of the containment isolation valves will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The containment isolation valves are not accident initiators. The proposed change will not revise the operability requirements (e.g., valve stroke time) for the containment isolation valves. Proper operation of the containment isolation valves 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 Final Safety Analysis Report, 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 change to the Technical Specifications does not impact any system or component that could cause an accident. The proposed change will not alter the plant configuration (no new or different type of equipment will be installed) or require any unusual operator actions. The proposed change will not alter the way any structure, system, or component functions, and will not significantly alter the manner in which the plant is operated. The response of the plant and the operators following an accident will not be different. In addition, the proposed change does not introduce any new failure modes. Therefore, the proposed change 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 change to remove the surveillance requirement to perform post maintenance testing of the containment isolation valves will not cause an accident to occur and will not result in any change in the operation of the associated accident mitigation equipment. The operability requirements for the containment isolation valves have not been changed, and proper operation of the containment isolation valves will still be verified, as appropriate, following maintenance activities. The containment isolation valves will continue to be able to mitigate the design basis accidents as assumed in the safety analysis. In addition, the proposed change will not adversely affect equipment design or operation, and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. Therefore, the proposed change will not result in a reduction in a margin of safety.

Docket No. 50-423
B18422

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-6-01
Containment Isolation Valves
Marked Up Pages

Millstone Nuclear Power Station, Unit No. 3
Technical Specifications Change Request 3-6-01
Containment Isolation Valves
Marked Up Pages

A change to the following Technical Specification page has been proposed.

Technical Specification Section Number(s)	Title(s) of Section(s)	Page and Revision Numbers
3/4.6.3	Containment Systems Containment Isolation Valves	3/4 6-15 Amend. 136

CONTAINMENT INTEGRITY

NO CHANGE
FOR INFORMATION
ONLY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or operator action during periods when containment isolation valves may be opened under administrative control per Specification 4.6.1.1a.

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of the Containment Leakage Rate Testing Program, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample.

NO CHANGE
FOR INFORMATION ONLY

November 2, 2000

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves or operator action during periods when containment isolation valves are opened under administrative control,** and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions; and

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. Deleted

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

** The following valves may be opened on an intermittent basis under administrative control. Manual valves 3SSP*V13, 3SSP*V14, 3HCS*V2, 3HCS*V3, 3HCS*V9, 3HCS*V10, 3HCS*V6, 3HCS*V13, 3CHS*V371, 3MSS*V885, 3MSS*V886, 3MSS*V887. Remote manual valves 3RHS*MV8701A, 3RHS*MV8701B, 3RHS*MV8702A, 3RHS*MV8702B.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE* with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

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

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

DELETED

4.6.3.1 ~~Each isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.~~

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once each REFUELING INTERVAL by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position, and

c. Verifying that on a Containment High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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*The provisions of this Specification are not applicable for main steam line isolation valves. However, provisions of Specification 3.7.1.5 are applicable for main steam line isolation valves.

November 2, 2000
NO CHANGE
FOR INFORMATION ONLY

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

When the Residual Heat Removal (RHR) System is placed in service in the plant cooldown mode of operation, the RHR suction isolation remotely operated valves 3RHS*MV8701A and 3RHS*MV8701B, and/or 3RHS*MV8702A and 3RHS*MV8702B are opened. These valves are normally operated from the control room. They do not receive an automatic containment isolation closure signal, but are interlocked to prevent their opening if Reactor Coolant System (RCS) pressure is greater than approximately 412.5 psia. When any of these valves are opened, either one of the two required licensed (Reactor Operator) control room operators can be credited as the operator required for administrative control. It is not necessary to use a separate dedicated operator.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates, as specified in the Containment Leakage Rate Testing Program, ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The Limiting Condition for Operation defines the limitations on containment leakage. The leakage rates are verified by surveillance testing, as specified in the Containment Leakage Rate Testing Program, in accordance with the requirements of Appendix J. Although the LCO specifies the leakage rates at accident pressure, P_a, it is not feasible to perform a test at such an exact value for pressure. Consequently, the surveillance testing is performed at a pressure greater than or equal to P_a to account for test instrument uncertainties and stabilization changes. This conservative test pressure ensures that the measured leakage rates

NO CHANGE FOR INFORMATION ONLY

3/4.6.3 CONTAINMENT ISOLATION VALVES

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 and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for these isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. FSAR Table 6.2-65 lists all containment isolation valves. The addition of deletion of any containment isolation valve shall be made in accordance with Section 50.59 of 10CFR50 and approved by the committee(s) as described in the NUQAP Topical Report.

3/4.6.4 COMBUSTIBLE GAS CONTROL

Hydrogen Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. Containment hydrogen concentration is also important in verifying the adequacy of mitigating actions. The requirement to perform a hydrogen sensor calibration at least every 92 days is based upon vendor recommendations to maintain sensor calibration. This calibration consists of a two point calibration, utilizing gas containing approximately one percent hydrogen gas for one of the calibration points, and gas containing approximately four percent hydrogen gas for the other calibration point.

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit or the Mechanical Vacuum Pumps are capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Post-LOCA performance of the hydrogen recombiner blowers is based on a series of equations supplied by the blower manufacturer. These equations are also the basis of the acceptance criteria used in the surveillance procedure. The required performance was based on starting containment conditions before the LOCA of 10.59 psia (total pressure), 120°F and 100% relative humidiy.

The surveillance procedure shall use the following methods to verify acceptable blower flow rate:

Attachment 4

Millstone Nuclear Power Station, Unit No. 3

Technical Specifications Change Request 3-6-01
Containment Isolation Valves
Retyped Page

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE* with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

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

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 DELETED

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once each REFUELING INTERVAL by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position, and

c. Verifying that on a Containment High Radiation test signal, each purge supply and exhaust isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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*The provisions of this Specification are not applicable for main steam line isolation valves. However, provisions of Specification 3.7.1.5 are applicable for main steam line isolation valves.