UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 3, 2010

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – ISSUANCE OF RELIEF REQUEST RR-04-02 REGARDING ALTERNATIVE VT-2 PRESSURE TESTING REQUIREMENTS FOR THE LOWER PORTION OF THE REACTOR PRESSURE VESSEL (TAC NO. ME3691)

Dear Mr. Heacock:

By letter dated March 30, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100900200), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted relief requests RR-04-02 and RR-04-03 regarding certain evaluation and testing requirements required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Specifically, relief request RR-04-02 requests relief from ASME Code-required VT-2 visual examination of pressure-retaining surfaces for the reactor vessel bottom head at normal operating pressure for Millstone Power Station, Unit No. 2 (MPS2). The Nuclear Regulatory Commission (NRC) staff's review of relief request RR-04-02 is contained in the enclosed safety evaluation. DNC withdrew relief request RR-04-03 by letter dated November 18, 2010 (ADAMS Accession No. ML103230045).

The NRC staff has reviewed relief request RR-04-02 and concludes, as set forth in the enclosed safety evaluation, that the proposed VT-2 visual examinations of the lower portion of the reactor pressure vessel at MPS2 provides assurance of structural integrity and that performing the VT-2 visual examination in accordance with the ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

Therefore, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.55a(a)(3)(ii), the NRC authorizes the use of RR-04-02, alternative VT-2 visual examinations requirements of the lower portion of the reactor pressure vessel for the fourth 10-year interval at MPS2. The fourth 10-year ISI interval began on April 1, 2010.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the authorized Nuclear Inservice Inspector.

If you have any questions, please contact the Project Manager, Carleen Sanders, at 301-415-1603.

Sincerely,

FOR HKC

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosure:
As stated

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELEIF REQUEST RR-04-02

INSERVICE INSPECTION SYSTEM LEAKAGE TEST OF THE

REACTOR PRESSURE VESSEL

MILLSTONE POWER STATION, UNIT NO. 2

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated March 30, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100900200), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted relief requests RR-04-02 and RR-04-03 regarding certain evaluation and testing requirements required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Specifically, relief request RR-04-02 requests relief from ASME Code-required VT-2 visual examination of pressure-retaining surfaces for the reactor vessel bottom head at normal operating pressure for Millstone Power Station, Unit No. 2 (MPS2). The Nuclear Regulatory Commission staff's review of relief request RR-04-02 is contained in this safety evaluation. DNC withdrew relief request RR-04-03 by letter dated November 18, 2010 (ADAMS Accession No. ML103230045).

2.0 REGULATORY REQUIRMENTS

The inservice inspection (ISI) interval of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety; or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require

Enclosure

that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ASME Code of Record for the MPS2 fourth 10-year ISI interval is the 2004 Edition with no Addenda. The Fourth 10-year interval began on April 1, 2010.

2.0 TECHNICAL EVALUATION

2.1 Code Requirements

The 2004 Edition of the ASME Code, Section XI, Table-2500-1 states the following for Examination Category B-P, "All Pressure Retaining Components"

Examination Category B-P, All Pressure Retaining Components							
					Extent and Frequency of Examination		
Item. No.	Parts Examined	Test Requirements	Examination Method [Note (1)]	Acceptance Standard	First Inspection Interval	Successive Inspection Intervals	Deferral of Examination to End of Interval
B.15.10	Pressure Retaining Components	System Leakage Test (IWB-5220)	Visual, VT-2	IWB-3522	Each refueling outage [Note (2)]	Same as the first interval	Not permissible

NOTES:

(1) Visual Examination of IWA-5240.
(2) The system leakage test (IWB-5220) shall be conducted prior to plant startup following a reactor refueling outage.

The 2004 Edition of ASME Code Section XI requires the system leakage test be conducted in accordance with the following guidelines:

IWB-5220 System Leakage Test

IWB-5221 Pressure

(a) The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power.
(b) The system test pressure and temperature shall be attained at a rate in accordance with the heat-up limitations specified for the system.

IWB-5222 Boundaries

(a) The pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity.

(b) The pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval shall extend to all Class 1 pressure retaining components within the system boundary.

2.2 ASME Code Components Affected

Component: Pressure-Retaining surfaces of the lower portion of the Reactor Pressure Vessel

Code Class: 1

2.3 Licensee Technical Evaluation

Licensee's Proposed Alternative

The licensee proposes to conduct the VT-2 visual examination of the pressure retaining surfaces of the reactor vessel bottom head following plant cooldown during each refuel outage. With substantially lower reactor coolant system (RCS) temperatures, the under vessel area will also be at a lower temperature and therefore less hazardous to personnel. The objective of the required VT-2 visual examination at normal operating pressure and normal operating temperature (NOP/NOT) is to detect evidence of leakage and assure the integrity of the RCS pressure boundary. This objective can also be achieved by a VT-2 visual examination performed during the refueling outage following the RCS cooldown. There is no insulation on the reactor vessel bottom head area and, therefore, evidence of any leakage and boric acid corrosion occurring during the fuel cycle would be detected by visual examination of this area at the end of the fuel cycle during the outage. The ability to detect any evidence of leakage and boric acid corrosion in this area during the refueling outage provides reasonable assurance of leak tight integrity of the reactor vessel bottom head area without exposing personnel to the environmental hazards associated with entry into this area during Mode 3 (Hot Standby) when RCS is at NOP/NOT.

Licensee's Basis for Alternative

The performance of the required VT-2 visual examination of the reactor vessel bottom head area during the hydrostatic and system leakage tests at normal operating pressure and temperature can create a hazardous situation for the inspectors. The licensee states that the "...area under the vessel is classified as a confined space with limited air circulation and limited access. With RCS at NOP/NOT, ambient temperatures in this area are very high due to the uninsulated condition of the vessel. The high temperature levels in this area will create a significant safety hazard to personnel entering this space. Additionally, the elevation of the RPV [reactor pressure vessel] in the cubicle is relatively low with about a 2-foot distance between the floor and bottom of the vessel." Consequently, the proximity of the reactor vessel in relation to the floor poses additional hazard for personnel performing VT-2 visual examination in this area who may inadvertently contact the uninsulated vessel surface with a potential for severe burn.

Since the vessel bottom head is uninsulated, any evidence of leakage and boric acid corrosion occurring during the fuel cycle can be detected by visual examination of this area at the end of the cycle during the outage. During the outage, RCS temperatures will be substantially lower under the vessel area, which minimizes the risk of any hazardous situation for the test crew.

The licensee also states that there are no bottom mounted instrumentation nozzles at MPS2. Consequently, degradation in the vessel wall is not expected in the absence of penetrations.

No significant reduction in radiation exposure is expected to result from this request; therefore radiation exposure is not included as a reason for this request.

2.4 NRC Staff Evaluation

The ASME Code requires that VT-2 visual examinations of the bottom of the reactor vessel be conducted during system leakage tests. The ASME Code requires that the VT-2 visual examination be conducted at pressure corresponding to 100% rated reactor power. DNC has proposed, as an alternative, to perform a VT-2 visual examination of the bottom of the reactor vessel following plant cooldown each refueling outage. The objective of the VT-2 visual examination is to detect evidence of leakage and thereby verify the integrity of the RCS pressure boundary. The NRC staff finds that since the vessel bottom is uninsulated, any evidence of leakage and boric acid corrosion that occurred during the previous fuel cycle can be readily detected by visual examination during the refueling outage. Therefore, the NRC staff finds that the objective of the VT-2 visual examination is met.

The licensee states that the high temperature at NOP/NOT creates a significant safety hazard for personnel inspecting the lower portion of the reactor pressure vessel. However, the licensee does not address the ability for NOP to be achieved at a lower temperature in accordance with the pressure/temperature curves in MPS2's technical specifications. It is unclear to the NRC staff if the same significant safety hazard would exist for personnel at the lower temperature permitted by the technical specifications. The NRC staff has determined this information is not necessary because the NRC staff finds that the minimum temperature permissible for NOP conditions is hot enough to create a hardship for the licensee and that the proposed alternative provides reasonable assurance of the structural integrity of the reactor vessel bottom head area. Therefore, based on the above, the NRC staff finds that complying with the required system leakage test would result in a hardship or unusual difficulty without a compensating increase in the level of quality or safety.

MPS2 does not have penetration in the lower head of the RPV, therefore bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," is not applicable. Since bulletin 2003-02 is not applicable to MPS2, this relief request has no impact on MPS2's actions associated with bulletin 2003-02.

3.0 CONCLUSION

Based on the discussions above, the NRC staff concludes that performing VT-2 visual examinations of the lower portion of the reactor pressure vessel following cooldown each refueling outage will meet the intention of the ASME Code and will verify the structural integrity of the RCS pressure boundary. The NRC staff also concludes based on the above discussion, that performing VT-2 visual examinations of the lower portion of the reactor pressure vessel at NOP would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the use of relief request RR 04-02 for the fourth 10-year ISI interval at MPS2.

All other ASME Code, Section XI, requirements for which relief has not been specifically requested and approved remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: P. Patnaik

Date: December 3, 2010

If you have any questions, please contact the Project Manager, Carleen Sanders, at 301-415-1603.

Sincerely,

/ra/ (EMiller for)

Harold K. Chernoff, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

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