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

January 10, 2014

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 - RELIEF REQUEST RR-04-14 FROM THE REQUIREMENTS OF THE ASME SECTION XI INSERVICE INSPECTION PROGRAM FOR REACTOR VESSEL SUPPORTS (TAC NO. MF0682)

Dear Mr. Heacock:

By letter dated February 15, 2013, Dominion Nuclear Connecticut, Inc. (the licensee) submitted a request for relief from the surface and visual examination of the reactor vessel supports required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWB-2500 and IWF-2500 for Millstone Power Station, Unit 2 (MPS2).

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(g)(5)(iii), the licensee requested relief from achieving full volumetric coverage of the reactor vessel supports on the basis that the code requirement is impractical.

The NRC staff determined that there is reasonable assurance of the capability of the reactor vessel supports to perform their design function based on the visual observation, ongoing efforts to detect degradation in the area due to boric acid leakage, and significant margin in the supports to accommodate the remaining required design basis loading. The NRC staff finds that complying with the specified ASME Code requirement would be impractical. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(g)(5)(iii). Therefore, the NRC authorizes the use of Relief Request RR-04-14 at the MPS2 for the duration of the fourth 10-year inservice inspection interval which ends on March 31, 2020.

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

If you have any questions, please contact the Millstone Power Station Project Manager, James Kim, at (301) 415-4125.

Sincerely,

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosure:
Relief Request

cc w/encl: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. RR-04-14

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NOS. 50-336

1.0 INTRODUCTION

By letter dated February 15, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML13056A426), Dominion Nuclear Connecticut, Inc. (the licensee) submitted a request for relief from the surface and visual examination of the reactor vessel supports required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWB-2500 and IWF-2500 for Millstone Power Station, Unit 2 (MPS2). Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 50.55a(g)(5)(iii), the licensee requested relief from achieving full volumetric coverage of the reactor vessel supports on the basis that the code requirement is impractical.

2.0 REGULATORY REQUIREMENTS

The inservice inspection (ISI) Program of the American Society of Mechanical Engineers Boiler and Pressure Vessel (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) 50.55a(g), except where specific relief has been granted by the U.S. Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(g)(6)(i). Regulation 10 CFR 50.55a(g)(5)(iii) requires that if the licensee has determined that conformance with certain ASME Code, Section XI requirements is impractical for its facility, the licensee shall notify the NRC and submit, as specified in 10 CFR 50.4, information to support the determination.

The 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (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 regulation requires that inservice examination of components and system pressure tests conducted during

Enclosure

the first ten-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) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable Code of Record for the fourth 10-year interval ISI program at MPS2 is the 2004 Edition of the ASME Code, Section XI, without Addenda. The fourth 10-year interval ISI program at MPS2 began on April 10, 2010 and is scheduled to end on March 31, 2020.

3.0 TECHNICAL EVALUATION

Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee submitted Relief Request RR-04-14 in order to obtain relief from the applicable inservice examination requirements for MPS2 of reactor vessel supports 2-RVS-1, 2-RVS-2 and 2-RVS-3.

The 2004 Edition of the ASME Code, Section XI, Articles IWB-2500 and IWF-2500 require that components be examined and tested as specified in Tables IWB-2500-1 and IWF-2500-1 of the ASME Code, Section XI. These tables specify, among other things, the examination techniques (volumetric, surface, and/or visual) and the examination boundary. For each of the subject components, the examination boundary is provided in the figure specified for that particular ASME Code, Section XI Item Number in Tables IWB-2500-1 and IWF-2500-1. For volumetric and surface exams, the required examination boundary includes essentially 100% of the volume or area specified in the applicable figure listed in Tables IWB-2500-1 and IWF-2500-1. In Relief Request RR-04-14, the licensee indicated that for the subject components, the actual volumetric and surface examination coverage during the fourth 10-year interval ISI program at MPS2 was completely limited by component configuration, design, geometry, physical interferences and accessibility issues. The licensee attempted to conduct examinations on the subject components to the extent practical, but the examinations were not successful. Provided below are the details concerning the staff's evaluation of Relief Request RR-04-14 and are based on the licensee's explanation regarding the impracticality of achieving the full examination coverage as required by the ASME Code, Section XI for the subject components.

3.1 Component Identification

Relief Request RR-04-14 addresses the following: ASME Code Class 1, Examination Category B-K Item No. B10.10 and Category F-A Item No. F1.40 components at MPS2 for Reactor Vessel Supports 2-RVS1, 2-RVS-2 and 2-RVS-3.

3.2 Applicable ASME Code, Section XI Requirements

The ASME Code, Section XI, Table IWB-2500-1, Examination Category B-K, requires a 100% surface examination of the pressure vessel welded attachments from the accessible side once every 10-year interval.

The ASME Code, Section XI, Table IWF-2500-1, Examination Category F-A requires visual examination (VT-3) of 100% of reactor vessel supports once each 10-year interval.

3.3 Licensee's Basis for Relief

In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the ASME Code, Section XI examination coverage requirements for the Examination Category B-K Item No. B10.10 ("pressure vessel welded attachment") and Category F-A Item Number F1.40 ("supports other than piping supports") for Reactor Vessel Supports 2-RVS-1, 2-RVS-2 and 2-RVS-3 in its submittal dated February 15, 2013 on the basis that the required examination coverage of "essentially 100 percent" is impractical due to the obstructions and limitations resulting from the design configuration, with additional consideration of the associated accumulated dose that would be incurred by altering the current configuration and geometry.

The licensee stated that, "in early 2012, MPS performed a review of industry operating experience (OE) and found instances where reactor vessel supports were not correctly identified for examination in the ISI Program. As a follow-up to this industry OE, a review of the MPS ISI Program was performed to determine if the reactor vessel supports were appropriately identified for examination as require by ASME Section XI. The review determined that these vessel supports were not included in the MPS2 ISI Program. The review was expanded to include the previous 10-year intervals of the MPS2 ISI Program. No evidence was found that these supports were ever included in the ISI Program. It is concluded these supports have never received the Code required examinations. This condition was entered in the MPS Corrective Action Program."

An inspection of the MPS2 reactor vessel supports during the fall of 2012 was not successful. Attempts were made to lower a camera through each of the six refuel cavity seal ring manways, but the location of supports in relation to the manways prevented adequate access for performing the examinations.

The licensee evaluated other potential entry points. Access through the piping penetration opening of the bio shield wall was determined to be inaccessible due to inadequate clearance between the insulated piping and the wall opening. In addition, due to the configuration of the insulation on the piping, removal could only be performed from the reactor vessel side of the bio shield wall. The permanently welded cavity seal ring would have to be removed in order to gain access to remove the insulation. The licensee concluded that in order to access the reactor vessel supports to perform the Code required examinations, extensive maintenance activities for removal of the permanently welded refuel cavity seal ring and removal of the insulation panels that surround the supports would be needed. The licensee determined that examination from below was not possible due to space limitations between the reactor vessel and the bio shield wall, such that the necessary scaffolding could not be constructed.

Removal of the cavity ring seal would require the removal of two welds. The licensee provided estimates of personnel doses that would be received if removal of the welds and refuel cavity seal ring were pursued. These estimates were as follows:

- Removal and reinstallation of refuel cavity seal ring - 20.6 rem [roentgen equivalent man]
- Removal and reinstallation of insulation panels - 74.4 rem
- Paint removal and surface preparation - 12 rem
- Perform examinations - 8 rem
- Health Physics support - 1 rem

The MPS2 reactor vessel supports were visually inspected (non-ASME Code) in 1988 following the cleanup of a boric acid leak at the reactor vessel flange. The permanent refuel cavity seal ring was installed in 2005. The visual inspection found no degradation of the reactor vessel supports. Since 1988, there has been no evidence of additional boric acid leakage that could affect the reactor vessel supports. Under vessel visual inspections for leakage have been performed during each refueling outage since 2003. These inspections have confirmed no boric acid leakage in the area.

The MPS2 reactor vessel supports were designed with significant margins. The reactor vessel support design included consideration of normal operating loads plus pipe rupture loads plus the maximum hypothetical seismic loads. MPS2 Final Safety Analysis Report, Section 3.A.5 discusses the Leak-Before-Break (LBB) analyses for the reactor coolant system (RCS) main coolant loops, the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping. These analyses, previously reviewed and approved by NRC staff, demonstrated that the probability of fluid system piping rupture was extremely low. Therefore, the LBB analysis eliminated the pipe rupture cases and significantly reduced the load applied to the reactor vessel supports. Seismic stresses and the percentage of allowable limits for the reactor vessel supports, specifically 16.8% for the maximum primary membrane and 38.9% for the membrane plus bending loading condition, provide additional margin. The licensee analysis concludes that "significant margin exists in the reactor vessel supports to accommodate the remaining required design basis loading, even considering possible support degradation (though considered to be unlikely). Based on the simple design, previous inspection results and significant margin between the support design and the required support load carrying capability, the reactor vessel supports are considered to be functional for the required design basis loading."

3.4 NRC Staff Evaluation

The NRC staff reviewed the information provided by the licensee concerning the impracticality of achieving full volumetric coverage of the subject reactor vessel supports, as discussed above. In addition to technical descriptions, the licensee provided drawings and a photo demonstrating the limited accessibility for examination of the reactor vessel supports. The staff concluded that current inspection technology, the geometric configurations of the reactor vessel supports and physical obstructions do not allow coverage of the ASME Code inspection volume. In order to provide access to effectively provide examination coverage of the reactor vessel supports, removal of the welds that attach the inner side of the cavity ring seal to the reactor vessel flange and the outer side of the ring to the refuel cavity door – thus allowing removal of the cavity seal ring - would be necessary. Additionally, performance of this removal activity involves an estimated accumulated dose of 116 rem and introduces an opportunity to damage the cavity seal ring and possibly require further repair or replacement. Therefore, examining essentially 100% of the ASME Code-required volumes is considered impractical.

In 1988, following the cleanup of a boric acid leak at the reactor vessel flange, the reactor vessel supports were visually inspected and no support degradation was identified. However, this was not an ASME Code inspection. A permanent refuel cavity seal ring was installed in 2005. Under vessel inspections have been conducted at each refueling outage since 2003 and have confirmed no boric acid leakage in the area. Additionally, the staff concluded that the reactor vessel supports, which normally have stress loads in compression, have significant margin to accommodate the remaining required design basis loading. Due to the earlier visual

observation, ongoing efforts to detect degradation in the area due to boric acid leakage, and significant margin in the supports to accommodate the remaining required design basis loading, the staff concludes that there is reasonable assurance of the capability of the reactor vessel supports to perform their design function.

4.0 CONCLUSION

Based on the above evaluation of MPS2 Relief Request RR-04-14, the staff concludes that the applicable ASME Code, Section XI examination requirements are impractical for the subject Class 1 components: reactor vessel supports 2-RVS-1, 2-RVS-2 and 2-RVS-3. Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), Relief Request RR-04-14 is authorized for the surface and visual examinations required by the ASME Code, Section XI, IWB-2500 and IWF-2500, through the end of the fourth 10-year ISI interval.

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

Principal Contributor: C. Fairbanks

Date: January 10, 2014

If you have any questions, please contact the Millstone Power Station Project Manager, James Kim, at (301) 415-4125.

Sincerely,

/ra/

Benjamin G. Beasley, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-336

Enclosure:
Relief Request

cc w/encl: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL1-1 Reading File
RidsNrrDorlLpl1-1
RidsNrrPMMillstone
RidsNrrLAKGoldstein
RidsAcrsAcnw_MailCTR
RidsNrrDorlDpr
RidsRgn1MailCenter
RidsNrrDeEvib
CFairbanks, NRR

ADAMS Accession No.: ML14007A687 *See memo date December 24, 2013

OFFICE	LPL1-1/PM	LPL1-1/LA	EVIB/BC	LPL1-1/BC
NAME	JKim	KGoldstein	SRosenberg*	BBeasley
DATE	01/09/14	01/09/14	12/24/13	01/10/14

OFFICIAL RECORD COPY