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

April 16, 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. 3 – ISSUANCE OF RELIEF REQUEST IR-3-10 REGARDING USE OF AMERICAN SOCIETY OF MECHANICAL ENGINEERING CODE, SECTION XI, APPENDIX VIII (TAC NO. ME1262)

Dear Mr. Heacock:

By letter dated April 28, 2009 (Agencywide Documents Access and Management System Accession Nos. ML091310666), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted relief requests for the third 10-year inservice inspection (ISI) interval program at Millstone Power Station, Unit No. 3 (MPS3). DNC requested the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements. Included in this request was Relief Request IR-3-10. By letter dated February 3, 2010 (ADAMS Accession No. ML100361055), DNC supplemented relief request IR-3-10. DNC submitted Revision 1 to Relief Request IR-3-10 in the February 3, 2010, letter. Relief Request IR-3-10, Revision 1, superseded IR-3-10 in its entirety. Relief Request IR-3-10, Revision 1, proposed an alternative to the frequency of the visual examination requirements of the reactor vessel hot leg nozzle-to-safe end welds in the reactor coolant system. The other relief requests contained in the April 28, 2009, request are being reviewed separately.

The Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that an alternate frequency of the visual examination requirements of the reactor vessel hot leg nozzle-to-safe end welds in the reactor coolant system provides an acceptable level of quality and safety.

Therefore, pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.55a(a)(3)(i), the NRC authorizes the use of an alternate frequency of the visual examination requirements of the reactor vessel hot leg nozzle-to-safe end welds in the reactor coolant system for the remainder of the third 10-year ISI interval for MPS3. The third 10-year ISI interval began on April 23, 2009, and is scheduled to be completed on April 22, 2019.

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,

Harold Chernoff, Chief
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosure:
As stated

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

REQUEST FOR RELIEF NO. IR-3-10

MILLSTONE POWER STATION, UNIT NO. 3

DOMINON NUCLEAR CONNECTICUT, INC.

DOCKET NUMBER 50-423

## 1.0 INTRODUCTION

By letter dated April 28, 2009 (Agencywide Documents Access and Management System Accession Nos. ML091310666), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted relief requests for the third 10-year inservice inspection (ISI) interval program at Millstone Power Station, Unit No. 3 (MPS3). DNC requested the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI requirements. Included in this request was Relief Request IR-3-10. By letter dated February 3, 2010 (ADAMS Accession No. ML100361055), DNC supplemented relief request IR-3-10. DNC submitted Revision 1 to Relief Request IR-3-10 in the February 3, 2010, letter. Relief Request IR-3-10, Revision 1, superseded IR-3-10 in its entirety. Relief Request IR-3-10, Revision 1, proposed an alternative to the frequency of the visual examination requirements of the reactor vessel hot leg nozzle-to-safe end welds in the reactor coolant system. The relief is requested pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.55a(a)(3)(i).

The third 10-year ISI interval at MPS3 began on April 23, 2009, and is scheduled to end on April 22, 2019.

## 2.0 REGULATORY REQUIREMENTS

The ISI 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 10 CFR 50.55a(g), except where specific relief has been granted by the Nuclear Regulatory Commission (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:

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(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 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 MPS3 third 10-year ISI interval is the 2004 Edition with no Addenda of Section XI of the ASME Code. In addition, as required by 10 CFR 50.55a(b)(2)(xv), ASME Code, Section XI, 2001 Edition with no Addenda is used for Appendix VIII, Performance Demonstration for Ultrasonic Examination Systems.

The regulatory requirements in 10 CFR 50.55a(g)(6)(ii)(E) define the requirements for reactor coolant pressure boundary bare metal visual inspections, in which, all licensees of pressurized water reactors (PWR) plants shall augment their ISI program by implementing ASME Code Case N–722 subject to the conditions specified in paragraphs 10 CFR 50.55a(g)(6)(ii)(E)(2) through (4).

## 3.0 TECHNICAL EVALUATION

### 3.1 Applicable Code Edition and Addenda

The 2004 Edition and no Addenda of the ASME Code, Section XI, is the Code of Record for the third 10-year ISI program at MPS3. In addition, volumetric examinations are to be conducted in accordance with the 2001 Edition with no Addenda of the ASME Code, Section XI, Appendix VIII, Supplement 10.

### 3.2 Components for Which Relief is Requested

| | |
|---|---|
| Component: | Reactor Vessel (RV) Hot Leg Nozzle-to-Safe End Dissimilar Metal (DM) Butt Welds |
| Code Class: | Class 1 |
| Examination Category: | PWR Components Containing Alloy 600/82/182 in Table 1 of ASME Code Case N-722 |
| System: | Reactor Coolant System |
| Code Item No.: | B15.90 in Table 1 of ASME Code Case N-722 |

The four RV hot leg nozzle-to-safe end DM butt welds at MPS3 that relief is requested are listed below.

| Component No. | Outside Diameter (OD) (inch) | Inside Diameter (ID) (inch) | Wall Thickness (inch) |
|---|---|---|---|
| 302-121-A | $34\ ^{7}/_{32}$ | $28\ ^{31}/_{32}$ | $2\ ^{20}/_{32}$ |
| 302-121-B | | | |
| 302-121-C | | | |
| 302-121-D | | | |

The susceptibility of nickel-based alloys to the primary water stress-corrosion cracking (PWSCC) in PWR plants is a safety concern. The susceptible materials in the subject welds include filler materials Alloy 82/182, which are utilized to weld the stainless steel safe ends to the RV nozzles.

3.3 Applicable Code Requirements

The regulatory requirements in 10 CFR 50.55a(g)(6)(ii)(E) define the requirements for reactor coolant pressure boundary bare metal visual inspections, in which, all licensees of PWR plants shall augment their ISI program by implementing ASME Code Case N–722 subject to the conditions specified in paragraphs 10 CFR 50.55a(g)(6)(ii)(E)(2) through (4). ASME Code Case N-722 requires that, in addition to the examination requirements of Table IWB-2500-1 of the ASME Code, Section XI, the examination requirements of Table 1 of Code Case N-722 shall be performed for PWR plants having partial or full penetration welds in Class 1 components containing Alloy 600/82/182 material.

Table 1 of Code Case N-722 requires bare metal visual examinations of the RV hot leg nozzle-to-safe end welds at a frequency of each refueling outage. In lieu of bare metal visual examination, Footnote 5 of Table 1 allows ultrasonic testing (UT) to be performed from the component inside diameter (ID) or outside diameter (OD) surface in accordance with the requirements of Table IWB-2500-1 and Appendix VIII of the ASME Code, Section XI, each refueling outage. Application of UT requirements specifies a demonstrated Root Mean Square Error (RMSE) of 0.125 inches if depth sizing is used.

Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds," which has been approved by the NRC staff in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15, provides an alternative to the Appendix VIII, Supplement 10 qualification requirements for dissimilar metal piping welds. Paragraph 3.3(c) indicates that examination procedures, equipment, and personnel are qualified for depth-sizing when the RMSE of the flaw depth measurements, as compared to the true depths, does not exceed 0.125 inches

Relief Request IR-3-13 authorizes the use of an alternate depth-sizing qualification requirement with a RMSE of no greater than 0.224 inches.

3.4 DNC's Proposed Alternative

DNC proposes to perform volumetric examinations of the subject welds from the ID surface using UT every other refueling outage in lieu of ASME Code Case N-722 required examination frequency of each refueling outage.

3.5 DNC's Basis and Justification

The licensee stated that the unique design of insulation packages surrounding the RV hot leg nozzle-to-safe end welds at MPS3 restricts access to the bare metal to perform the required bare metal visual examinations specified in ASME Code Case N-722. The licensee stated that the removal of insulation packages surrounding the welds to gain access to the bare metal is impractical due to the unique bolt in-place design for attaching the insulation packages to each nozzle. The insulation package is comprised of at least 14 blocks of heavy insulation materials weighing from 200 lbs to 1200 lbs each, bolted in-place, in a restricted location under the pit seal of the RV flange. For removal and reinstallation of each nozzle's insulation packages, scaffolding would need to be erected. The heavy insulation packages surrounding each nozzle need to be rigged to make the bare metal visual inspections of each weld possible. Removal of these insulation blocks to permit the bare metal visual examination is estimated to require 105 hours of work per nozzle, with an estimated personnel exposure to radiation dose of 3.69 Rem per nozzle. The total radiation dose impact for the bare metal visual examinations of the four hot leg nozzle-to-safe end welds is estimated to be approximately 15 Rem.

The licensee stated that a flaw tolerance evaluation was performed for the RV hot leg nozzle-to-safe end welds of MPS3 in order to provide technical basis and justification for extending the Code Case N-722 volumetric examination requirements of the RV hot leg nozzle-to-safe end welds from once each refueling outage to every other refueling outage. The flaw tolerance evaluation postulates an initial flaw and projects its subsequent growth in the time between examinations based on accepted flaw growth correlations and the limits of flaw stability identified in IWB-3640 of the ASME Code, Section XI. The licensee stated that the flaw tolerance evaluation shows that between examinations a postulated initial ID surface connected flaw in the subject welds will not propagate to the extent that the ASME Code, Section XI, IWB-3640 limits are exceeded.

The licensee stated that past volumetric examinations achieved the Section XI required coverage and that no flaws were detected in the subject welds. The licensee's independent flaw tolerance evaluation assumed an initial circumferential flaw and an axial flaw at the nozzle-to-safe end Alloy 82/182 welds with 10% through-wall depth and limited length. The licensee also stated that the assumed flaw depth is similar to the ISI acceptance criteria of Table IWB-3514-2 of the ASME Code, Section XI, for returning components into service and therefore is a conservative and reasonable assumption. Further, the licensee stated that an aspect ratio (AR) of 2, the flaw length divided by flaw depth, was assumed for the axial flaw since the PWSCC is limited to the width of the Alloy 82/182 weld. In addition, an AR of 6 was assumed for circumferential flaw for the outlet nozzles. As for the circumferential flaw, an initial flaw depth would be 0.25-inch and an initial flaw length would be 1.58-inches for the outlet nozzles. Since no detectable flaws were found in the subject DM welds during the spring 2007 volumetric examinations, the licensee considered it highly unlikely, with the qualified volumetric examinations, that a flaw of this size would go undetected. Moreover, the licensee stated that the residual stresses considered in the analyses were based on the RV nozzle residual stress profiles with no ID surface weld repair, since an extensive review of the original fabrication radiography and the available manufacturing records of the subject welds did not show any significant ID surface weld repairs. In addition, the licensee stated that the through-wall stress distribution profile was represented by a cubic polynomial.

The licensee summarized the results of its flaw tolerance evaluation in flaw evaluation charts (i.e. plots of the flaw depth to wall thickness ratio versus the inverse of flaw AR), displaying allowable flaw size curves for plant operation duration of 36 months and 54 months. These flaw evaluation charts show available margin for an assumed initial flaw size. The flaw evaluation charts show that if a given flaw falls below the allowable flaw size curve for a given plant operation duration, then the flaw will not grow to the maximum end-of-evaluation period allowable flaw size and will not result in leakage. In addition, the licensee presented PWSCC crack growth curves which displayed the service life required to reach either the IWB-3640 acceptable flaw size or a 100% through-wall flaw. The licensee stated that based on the IWB-3640 end-of-evaluation period allowable flaw size, it would take at least 48 effective full-power months (EFPM), for an axial flaw with AR of 2 and an initial 10% through-wall depth, to reach the end-of-evaluation period allowable flaw depth. For a circumferential flaw with AR of 6 and an initial 10% through-wall depth, it would take 46.2 EFPM to reach the end-of-evaluation period allowable flaw depth. In addition, the service life required for both axial and circumferential flaws to reach 100% through-wall thickness is slightly longer. The licensee stated that for conservatism, it proposes to volumetrically inspect the RV hot leg nozzle-to-safe end DM welds of MPS3 every other operating cycle (approximately 36 months). The licensee also indicated that it is only the through-wall flaw that is detectable by bare metal visual examination and since the flaw analysis is based on limits prior to challenging IWB-3640 or leakage, the alternative examination frequency does not introduce any significant risk.

### 3.7 Duration of Relief

The relief is requested for the third 10-year ISI interval of MPS3, which began on April 23, 2009, and is scheduled to end on April 22, 2019.

## 4.0 STAFF EVALUATION

The NRC staff has reviewed the information provided by the licensee in RR IR-3-10, Revision 1, which pertains to the frequency of volumetric examinations in lieu of bare metal visual examinations as defined by ASME Code Case N–722 for the RV hot leg nozzle-to-safe end welds. Licensees are required to meet Code Case N-722 with conditions in accordance with 10 CFR 50.55a(g)(6)(ii)(E). The filler materials, Alloy 82/182, used in the RV hot leg nozzle-to-safe end DM butt welds of MPS3 are susceptible to PWSCC which is a safety concern. Table 1 of Code Case N-722 requires bare metal visual examinations of the RV hot leg nozzle-to-pipe DM welds each refueling outage. In lieu of bare metal visual examinations, Footnote 5 of Table 1 of Code Case N-722 allows UT examinations to be performed from the ID or OD surface at the same frequency. The licensee proposed to perform UT examinations of the RV hot leg nozzle-to-safe end DM welds of MPS3 from the ID surface every other refueling outage in lieu of Code Case N-722 required examination frequency of each refueling outage. To justify extending the Code Case examination frequency requirement, the licensee performed a flaw tolerance analysis to show that between examinations a postulated initial ID surface-connected flaw in the subject welds will not propagate to the extent that the ASME Code, Section XI, IWB-3640 limits are exceeded.

After reviewing the licensee's flaw tolerance analysis, the NRC staff independently performed flaw evaluation and sensitivity analyses because of concerns with the licensee's flaw tolerance analysis. These analyses were performed to confirm DNC's results. The NRC staff's concerns include the following:

a. The licensee assumed a postulated undetected initial circumferential or axial flaw of 10% through-wall at the nozzle-to-safe end Alloy 82/182 weld. This assumption is potentially non-conservative since it may not be reasonable to expect that all 10% through-wall flaws would be reliably detected.

b. An AR of 6 was assumed for the circumferential flaw. This assumption may be non-conservative since longer flaws attributed to PWSCC have been detected in recent years.

c. The licensee assumed in its analysis that no repairs had been made to the welds during construction. This assumption is potentially non-conservative since the records created by the original manufacturer would not necessarily include all construction repairs made to welds.

d. The residual stress distribution profile was represented by a cubic polynomial. Use of a cubic polynomial to represent the residual stress distribution is potentially non-conservative.

The NRC staff has performed an independent flaw analyses to determine the leak and rupture characteristics of the subject welds to a postulated ID surface-connected flaw. The analyses were performed based on the requirements of the ASME Code, Section XI, IWB-3640 and assumed that the flaws were due to PWSCC. The piping loads from various loading conditions, the weld thickness, and weld diameter provided in RR IR-3-10 were used in the analysis. The safe-end as-built length of 2.26 inches for the MPS3 hot leg nozzle, provided by the licensee in the February 3, 2010[1], letter was similar to the safe-end length assumed for the vessel nozzle in the NRC staff's analyses.

For the geometry of the vessel nozzle, the NRC staff used the geometry from a previous NRC staff analysis of a Westinghouse vessel nozzle. The geometry of the MPS3 vessel nozzle was assumed to be similar to the geometry in the previous analysis for the purposes of the NRC staff's weld residual stress (WRS) calculations. The NRC staff used two-dimensional, axi-symmetric finite element (FE) models to perform the WRS analyses. The low alloy steel nozzle, Alloy 82/182 butter and nozzle-to-safe end DM weld, stainless steel safe-end, and safe end-to-pipe stainless steel butt weld were all modeled in the FE analysis. The weld modeled was a U-groove type weld. Typical size weld beads were simulated in the WRS calculation.

[1] The 2.26 inch dimension in the February 3, 2010, letter was confirmed to be the as-built length as documented by e-mail dated February 23, 2010 (ADAMS Accession No. ML100610235).

In these analyses, the NRC staff performed a sensitivity study to understand the effects of certain uncertainties on the results. In all cases, the NRC staff assumed an initial flaw depth of 25%. The NRC staff computed WRS assuming that a 50% repair was made in the DM metal weld during construction prior to simulating the safe end-to-pipe field weld. In addition, the NRC staff analyzed cases involving 15% back chipping and last pass welding from the ID. The NRC staff also analyzed cases with and without a safe end-to-pipe field weld. Further, the NRC staff analyzed a case with a 50% partial arc repair without the safe end-to-pipe field weld.

The NRC staff's analyses assumed a semi-elliptical flaw shape with an initial AR of 2 and permitted the aspect ratio to be governed by the growth of the flaw at the surface and at the deepest part of the flaw. The NRC staff used the Anderson's influence function to calculate the stress intensity value (K-value) at these two locations. Conducting these analyses and allowing both the depth and length to change as driven by the K-value allows a more natural aspect ratio to form than assuming a constant aspect ratio. The final aspect ratio in the NRC staff's analyses and in all cases exceeded the licensee's assumed AR of 6.

For the PWSCC crack growth rate, the NRC staff used the material reliability program (MRP) -115 growth rates at 75% percentile of the data. The NRC staff used the net-section collapse criteria, also known as the limit load analysis, and Alloy 182 Z-factor as developed by Engineering Mechanics Corporation of Columbia, "Determination of the Elastic-Plastic Fracture Mechanics Z-factor for Alloy 82/182 Weld Metal Flaws for Use in the ASME Section XI Appendix C Flaw Evaluation Procedures." The limit load analysis is documented in ASME Code, Section XI, Appendix C, Article C-5000.

The NRC staff calculated time to first leakage and time between leakage and rupture. The NRC staff's analyses confirmed the analyses performed by DNC. Based on the DNC analyses and the NRC staff's confirmatory analyses results, the NRC staff has concluded that performing UT examinations every other refueling outage would provide an acceptable level of quality and safety.

## 5.0 CONCLUSION

Based on the above review, the NRC staff concludes that the proposed alternate examination frequency for the RV hot leg nozzle-to-safe end welds in the reactor coolant system provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for the remainder of the MPS3 third 10-year ISI interval.

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.

Principal Contributor: A. Rezai

Date: April 16, 2010

at 301-415-1603.

Sincerely,

*/ra/*

Harold Chernoff, Chief
Plant Licensing Branch 1-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

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