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

November 30, 2009

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-3-04 REGARDING USE OF AMERICAN SOCIETY OF MECHNICAL ENGINEERING CODE, SECTION XI, 2004 EDITION (TAC NO. ME1256)

Dear Mr. Heacock:

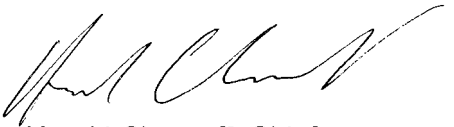
By letter dated April 28, 2009, as supplemented by the letter dated August 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091310666 and ML092390141, respectively), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) submitted relief requests for the third 10-year in-service inspection (ISI) interval at Millstone Power Station, Unit No. 3 (MPS3). The licensee requested the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2004 Edition, no addenda requirements. Relief Request IR-3-04 proposed authorization to defer repair/replacement activities for degraded brazed joints in the Class 3 water system as an alternative to the requirements of ASME Code, Section XI, 2004 Edition, no addenda.

The results of the Nuclear Regulatory Commission (NRC) staff's review, as contained in the enclosed Safety Evaluation, indicate that DNC's performance of an ASME Code repair or replacement of the degraded brazed joints would result in hardship without a compensating increase in the level of quality and safety.

Therefore, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.55a(a)(3)(ii), DNC's request for relief is authorized for the remainder of the third 10-year ISI interval for MPS3 on the basis that the proposed brazed joint assessment methodology in IR-3-04, as an alternative to the requirements of ASME Code, Section XI, 2004 Edition, is acceptable because it provides reasonable assurance of structural integrity of the degraded brazed joints. The third 10-year ISI interval for MPS3 began on April 23, 2009, and is scheduled to be completed on April 22, 2019.

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

RELIEF REQUEST IR-3-04

FOR THE THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NUMBER 50-423

1.0 INTRODUCTION

By letter dated April 28, 2009, as supplemented by letter dated August 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML091310666 and ML092390141, respectively), Dominion Nuclear Connecticut, Inc., (DNC or the licensee), submitted for review and approval Relief Request IR-3-04 to defer repair/replacement activities for degraded brazed joints in the Class 3 Service Water System for Millstone Power Station, Unit No. 3 (MPS3) during the third 10-year in-service inspection (ISI) interval. The April 28, 2009, letter requested relief from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2004 Edition, no addenda.

On August 24, 2009, the licensee submitted the revised Relief Request, IR-3-04, Revision 1, as part of its response to the U.S. Nuclear Regulatory Commission (NRC) staff's request for additional information.

On February 28, 2007 (ADAMS Accession No. ML070580514), the NRC approved a similar Relief Request, IR-2-38, for the second 10-year ISI interval for MPS3. Relief Request IR-3-04 is the continuation of Relief Request IR-2-38 and will be used in the third 10-year ISI interval, which began April 23, 2009, and is scheduled to be completed on April 22, 2019.

2.0 REGULATORY REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g) specifies that ISI of nuclear power plant components shall be performed in accordance with the requirements of the ASME Code, Section XI, except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) of 10 CFR 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 preservice 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 ISI 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 Code of record for the third 10-year ISI program and its evaluation at MPS3 is the ASME Code, Section XI, 2004 Edition with no addenda.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

3.1.1 Components for Which Relief is Requested

ASME Code Class 3 service water system piping with brazed joints with the nominal pipe size of 3 inches and smaller.

3.1.2 Applicable ASME Code Edition and Addenda

The Code of record for the third 10-year ISI program and its evaluation at MPS3 is the ASME Code, Section XI, 2004 Edition with no Addenda.

3.1.3 Applicable ASME Code Requirement

If leakage of a Class 3 brazed connection is discovered during the course of normal operation, the ASME Code, Section XI, article IWA-4000, "Repair/Replacement Activities," applies and the joint must be repaired or replaced in accordance with that article. However, if the leakage is discovered during a scheduled leak test, the joint must be evaluated and repaired in accordance with IWD-3000, "Acceptance Standards," as clarified by the following:

ASME Code, Section XI, IWD-3000, does not have acceptance criteria for Class 3 components. IWD-3500, "Acceptance Standards," refers to IWC-3500, "Acceptance Standards." IWC-3516, "Examination Category C-H, All Pressure Retaining Components," states that "[t]hese standards are in the course of preparation. The standards of IWB-3522 may be applied."

ASME Code, Section XI, IWB-3522.1 establishes the acceptance standards for visual examination, VT-2, in which leakage of non-insulated and insulated piping is listed as a relevant condition. IWB-3522.1 states that such relevant conditions that may be detected during the conduct of system pressure tests shall require correction to meet the requirements of IWB-3142, "Acceptance," and IWA-5250, "Corrective Action," prior to continued service.

ASME Code, Section XI, IWA-5250, in the context of a system leak test, requires identification of the source of leakage for evaluation of its corrective action which may include repair/replacement activities.

ASME Code, Section XI, IWB-3142, permits acceptance of visually identified conditions under the requirements of IWB-3142.2, "Acceptance by Supplemental Examination."

ASME Code, Section XI, IWB-3200, "Supplemental Examinations," permits supplemental surface or volumetric examinations to determine the extent of the unacceptable conditions and the need for corrective measures, repairs, analytical evaluation, or repair/replacement activities.

3.1.4 Licensee's Proposed Alternative and Basis for Use

In the course of plant operation, brazed joints are sometimes observed to be leaking at a very low rate (i.e. weepage) through a defect in the braze bond between the pipe and fitting. Applicable ASME Code requirements depend on whether the leak is discovered in the course of normal plant operation or during a scheduled leak test. As discussed above, the ASME Code, Section XI, does not have rules applicable to evaluation of weepage through brazed joints caused by defects in braze bonding between piping and fittings.

In lieu of the immediate repair requirement of the ASME Code, Section XI, IWA-5250 or IWA-4000, the licensee proposed an alternative which requires that a supplemental ultrasonic test (UT) examination be performed and the inspection results be compared with alternative acceptance criteria. The UT examination will establish the extent of braze bond within the joint. The UT results will be compared with pre-established brazed joint bond levels required for structural integrity of the specific piping under consideration that accounts for the design basis loadings applicable to the condition. This will establish the basis for determining joint integrity to the extent required for system operability.

The lack of full braze bonding originates from construction or fabrication, and is not progressive over time. However, the proposed methodology provides for continued monitoring until a resolution of the nonconforming condition (e.g., weepage) occurs through repair/replacement activities. Periodic monitoring of the joint and its leakage verifies that assumptions used for the assessment remain valid. The licensee has validated the overall methodology by performance of physical testing on an array of simulated bond configurations, and several brazed joints salvaged from MPS3 piping.

The proposed alternative is limited to brazed service water piping (typically constructed of copper-nickel or Monel piping and cast bronze fittings) or on-skid equipment piping that has a design pressure of 150 psig or less and a design temperature of 150 degrees Fahrenheit or less. The piping nominal size is limited to 3 inches (nominal diameter) maximum. The brazed joint will be examined by UT using a straight beam technique that monitors the relative strengths of signals returned from the internal diameter of the pipe and the fitting.

An assessment of the joint integrity using this methodology includes the following considerations: system performance and indirect effects assessments, adjustment of bond readings to account for uncertainties, a review of design basis stress analysis of the piping to determine required joint strength, and comparison of the adjusted bond readings with the

prequalified bond levels that have been shown empirically by physical testing to assure structural integrity.

As a prerequisite to structural assessment, knowledgeable engineering personnel assess the effect of the leak on the system and other nearby equipment. Typically, a brazed joint with a defect in the braze material bonding will leak only drops per minute. The actual leak rate will be estimated and compared to service water system margins for loss or diversion of flow. In addition, a walkdown will be performed to identify any nearby equipment that may be affected by dripping or impingement spray from the leak. If required, a drip collection device or spray shield will be installed and maintained for the duration that the leak continues.

The proposed alternative specifies that if the average measured bond reading is 60 percent or above, then no further assessment is required since the bond strength exceeds piping strength. If the average is less than 60 percent, then the bond readings, as documented in the UT procedure, are adjusted downwards on a sliding scale, such that all readings at 10 percent and below are assumed to be zero,

The proposed alternative requires that the licensee assess structural integrity of the joint by reviewing the Construction Code qualification stress analysis of record to determine design basis loadings at the subject braze joint. Pressure, deadweight, and safe shutdown earthquake (SSE) loadings are included. The loads are either used directly or expressed in terms of equivalent pipe stress so that stress analysis outputs may be used directly.

The proposed alternative requires periodic monitoring to assure that the assumptions of the assessment remain valid. This monitoring will be in addition to the normal daily plant operator rounds during which personnel are observant for signs of leakage. The monitoring will be by visual observation of the appearance of the joint and its leak rate, plus re-examination of the joint by UT to reconfirm the percent bonding. The frequency of the monitoring will be approximately once every 3 months. The monitoring will continue as described until the joint is repaired or replaced. If there are changes in the nonconforming condition of an evaluated brazed joint with weepage that may impact its assessment for adequate structural integrity or its functionality, the licensee will generate a Condition Report in accordance with the Millstone Power Station Corrective Action Program and the UT readings on the joint will be repeated and reassessed.

If the assessment can conclude that a brazed joint with leakage retains adequate structural integrity and functionality, an operability determination can be used to document an operable but not fully-qualified status. A timely repair/replacement activity can be planned, commensurate with safety, and in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Consistent with the Millstone Power Station Corrective Action Program, the permanent Code repair/replacement for this nonconforming condition will be considered timely when completed during the next cold shutdown of sufficient duration, or the next refueling outage, whichever comes first.

If a joint does not have adequate bond by the assessment, the methodology for determining the adequacy of structural integrity of the joint is not applicable. Prompt repair/replacement of the joint, or temporary non-Code repairs subject to NRC review and approval may be an option, consistent with considerations in NRC's Regulatory Issue Summary 2005-20, "Operability

Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality of Safety," for the resolution of degraded and nonconforming conditions.

The proposed alternative requires that if leakage is detected in one brazed joint, the licensee will perform augmented leakage examination (i.e., expanded sample inspection) up to five similar brazed joints. The additional joints will be selected based on consideration of adjacency, opposite train, fitting type, or other factors that may be evident from the specific condition. Selected joints for augmented examination will be consistent with ASME Code Case N-513-2, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1." If leakage is observed in similar joints in the augmented examination, the resolution of each nonconforming condition will be evaluated in accordance with the Millstone Power Station Corrective Action Program, and the extent of condition will be documented and addressed.

3.5 Duration Of The Relief Request

DNC requested approval of Relief Request IR-3-04 for the third 10-year ISI interval, which began on April 23, 2009, and is scheduled to be completed on April 22, 2019.

3.2 NRC Evaluation

The NRC approved Relief Request IR-2-38 for the second 10-year ISI interval at MPS3, which is the precedent to Relief Request IR-3-04, Revision 1. Both relief requests provide the same requirements and technical basis for the structural integrity assessment of degraded brazed joints. The NRC staff did not repeat the same evaluation for IR-3-04, as was previously performed for IR-2-38 in 2007. Instead, the NRC staff evaluated the changes between Relief Request IR-3-04 and Relief Request IR-2-38 and the past operating experience of using Relief Request IR-2-38.

The licensee stated that the plant must be shut down in order to perform the ASME Code repair on the degraded brazed joint because certain safety-related systems or components will not be available during the repair, which would be a violation of the Technical Specification requirements. A plant shutdown would unnecessarily cycle plant components, which is not desirable in maintaining the structural integrity of the safety-related components. The NRC staff finds that the need to shut down the plant for implementing an ASME Code repair of the degraded braze joint would result in hardship to the licensee without a compensating increase in the level of quality and safety when the structural integrity of the degraded joint and the system functionality can be ensured by appropriate evaluation.

The licensee's proposed structural integrity assessment methodology will allow the degraded brazed joint to remain in service for a limited period of time provided that the structural integrity of the degraded brazed joint is assured by UT examination and/or analytical evaluation and the effects of leakage are appropriately assessed and mitigated to ensure the functionality of the affected system.

The August 24, 2009, letter revised the original version of IR-3-04 with applicable changes to the final brazed joint methodology. In addition, updated responses to the requests for additional

information and supplementary responses associated with the review of Relief Request IR-2-38 have also been included in Revision 1 of IR-3-04. The NRC staff finds that Relief Request, IR-3-04, Revision 1, contains the same requirements and technical basis as Relief Request IR-2-38 and, therefore, is acceptable.

In Section 5.3.4 of Relief Request IR-3-04, the licensee discussed the use of safety factors from ASME Code, Section XI, Appendix C paragraph C-3320(b) and C-3420(a), with these being the same safety factors permitted in ASME Code, Section XI, Code Case N-513-1. The NRC staff noted that Code Case N-513-1 has been superseded by Code Case N-513-2, as stated in Regulatory Guide 1.147, Revision 15.

The August 24, 2009, letter states that the safety factor of 1.5 that is utilized in the brazed joint assessment methodology is conservative relative to testing that was performed on brazed piping joints having actual and simulated flaws, as described in the submittal and its supplements. The discussion of safety factors from the ASME Code for piping components and flaws in the submittal was intended to show that, while the joint configuration is different than those directly addressed by Code rules (including Code Case N-513-2), the selected safety factor was comparable and consistent with Code guidance. Code Case N-513-2 does not directly contain safety factors but instead refers to ASME Code, Section XI, Appendix C. The safety factor of 1.5 used for brazed joint assessment remains conservative to the level D safety factors of 1.3 for SFm (safety factor for membrane stress based on service level, dimensionless) and 1.4 for SFb (safety factor for membrane stress based on service level, dimensionless) listed in the ASME Code, Section XI, 2004 Edition, Appendix C, paragraph C-2621 for circumferential flaws; and 1.3 for SFm as listed in paragraph C-2622 for axial flaws. The licensee indicated that the reference to Code Case N-513-1 was inadvertent.

The NRC staff finds that DNC has used appropriate safety factors from the ASME Code, Section XI, Appendix C to demonstrate structural integrity of the degraded braze joint.

The August 24, 2009, letter states that the brazed joint methodology, as finally approved, was employed in three instances during the last (i.e., second) ISI interval. In two of the instances, the leaking brazed joints were repaired within 90 days of discovery and in the third case, the repair was performed at the next refueling outage. For the third case, the licensee submitted a summary of the original assessment, subsequent UT monitoring, and final repair. The NRC staff reviewed the summary of actual application and finds that the licensee followed the specific requirements in Relief Request IR-2-38 to disposition a degraded brazed joint. Under Relief Request IR-2-38, the three degraded brazed joints did not fail catastrophically and the degraded joints were repaired within the required period.

This operating experience demonstrates that the methodology proposed in Relief Request IR-3-04, Revision1, which is the same as in Relief Request IR-2-38, provides reasonable assurance that structural integrity of the degraded brazed joints are maintained.

4.0 CONCLUSION

On the basis of the above review, the NRC staff concludes that performance of an immediate ASME Code repair or replacement of the degraded brazed joints at MPS3 would result in hardship without a compensating increase in the level of quality and safety. The NRC staff also

concludes that the proposed brazed joint assessment methodology in Relief Request IR-3-04, Revision 1, as an alternative to immediate ASME Code repair or replacement, is acceptable because it provides reasonable assurance of structural integrity of the degraded brazed joints. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the proposed alternative in Relief Request IR-3-04, Revision 1, for MPS3 for the remainder of the 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: J. Tsao

Date: November 30, 2009

If you have any questions, please contact the Project Manager, Carleen Sanders, 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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