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

September 4, 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. 3 - RELIEF FROM THE REQUIREMENTS OF THE ASME CODE SECTION XI, REQUIREMENTS FOR REPAIR/REPLACEMENT OF CLASS 3 SERVICE WATER VALVES (TAC NO. MF1314)

Dear Mr. Heacock:

By letter dated March 28, 2013, as supplemented by letter dated February 6, 2014, Dominion Nuclear Connecticut, Inc. (DNC, the licensee) submitted relief request IR-3-17 to the Nuclear Regulatory Commission (NRC) for relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for repair/replacement of ASME Code Class 3 service water valves.

DNC requested relief for seven ball valves up to 3 inches in size because they could not be isolated and replaced while operating. DNC requested relief for a few weeks until the April 2013 refueling outage started and the valves could be replaced. Although the NRC review could not be completed entirely before the valves were replaced in the refueling outage, the completed review affirmed the acceptability of the relief.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii) the licensee requested relief from ASME Section XI, IWA-5250(a)(3) and IWA-4000 to defer the repair/replacement of weeping ASME Code Class 3 aluminum bronze service water valves until the next refueling outage scheduled for April 2013 on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the licensee's request and concludes that, as set forth in the enclosed safety evaluation, the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the subject components and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative at Millstone Power Station Unit 3 for a period of time not to extend beyond the end of the refueling outage performed in April 2013.

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

If you have any questions, please contact Mohan Thadani, at (301) 415-1476 or email Mohan.Thadani@nrc.gov.

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. 50-423

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. IR-3-17

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT 3

1.0 INTRODUCTION

By letter dated March 28, 2013 (Agencywide Document Access and Management System (ADAMS) Accession No. ML13091A038), as supplemented by letter dated February 6, 2014 (ADAMS Accession No. ML14041A178), Dominion Nuclear Connecticut, Inc. (the licensee), requested relief from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for repair/replacement of ASME Code Class 3 service water valves.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii) the licensee requested relief from ASME Section XI, IWA-5250(a)(3) and IWA-4000 to defer the repair/replacement of weeping ASME Code Class 3 aluminum bronze service water valves until the refueling outage, which occurred in April 2013, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY REQUIREMENTS

In this relief request the licensee requests authorization of an alternative to the regulatory requirements of articles IWA-5250(a)(3) and IWA-4000 of Section XI of the ASME Code pursuant to 10 CFR 50.55a(a)(3)(ii).

Adherence to article IWA-4000 of Section XI of the ASME Code is mandated by 10 CFR 50.55a(g)(4) which states, in part, that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, or Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME Boiler and Pressure Vessel Code.

The 10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if the licensee demonstrates (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.

Based on the above analysis, the NRC staff concludes that regulatory authority to authorize an alternative to article IWA-4000 of Section XI of the ASME Code, as requested by the licensee, exists.

3.0 TECHNICAL EVALUATION

3.1 Applicable Code Edition and Addenda

Repair and Replacement: ASME Boiler and Pressure Vessel Code, Section XI, 2004 Edition, No Addenda

3.2 Components for Which Relief is Requested

ASME Code Class: ASME Code Class 3
Reference: ASME Section XI, IWA-5250
Examination Category: D-B
Item Number: D2.10
Description: Millstone Power Station Unit 3 "A" and "B" trains, Service Water System Valves
Components: Neles-Jamesbury valves fabricated from alloy C95400 aluminum bronze (SB 148 Grade CDA 954). See below.

Valve Number	Train	Valve Size/Type	Descriptor
3SWP*V658	A	3" Flat Face Ball Valve	Suction isolation valve for the 'A' MCC and Rod Control area service water booster pumps
3SWP*V659	A	3" Flat Face Ball Valve	Return isolation valve for the 1A MCC and Rod Control area room cooler
3SWP*V661	B	3" Flat Face Ball Valve	Return isolation valve for the 1B MCC and Rod Control area room cooler
3SWP*V663	A	1 1/2" Flat Face Ball Valve	Return isolation valve for the 'A' charging pump cooler
3SWP*V664	B	1 1/2" Flat Face Ball Valve	Supply isolation valve for the 'B' charging pump cooler
3SWP*V665	B	1 1/2" Flat Face Ball Valve	Return isolation valve for the 'B' charging pump cooler
3SWP*V668	B	3" Flat Face Ball Valve	Suction isolation valve for the 'B' MCC and Rod Control area service water booster pumps

3.3 Reason for Request

In its request the licensee stated that:

1. the valves are constructed from aluminum bronze alloy C95400 which is subject to selective leaching;

2. the subject valves were observed to be weeping;

3. the observed weeping was attributed to selective leaching;

4. the subject valves were not isolatable;

5. replacement valves were not available and were not to be available within the 72 hour period available to return two trains of service water to service. As a result, if this request were not authorized, a plant shutdown would have been required;

6. the valves were being replaced in the refueling outage (April 2013). (The NRC staff notes that the valves were discovered weeping approximately 1 month prior to the proposed replacement date.);

7. replacing the subject valves at the next refueling outage will not create a safety issue, because selective leaching progresses slowly, i.e., a significant change in the condition of the valve is not expected during the interval prior to April 2013 refueling outage.

Based on the above information the licensee proposes that making the repairs prior to the April 2013 refueling outage constitutes a hardship without a compensating increase in the level of quality or safety.

3.4 Proposed Alternative

The licensee's proposed alternative consists of three actions:

1. At the next refueling outage (April 2013) replace the valves with valves constructed from a more suitable material;

2. Once per shift inspect the valves for leakage. If leak rates exceed one drop per minute an engineering evaluation will be conducted;

3. To provide defense in depth, install a structurally robust enclosure around the weeping valves. The enclosure will capture the ends of the piping and will be sealed at its mating surfaces and where the enclosure surrounds the pipe. The enclosure will be designed to withstand normal operating pressure and design temperature of the service water system and will be seismically supported. The

enclosure will have a drain valve which will be maintained in the open position to facilitate leakage monitoring but can be closed at the discretion of the shift manager

3.5 Licensee's Technical Basis

In its request the licensee provides the following information in support of its request to defer repair/replacement of the subject components until the next refueling outage which was scheduled for April 2013:

a. For the latter portion of the time period covered by this relief request, i.e., from the time that a mechanical enclosure was installed until the time the subject valves are replaced, the licensee proposed that:

   1. The mechanical enclosure as described in section 3.4, above provides defense in depth such that the mechanical enclosure will provide both structural and leak tight integrity to the piping system in the event that the valves fail structurally or that the leak rate increases unacceptably.

b. For the entire period of the time period covered by this relief request, the licensee proposed that:

   1. The piping system under consideration is a moderate energy system (100 psi design, 50 psi operating at a temperature of between 33°F and 75°F).

   2. Selective leaching is a degradation mechanism which progresses slowly; little change in the condition of the valves is expected between when leakage was detected and the next scheduled refueling outage (a time period of roughly 1.5 months).

   3. The potential for spray from the subject valves has been analyzed. Adjacent safety related equipment is either designed to operate in the presence of spray or is shielded.

   4. The potential for flooding from the subject valves has been analyzed. The flooding analysis for this location bounds the possible conditions resulting from a failure of the subject valves.

   5. Using experimentally determined material properties for aluminum bronze material of the same type but different heat treatment from Surry Power Station which was reported to be 100 percent selectively leached based on visual observation (color), the subject valves meet the structural requirements of the ASME Code.

   6. The heat treatment used for the subject valves is considered to be more resistant to selective leaching than that used for the Surry valves. As a

result the material properties of the subject valves are expected to be better than those of the Surry valves.

c. Following removal of the subject valves in April 2013, the valves were subjected to destructive analyses. These analyses included:

1. Hydrostatic testing which indicated that the valves did not structurally fail at pressures well in excess of system design pressure.

2. Chemical testing of degraded areas of the valves to determine the extent to which aluminum had been leached from the material.

3. Metallographic analysis.

4. A bend test of the piping/valve system which demonstrated the ability of the valves to withstand bending loads significantly in excess of design requirements.

3.6 Staff Evaluation

The NRC staff notes that this safety evaluation covers two distinct periods of time. The first time period is from the discovery of leakage until the mechanical enclosures were installed. The second time period is from the installation of the enclosures until the valves were replaced.

As previously stated in the Regulatory Evaluation section (2.0) of this safety evaluation, prior to authorizing the proposed alternative under 10 CFR 50.55a(a)(3)(ii), the NRC staff must find that the technical information provided in support of the proposed alternative is sufficient to demonstrate that compliance with ASME Code Section XI, IWA-5250(a)(3) and IWA-4000; (a) would result in a hardship or unusual difficulty; and,(b) would not provide a compensating increase in the level of quality and safety when compared to the proposed alternative. If these criteria are met, the staff finds that the proposed alternatives to ASME Code requirements will provide reasonable assurance of structural integrity or leak tightness of the subject components.

The NRC staff reviewed the licensee's basis for hardship. The NRC staff concludes that the licensee's assertion that the subject valves are not currently available and, therefore, cannot be installed within the allowable 72 hour time period to be reasonable. The NRC staff also concludes that the licensee's assertion, that if repairs are not completed within 72 hours then a plant shutdown is required, to be accurate. The NRC staff conclude that the necessity to shut down the plant constitutes a hardship. This satisfies the first condition of 10 CFR 50.55a(a)(3)(ii).

In considering the second condition of 10 CFR 50.55a(a)(3)(ii), whether adherence to the ASME Code requirement would provide an increase in quality and safety commensurate with the hardship or unusual difficulty imposed by meeting the code requirement, the staff evaluated the technical basis for the alternative as proposed by the licensee and described in section 3.5, above.

For the second time period, i.e., between the installation of the mechanical enclosure and the replacement of the valves, the staff finds that the installation of an enclosure which is capable of withstanding piping design and seismic loads; sealing the valve against leakage so as to prevent spray and flooding in combination with visual examinations of the valve enclosure on a once per shift basis provides reasonable assurance of the structural and leak tight integrity of the system. Based on this conclusion, the NRC staff determines that adherence to the ASME Code requirement does not provide a compensating increase in the level of quality and safety when compared to the proposed alternative. For the second time period the NRC staff finds that the second criterion in 10 CFR 50.55a(a)(3)(ii) is, therefore, met.

For both time periods, i.e., from the discovery of leakage to the replacement of the valves the NRC staff considered the spray analysis, the flooding analysis and the structural analysis provided by the licensee. The NRC staff concluded that there is no reason to object to either the methods or the conclusion reached by the licensee concerning either spray or flooding as a result of leakage from the subject valves. In considering the structural integrity of the valves, the NRC staff considered the testing conducted following the removal of the valves. Given the limited range of valve sizes and material, the NRC staff considers the results of the hydrostatic testing in which the subject valves did not structurally fail at pressures far greater than the system design pressure, in conjunction with the bend test conducted in which one of the valves withstood bending stresses significantly in excess of design requirements to be sufficient evidence that the subject valves possessed adequate structural integrity throughout both time periods considered in this safety evaluation. Based on this conclusion the NRC staff determines that adherence to the ASME Code requirement does not provide a compensating increase in the level of quality and safety when compared to the proposed alternative. For both time periods the NRC staff concludes that the second criterion in 10 CFR 50.55a(a)(3)(ii) is, therefore, met.

In addition to the above evaluations, the NRC staff considered the information submitted by the licensee in support of the concept that the tensile test data obtained from the Surry Power Station valve represents a bounding condition which may be used in structural analyses. The licensee notes that the Surry data were from aluminum bronze alloy C95400 in the as cast condition and that the Millstone valves were constructed from the identical alloy but were provided with one of two heat treatments. The Millstone sand cast valves were provided with a temper anneal consisting of 1.5 hours at 1150-1200°F. The Millstone centrifugally cast valve received a quench and temper heat treatment (solution anneal at 1600°F followed by quenching and tempering). The valve provided with the quench and temper heat treatment was reported to have significantly lower degradation due to selective leaching than the other valves. The licensee proposes that both heat treatments limit the formation of the selectively leachable phase(s) when compared to the as cast condition and that the heat treated valves should, therefore, have better fully leached material properties than the as cast Surry valve.

The NRC staff concludes that the arguments presented by the licensee to be non persuasive for either the subject valves, other, especially larger, valves made from the same material, or valves made from different aluminum bronze alloys. This position is based primarily on consideration that:

1. The Surry data are from a very limited data set. Given the existence of part to part variability in material properties and tensile test results, a conclusion that the Surry data represent the minimum, fully degraded tensile strength for the alloy employed or for aluminum bronze in a more general sense, remains statistically questionable.

2. While the exclusion of the Surry datum which failed outside the gauge length would generally be considered appropriate when testing undamaged material, in the present case it raises questions regarding the potential existence of at least localized regions where the residual tensile strength of the material may have been significantly less than the final value proposed. The NRC staff recognizes that this issue can never be fully resolved.

3. The available data do not appear to support the concept that a component that visually appears to have been selectively leached 100 percent through wall will not experience further decline in material properties. The NRC staff's position on this issue is based on correlation of the visual appearance of the component with the decrease in the concentration of aluminum observed. It is presumed that material properties will correlate with the available aluminum concentration and will continue to fall until all the aluminum available in the leachable phase(s) has been removed. Some of the data, e.g., valve 663, shows a very distinct change in the concentration of aluminum associated with the visual line of demarcation between leaching and no leaching. Alternatively, other data, e.g., valves 662 and 664, visually appear to be 100 percent through wall leached. However, the residual concentrations of aluminum in these samples are much higher than those in valve 663. This indicates to the NRC staff that aluminum will continue to leach from these valves and that the tensile properties of this material will continue to degrade with time. Based on this information the NRC staff concludes that, in the absence of chemical data supporting visible observations, it is not possible to conclude whether a component is fully leached. Given that chemical data are not available for the Surry data, these data cannot be assumed to represent a lower bound for the properties of the material.

4. Due to the fact that valves from both Surry and Millstone experienced through wall leakage, it is apparent that the heat treatment approach used, as cast or temper anneal, was not effective. Despite the apparent failure of both heat treatment approaches the NRC staff considered the assertion by the licensee that the Millstone heat treatment approach was better than the Surry approach and could be expected to produce better material properties. To the contrary, based on information available from the Copper Development Association (Aluminum Bronze Alloys Corrosion Resistance Guide, Publication 80, 1981), it appears to the NRC staff that the microstructure produced by the temper anneal heat treatment used at Millstone should be expected to be more susceptible to selective leaching than the as cast microstructure found in the Surry valves. This observation is based on the concept that, unlike nickel aluminum bronze where prolonged time at temperature, e.g., 6 hours at 675°C, is used to convert the beta phase to alpha plus kappa (which is resistant to selective leaching), in aluminum bronzes (without nickel) cooling slowly from above 600°C permits the beta phase to convert to alpha plus gamma 2 (which is susceptible to selective leaching).

This information appears to indicate that the temper anneal provided to the Millstone valves provided a more, rather than less, susceptible microstructure when compared with the Surry valves.

5. The preceding analysis does not, however, appear to apply to the quench and temper heat treatment applied to the Millstone centrifugally cast valve. The quench and temper heat treatment reheats the as cast valve sufficiently that any gamma 2 phase formed during cooling from casting is redissolved. Quenching minimizes the time in which the gamma 2 phase can form and, therefore, the quantity which does form. Tempering, if done at a sufficiently low temperature will improve the properties of the valve without creating any gamma 2 material.

Based on the above analysis, the NRC staff concludes that the information provided by the licensee in support of the concept that the Surry data establish a minimum tensile strength which can be used in ASME Code calculations to demonstrate acceptable structural strength of a component to be non-persuasive. However, the NRC staff finds that the presence of the mechanical enclosure, the hydrotest results, and the other data obtained through destructive analysis of the valves to be sufficient to demonstrate the structural integrity of the subject valves (and no others). Therefore, the NRC staff concludes that, for the subject valves, the technical requirements of 10 CFR 50.55a(a)(3)(ii) have been met and that the licensee's proposal provides reasonable assurance of structural and leak tight integrity of the subject components. The staff, therefore, has no technical basis that would preclude it from authorizing an alternative to Article IWA-4000, of Section XI, of the ASME Code as requested by the licensee.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the subject components and that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the proposed alternative at Millstone Power Station Unit 3 for a period of time not to extend beyond the end of the refueling outage of April 2013.

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

Principle Contributor: D. Alley

Date: September 4, 2014

If you have any questions, please contact Mohan Thadani, at (301) 415-1476 or email Mohan/Thadani@nrc.gov.

Sincerely,

/RA/

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

Docket No. 50-423

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As stated

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