



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

August 12, 2015

Mr. Kevin Davison
Site Vice President
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 –
REQUESTS 1-RR-5-3 AND 2-RR-5-3 ASSOCIATED WITH THE FIFTH 10-YEAR
INTERVAL FOR THE INSERVICE INSPECTION PROGRAM (TAC
NOS. MF4837 AND MF4838)

Dear Mr. Davison:

By letter dated September 15, 2014, Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, submitted a request for use of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2007 Edition through the 2008 Addenda, for the Prairie Island Nuclear Generating Plant, Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(a)(3)(i) (retitled paragraph 50.55a(z)(1) by *Federal Register* notice 79 FR 65776, dated November 5, 2014), the licensee submitted requests 1-RR-5-3, Revision 0, and 2-RR-5-3, Revision 0, to use the alternative for bolted connections in borated water systems on the basis that the alternative examination provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the proposed alternative and determined, as set forth in the enclosed safety evaluation, that NSPM adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), and that the proposed alternative provides reasonable assurance of structural and leak-tight integrity of the mechanical connections. The NRC staff authorizes the use of 1-RR-5-3 and 2-RR-5-3 at Prairie Island, Units 1 and 2, for the duration of the fifth 10-year inspection interval of the Inservice Inspection Program, which is effective from December 21, 2014, through December 20, 2024.

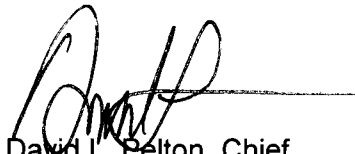
All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested and authorized by NRC staff remain applicable, including a third-party review by the Authorized Nuclear Inservice Inspector.

K. Davison

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If you have any questions, please contact Terry A. Beltz at 301-415-3049, or via e-mail at Terry.Beltz@nrc.gov.

Sincerely,

A handwritten signature in black ink, appearing to read 'D. Pelton', followed by a horizontal line.

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via ListServ



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

FOR REQUESTS 1-RR-5-3 AND 2-RR-5-3

REGARDING ALTERNATIVE REQUIREMENTS FOR BOLTING

AFFECTED BY BORATED WATER LEAKAGE

NORTHERN STATES POWER COMPANY – MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated September 15, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14258A073), Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, requested use of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2007 Edition through the 2008 Addenda, associated with the fifth 10-year interval for the Prairie Island Nuclear Generating Plant (Prairie Island), Units 1 and 2, Inservice Inspection (ISI) Program.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(a)(3)(i), the licensee submitted 1-RR-5-3, Revision 0, and 2-RR-5-3, Revision 0, associated with bolted connections affected by borated leakage at Prairie Island, Units 1 and 2, for use of two alternatives to the requirements of IWA-5250(a)(2) of ASME Code, Section XI, on the basis that the alternatives provide an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Inservice inspection of Class 1, 2, and 3 components is to be 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 Commission pursuant to 10 CFR 50.55a(g)(6)(i).

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 preservice examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Enclosure

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, which was incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, or the optional ASME Code cases listed in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," October 2014 (ADAMS Accession No. ML13339A689)..

By *Federal Register* notice 79 FR 65776, dated November 5, 2014, which became effective on December 5, 2014, the paragraphs headings in 10 CFR 50.55a were revised. Accordingly, relief requests that had been previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1) and relief requests that had been previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrate (1) the proposed alternative would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request the use of an alternative and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 Component(s) Affected

The components affected are ASME Code Class 1, 2, and 3, bolted connections in systems bolated for reactivity control in Examination Categories B-P, C-H, and D-B.

3.2 Applicable Code Edition and Addenda

The Code of record for the fifth 10-year ISI interval at Prairie Island, Units 1 and 2, for Code Class 1, 2, and 3, components is the 2007 Edition through the 2008 Addenda of the ASME Code.

3.3 Duration of Request

The licensee submitted these relief requests for the fifth 10-year ISI interval, which started on December 21, 2014, and is scheduled to end on December 20, 2024.

3.4 ASME Code Requirement

The applicable Code requirement is IWA-5250, "Correction Action." Leakage identified during the pressure tests performed in accordance with the subject Examination Categories is subject to corrective actions to meeting the requirements of IWA-5250. In IWA-5250(a)(2), it states that if leakage occurs at bolted connections in a system borated for the purpose of controlling reactivity, then one of the bolts shall be removed and a VT-3 examination performed and evaluation in accordance with IWA-3100. The bolt removed shall be the one closest to the source of leakage. If the removed bolt has evidence of degradation, then all the remaining bolting in the connection is required to be removed, and VT-3 examined and evaluated in accordance with IWA-3100.

As an alternative to code requirement of IWA-5250(a)(2), IWA-5251 contains provisions for correcting leakage and performing an evaluation of joint integrity in lieu of removing the bolt for the VT-3 examination.

3.5 Proposed Alternative

When a leak is identified at a bolted connection in systems borated for the purpose of controlling reactivity, the licensee proposes to either meet the requirements of IWA-5250(a)(2) or IWA-5251, or to stop the leak and address the cause of the leak using their corrective action program, and then replace all of the bolting at the connection in accordance with IWA-4000. A VT-3 of the removed bolting will not be performed, as all bolting will be discarded and not reused.

The licensee also proposes to substitute a VT-1 visual examination for the VT-3 visual examination required when implementing IWA-5250(a)(2).

Code Case N-775, "Alternative Requirements for Bolting Affected by Borated Water Leakage," of the ASME Code, Section XI was approved by ASME on June 24, 2010, and published in Supplement 2 of the Nuclear Code Case Book. This code case provides an alternative to IWA-5250(a)(2). The code case requires the following to be completed:

1. Corrective action shall be taken to stop the leak. The cause of the leakage shall be addressed in accordance with the Owner's corrective action program.
2. All pressure retaining bolting at the leaking connection shall be replaced in accordance with IWA-4000 (IWA-7000 in the 1989 Edition with the 1990 Addenda and earlier editions and addenda). A VT-3 visual examination of the removed bolting is not required.

The fundamental purpose of performing the VT-3 of the bolt closest to the leak on systems borated for the purpose of controlling reactivity is to determine the condition of the remaining bolting, which may affect the integrity of the connection. For those systems that are borated, if there is a leak, the boric acid that forms on the pressure boundary material has been shown to

corrode and cause degradation. If all of the bolts are replaced, the integrity of the connection is maintained and therefore provides an acceptable alternative.

3.6 NRC Staff Evaluation

The current ASME Code requires a VT-3 examination of bolting, even if all of the bolting is to be replaced. The rationale is that bolting may provide insights into use of incorrect materials or problems associated with the connection itself. ASME Code Case N-775 would allow restoration of the connection through maintenance and replacement activities, and thus eliminate the requirement for a VT-3 examination.

The NRC staff has supported ASME Code Case N-775 in various ASME Code Committee meetings, and has already approved a previous Code Case N-566-2, "Corrective Action for Leakage Identified at Bolted Connections, Section XI, Division 1," which provided alternative requirements to IWA-5250(a)(2) and eliminated the VT-3 examination requirement. ASME Code Case N-775 requires the cause of the leak to be addressed by the Owner's corrective action program, and for replacement of all bolting in the leaking connection. The regulations in Appendix B of 10 CFR Part 50, requires the Owner to identify and correct failures, defective materials, and equipment and, in the case of significant conditions adverse to quality, assure the cause is determined and actions taken to preclude repetition. The staff considers replacement of all the bolting in the joint and the determination of the cause of any degradation will ensure continued integrity of the bolting connection.

This safety evaluation does not authorize the use of ASME Code Case N-775. However, under the conditions described by the licensee, the NRC staff finds that the licensee's proposed alternative is consistent with the established NRC position concerning the adequacy of replacing bolting without performing the Code-required VT-3 examination and provides reasonable assurance of structural and leak-tight integrity of the mechanical connection.

3.7 Summary

Based on the above evaluation, the NRC staff finds that the proposed alternatives in 1-RR-5-3 and 2-RR-5-3 will continue to provide an acceptable level of quality and safety, and that the regulatory requirements set forth in 10 CFR 50.55a(z)(1) have been met.

4.0 CONCLUSION

As set forth above, the NRC staff finds that the licensee adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), and is in compliance with the requirements of 10 CFR 50.55a with the authorization of the licensee's proposed alternative.

The NRC staff further finds that the alternative method proposed by the licensee in 1-RR-5-3 and 2-RR-5-3 will provide an acceptable level of quality and safety, and that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), and remains in compliance with the requirements of the ASME Code, Section XI.

Based on the above evaluation, the NRC staff authorizes the licensee's proposed alternatives contained in 1-RR-5-3 and 2-RR-5-3 for the fifth 10-year ISI interval at Prairie Island, Units 1 and 2, which commenced on December 21, 2014, and will end on December 20, 2024.

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

Principal Contributor: Keith Hoffman, NRR/DE/EPNB

Date: August 12, 2015

K. Davison

- 2 -

If you have any questions, please contact Terry A. Beltz at 301-415-3049, or via e-mail at Terry.Beltz@nrc.gov.

Sincerely,

/RA/

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

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Safety Evaluation

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