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-5-1 AND 2-RR-5-1 ASSOCIATED WITH THE FIFTH 10-YEAR INTERVAL FOR THE INSERVICE INSPECTION PROGRAM (TAC NOS. MF4833 AND MF4834)

Dear Mr. Davison:

By letter dated September 15, 2014, as supplemented by letter dated February 4, 2015, Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, submitted a request for relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for the Prairie Island Nuclear Generating Plant (Prairie Island), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(a)(3)(ii) (retitled paragraph 50.55a(z)(2) by *Federal Register* notice 79 FR 65776, dated November 5, 2014), the licensee submitted requests 1-RR-5-1, Revision 0, and 2-RR-5-1, Revision 0, pertaining to the system leakage test of the reactor pressure vessel head flange seal leak-off line piping.

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)(2), and that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff authorizes the use of 1-RR-5-1 and 2-RR-5-2 at Prairie Island, Units 1 and 2, for 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.

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

Sincerely,

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
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

REGARDING SYSTEM LEAKAGE TEST OF THE

REACTOR PRESSURE VESSEL HEAD FLANGE SEAL LEAK-OFF LINE PIPING

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), as supplemented by letter dated February 4, 2015 (ADAMS Accession Nos. ML15036A254), Northern States Power Company – Minnesota (NSPM, the licensee), doing business as Xcel Energy, requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 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)(ii), the licensee submitted requests 1-RR-5-1, Revision 0, and 2-RR-5-1, Revision 0, pertaining to the system leakage test of the reactor pressure vessel (RPV) head flange seal leak-off line piping 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 EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), the ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the 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.

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

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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 U.S. Nuclear Regulatory Commission (NRC) staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Components Affected

The components affected are ASME Code Class 1 pipes. In accordance with the ASME Code, Section XI, IWB-2500 (Table IWB-2500-1), these pipes are classified as Examination Category B-P, Item Number B15.20.

The pipes under consideration are the RPV head flange leak-off connection lines from the reactor vessel flange to the 3/8-inch reducers. The licensee identified these lines as Line Numbers 1-RC-9A and -9B for Prairie Island, Unit 1, and Line Numbers 1-2RC-9A and -9B for Unit 2. In its February 4, 2015, letter, the licensee provided information regarding the material of construction (American Society for Testing and Materials (ASTM) A-376 TP 304 stainless steel) of the lines.

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, is the 2007 Edition through 2008 Addenda of the ASME Code.

3.3 Duration of Relief Request

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

3.4 ASME Code Requirement

The ASME Code, Section XI, IWB-2500, Table IWB-2500-1, Examination Category B-P, Item Number 15.20, requires the system leakage test be conducted according to IWB-5220 and the associated VT-2 visual examinations according to IWA-5240 once at or near the end of the interval. In accordance with IWB-5221(a), the system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power. In accordance with IWB-5222(a), the pressure-retaining boundary during the system leakage test

shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity. In accordance with IWB-5222(b), the Class 1 pressure-retaining boundary, which is not pressurized when the system valves are in the position required for normal reactor startup, shall be pressurized and examined at or near the end of the inspection interval. This boundary may be tested in its entirety or in portions and testing may be performed during the testing of the boundary of IWB-5222(a).

3.5 Proposed Alternative

The licensee proposed an alternative to IWB-5222(b). The proposed alternative is to perform the VT-2 visual examinations of the accessible portions of the RPV flange leak-off lines each refueling outage when these lines are subjected to static pressure head of 10 pounds per square inch gauge (psig) during refueling operations. During refueling operations, the static pressure head of 10 psig is produced from the normal refueling water level above the reactor vessel flange when the reactor cavity is flooded.

3.6 Basis for Hardship

In its September 15, 2015, submittal, the licensee stated, in part, that

> The configuration of [the flange seal leak-off lines] precludes manual testing while the vessel head is removed because of the odd configuration of the vessel taps, combined with the small size of the tap and the high test pressure requirement (2235 psig minimum), which prevent the taps in the flange from being temporarily plugged. Failure of this seal could possibly cause ejection of the device used for plugging the vessel taps. Machining, installing and removing the plugs or pressure connections would require significant time at the vessel flange and would expose personnel to excessive dose. [In its February 4, 2015, letter, the licensee provided an estimated radiation dose per test for each unit, which is 4.5 roentgen equivalent man (rem).] The plug or pressure connection itself would also introduce a foreign material exclusion issue at the edge of the open reactor vessel. Plugging or installing a connection would require machining threads in each flange opening with a concern over chips that may become a foreign material threat for fuel integrity or in the lines themselves.
>
> The top head of the [RPV] contains two grooves that hold O-rings. The O-rings are held in place by a series of retainer clips. The retainer clips are contained in a recessed cavity in the top head. If the licensee performed a pressure test from the leak-off line side with the head on, the inner O-ring would be pressurized in a direction opposite to its normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring is made of thin silver plating material and could likely be damaged by this deformation into the recessed areas on the top head.
>
> The use of [an external pneumatic high test pressure requirement would create] an unnecessary safety risk for the inspectors and test engineers in the unlikely

event of a test failure due to the large amount of stored energy contained in air pressurized to 2235 psig.

3.7 Basis for Use

In its letter dated February 4, 2015, the licensee stated, in part, that

> The majority of the leak-off piping is not accessible as it is under the refueling cavity floor and behind the shielding wall. Some of the underfloor piping may be partially accessible from the refueling cavity sand plug covers; however, the sand plug covers are not normally removed. The accessible segments are insulated. The inaccessible segments will not be directly examined, but examined for indication of leakage at the penetrations as required by IWA 5241....

The licensee stated that the RPV head flange leakage detection system consists of two lines piped to a common temperature element. One line is piped from the space between the two concentric reactor vessel flange O-rings and the other is piped from outside the second O-ring. Leakage is indicated by a control room temperature alarm at 140 degrees Fahrenheit (°F). Manipulation of the associated valves allows the plant operators to determine if potential leakage is past the inner O-ring, or past both O-rings. The lines will only be pressurized in the event of a failure of the inner O-ring.

In the February 4, 2015, letter, the licensee stated, in part, that

> There is no known plant specific or fleet operating experience regarding the degradation of the subject piping due to known degradation mechanisms. No instances of the reactor pressure vessel flange leak-off line degradation were found through a search of the Institute for Nuclear Power Operations operating experience website.
>
> [Prairie Island, Units 1 and 2 have] a reactor coolant system (RCS) Leakage Monitoring Program that monitors both identified and unidentified leakage. Plant operators perform an RCS leak rate test on a daily basis that calculates the coolant inventory balance and records the run time for Containment Sump A, the containment humidity, and the counts of from R-11 and R-12 particulate and gaseous radioactivity monitors. Along with the RCS Leakage Monitoring Program, the plant Technical Specifications state that RCS operational leakage shall be limited to: no pressure boundary leakage, 1 gallon per minute (gpm) unidentified leakage, 10 gpm identified leakage and 150 gallons per day (gpd) primary to secondary leakage through any one steam generator.
>
> RCS leakage past the RPV flange O-rings to the [reactor coolant drain tank (RCDT)] would be considered identified leakage. However, through wall leakage of the RPV leak-off piping would be considered pressure boundary leakage and therefore the system would be considered inoperable.

3.8 NRC Staff Evaluation

The NRC staff has evaluated 1-RR-5-1 and 2-RR-5-1 pursuant to 10 CFR 50.55a(z)(2). The NRC staff focuses on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship.

3.8.1 Hardship

The NRC staff found that requiring the licensee to comply with IWB-5222(b) and conduct system leakage test of the RPV head flange seal leak-off lines piping at the RCS operating pressure would result in hardship. The basis for the hardship is as follows. To conduct the ASME Code-required system leakage test of the leak-off piping when the reactor head is removed during refueling, the licensee would have to modify the existing RPV head flange taps to install plugs and/or test connections to facilitate for pressurizing the piping by use of an external pump. The activities associated with installing the plugs and/or the test connections, pressurizing the piping to the RCS pressure and conducting the ASME Code-required system leakage test, and removing the plugs after completion of test would cause personnel to incur additional radiation dose, and could introduce foreign materials into the reactor pool as well as the lines. Pressurizing the test connections to the RCS operating pressure would create personnel safety hazards in the event of a leak or break in any of the test connections.

Pressurizing the lines to conduct the ASME Code-required system leakage test when the RPV head is installed would not be possible due to design and configuration of the RPV head flange taps and the inner O-ring. The inner O-ring is designed to withstand pressure in one direction only, pressurizing in the opposite direction could damage the inner O-ring, and even result in unsuccessful test. Therefore, the NRC staff determined that concerns from the Foreign Material Exclusion program and an as low as is reasonably achievable criteria constitute a hardship.

3.8.2 Test Pressure

In evaluating the licensee's proposed alternative, the NRC staff assessed whether it appeared that the licensee used the highest achievable test pressure to conduct system leakage testing and the manner in which the licensee adequately preformed the testing and the associated VT-2 visual examinations of the piping for leakage. The NRC staff found that the licensee will use the highest pressure that is obtainable without major modifications to existing configuration of the lines to test the RPV leak-off piping for leakage. Specifically, the licensee's proposed system leakage test will subject the piping to the test pressure (i.e., static pressure head of 10 psig) developed from the elevation of refueling water above the vessel flange during the refueling cavity flood-up, which eliminates a need for major design modifications to existing configurations of both the vessel flange and the leak-off lines. The licensee will perform the VT-2 visual examination of accessible, inaccessible, insulated, and non-insulated portions of the piping in accordance with IWA-5240 to detect any leakage if it originated from an existing flaw in the piping and its welded connections after maintaining the static test pressure. Therefore, the NRC staff found that the licensee's alternative system leakage test subjects the piping under consideration to a test pressure that is as high as reasonably achievable.

3.8.3 Safety Significance of Alternative Test Pressure

In addition to the analysis described above, the NRC staff evaluated the safety significance of performance of the system leakage test at an alternative reduced pressure. The NRC staff notes that the leak-off piping is made of stainless steel. The degradation mechanism could be fatigue and stress-corrosion cracking. However, fatigue crack is known to have relatively slow growth and field experience has shown that stress-corrosion cracking under normal operating conditions is not expected to be a problem. Significant degradation would likely be detected by the system leakage test performed under proposed maximum obtainable static pressure head.

The NRC staff notes that if in an unlikely event, this piping developed a through-wall flaw and a leak, the plant existing reactor coolant leakage detection systems will be able to identify the leakage during normal operation, and the licensee will take appropriate corrective actions in accordance with the plant technical specifications. Therefore, the NRC staff determined that based on the alternative system leakage testing that subject these piping to the maximum obtainable static pressure head and the performance of the ASME Code-required VT-2 visual examinations, it is reasonable to conclude that if significant service-induced degradation had occurred, evidence of it would be detected either by the examinations that the licensee performed or the RCS leakage detection systems.

3.9 Summary

Based on the above evaluation, the NRC staff finds that the proposed system leakage testing using the proposed test pressure is adequate to provide a reasonable assurance of structural integrity and leak tightness of the RPV flange seal leak-off lines piping.

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)(2), 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 provides reasonable assurance of structural integrity and leak tightness of the RPV head flange seal leak-off lines piping, and that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above evaluation, the NRC staff authorizes the licensee's proposed alternative contained in 1-RR-5-1 and 2-RR-5-1 for the fifth 10-year ISI interval at the 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: Ali Rezai, NRR/DE/EPNB

Date: August 12, 2015

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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