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

June 4, 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 – RELIEF REQUESTS (1-RR-5-7 AND 2-RR-5-7) FOR FIFTH 10-YEAR INTERVAL FOR THE INSERVICE INSPECTION PROGRAM (TAC NOS. MF4843 AND MF4844)

Dear Mr. Davison:

By letter dated September 15, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14258A124), as supplemented by letter February 4, 2015 (ADAMS Accession Nos. ML15036A252) and e-mail dated April 10, 2015 (ADAMS Accession No. ML15103A007), 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 examination frequency requirements of Code Case N-729-1, at the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i) (retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014), the licensee submitted relief requests 1-RR-5-7, Revision 0, and 2-RR-5-7, Revision 0, for the respective units to use proposed alternatives to the examination frequency as specified in ASME Code Case N-729-1 on the basis that the alternative examination provides an acceptable level of quality and safety.

The 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 remains in compliance with the requirements of the ASME Code, Section XI, ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D), for which relief was not requested. The NRC staff authorizes the use of relief requests 1-RR-5-7 and 2-RR-5-7 at PINGP, Units 1 and 2, respectively, for the duration up to December 20, 2020.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR RELIEF REQUESTS 1-RR-5-7 AND 2-RR-5-7

REGARDING INSPECTION OF REACTOR VESSEL CLOSURE HEAD NOZZLES

NORTHERN STATES POWER COMPANY – MINNESOTA

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

(TAC NOS. MF4843 AND MF4844)

1.0 INTRODUCTION

By letter dated September 15, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14258A124), as supplemented by letter February 4, 2015 (ADAMS Accession Nos. ML15036A252) and e-mail dated April 10, 2015 (ADAMS Accession No. ML15103A007), 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 examination frequency requirements of Code Case N-729-1, at 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), Section 50.55a(a)(3)(i), the licensee submitted relief requests 1-RR-5-7, Revision 0, and 2-RR-5-7, Revision 0, for the respective units to use proposed alternatives to the examination frequency as specified in ASME Code Case N-729-1 on the basis that the alternative examination provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The regulation in 10 CFR 50.55a(g)(6)(ii) states, in part, that the Commission may require a licensee to follow an augmented inservice inspection (ISI) program for systems and components for which the Commission deems necessary added assurance of structural reliability.
The regulation in 10 CFR 50.55a(g)(6)(ii)(D) states, in part, that all licensees of pressurized water reactors shall augment their ISI program with ASME Code Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section.

Enclosure

In this request, the licensee requested relief from the examination frequency required by Code Case N-729-1 and, therefore, has also requested relief from 10 CFR 50.55a(g)(6)(ii)(D).

The regulation in 10 CFR 50.55a(z) 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: (1) the proposed alternatives would provide an acceptable level of quality and safety, or (2) 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, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to authorize the proposed alternative requested by the licensee.

As stated above, the licensee requests authorization of a proposed alternative to the examination frequency as specified in ASME Code Case N-729-1 pursuant to 10 CFR 50.55a(a)(3)(i). The NRC reorganized 10 CFR 50.55a such that relief requests previously submitted pursuant to paragraph 50.55a(a)(3)(i) are retitled under the equivalent paragraph 50.55a(z)(1), as provided in the *Federal Register* (79 FR 65776, dated November 5, 2014).

The paragraph headings in 10 CFR 50.55a were changed by *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014. For example, 10 CFR 50.55a(a)(3)(i) is now 50.55a(z)(1), and 50.55a(a)(3)(ii) is now 50.55a(z)(2). The cross-reference tables which are cited in the *Federal Register* notice may be found at ADAMS Accession No. ML14015A191, and ADAMS Package Accession No. ML14211A050.

The fifth 10-year inspection interval of the ISI Program at the Prairie Island Nuclear Generating Plant, Units 1 and 2, commenced on December 21, 2014, and is currently scheduled to end on December 20, 2024.

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 ASME Code Components

The affected components are ASME Class 1, reactor vessel closure head (RVCH) penetration nozzles and partial penetration welds, which are fabricated from Inconel SB-167 (Alloy 690) UNS N06690. The nozzle J-groove welds are fabricated from ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152), 52/152 weld materials. The original Prairie Island RVCHs contained penetration nozzles which were manufactured with Alloys 600/82/182 materials, and were subsequently replaced with new RVCHs using Alloys 690/52/152 material for the penetration nozzles in May 2006 (for Unit 1) and May 2005 (for Unit 2).

3.2 In-Service Inspection Interval

The original proposed duration of the alternative, a nominal 20-year inspection interval for both Prairie Island Units 1 and 2, would have occurred in the sixth ten-year ISI interval scheduled to being in December 2024.

The NRC staff requested additional clarification for the duration of the proposed alternative in an e-mail dated January 26, 2015 (ADAMS Accession No. ML15030A008). In the letter dated February 4, 2015 (ADAMS Accession Number ML ML15036A252), the licensee revised the duration of the 10 CFR 50.55a request to a nominal 15-year inspection interval based from the time that each reactor vessel head was placed in service.

The 15-year inspection interval would require examinations of either unit to be completed by December 20, 2020. Therefore, the corresponding reactor vessel head examinations are projected to be performed in the fall of 2019 (for Unit 2) and the fall of 2020 (for Unit 1). This revision to the inspection interval would place the proposed alternative into the ISI fifth interval which began December 21, 2014, and will end December 20, 2024.

3.3 Applicable Code Edition and Addenda

As stated above, Prairie Island, Units 1 and 2, commenced the fifth 10-year inspection interval of the ISI Program on December 21, 2014, and is required to follow the ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

3.4 Applicable Code Requirement

Section 50.55a(g)(6)(ii)(D) of 10 CFR requires, in part, that licensee augment their inservice inspection program in accordance with ASME Code Case N-729-1 subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D). Inspection Item B4.40 on Table 1 of ASME Code Case N-729-1 requires volumetric/surface examinations be performed within one inspection interval (nominally 10 calendar years) of the inservice date for a replaced RVCH. The required volumetric and surface examinations would thus have to be completed by the fall of 2015 (for Unit 2) and the fall of 2016 (for Unit 1) in order to fulfill the requirements of Code Case N-729-1.

3.5 Licensee's Proposed Alternative and Basis

3.5.1 Proposed Alternative

The licensee proposes to delay the next required inspections for a period of approximately 5 years. The licensee proposes to accomplish the inspections in accordance with ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) during scheduled outages in the fall of 2019 (for Unit 2) and the fall of 2020 (for Unit 1).

3.5.2 Licensee's Bases for Use of the Proposed Alternative

Use of the proposed alternative is focused on three topics of consideration. The first topic addresses the concept that the inspection interval in Code Case N-729-1 is based on primary water stress corrosion cracking (PWSCC) crack growth rates for Alloy 600/82/182. The second topic addresses bare metal visual examinations conducted on the licensee's replacement RVCHs. The third topic addresses a plant-specific factor of improvement analysis conducted by the licensee.

First Basis

In addressing its first basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in ASME Code Case N-729-1 for Alloy 600/82/182 are based on re-inspection years (RIY) equal to 2.25, and that this value is based on PWSCC crack growth rates as defined in the 75^{th} percentile curve contained in Materials Reliability Program (MRP)-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," and MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds." The licensee further asserts that the PWSCC crack growth rates of Alloy 690/52/152 are significantly lower than those of Alloy 600/82/182 and, therefore, merit a longer inspection interval. The licensee bases its assertion on: (1) the lack of cracking identified in other Alloy 690 components, such as steam generators and pressurizers, in the approximately 20 years that Alloy 690 has been in service in these components; (2) the failure to observe cracking during inspections already performed in replacement heads (13 of 40 replacement heads have been examined which includes heads which operate at higher temperatures than the heads under consideration); (3) the similarity of the inspected heads to the heads under consideration regarding configuration, manufacture, design and operating conditions; and (4) laboratory test data for Alloys 690/52/152 as contained in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles."

Second Basis

In addressing its second basis for use of the proposed alternative, the licensee stated that bare metal visual examinations were performed on the Prairie Island, Units 1 and 2, replacement RVCHs in 2011 and 2013, respectively, in accordance with ASME Code Case N-729-1, Table 1, item B4.30. The visual examinations were performed by VT-2 qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. The examinations revealed no indications of nozzle leakage (e.g. boric acid deposits) on the surface or near a nozzle penetration. The licensee also indicated that the examinations will be performed again during the Unit 2 refueling outage (2R30), scheduled to commence in September 2017.

The licensee stated in its September 14, 2014, submittal, that the bare metal visual examination for Unit 1 was scheduled for the upcoming refueling outage (1R29), scheduled to commence in October 2014. During the NRC staff's review of this relief request, the licensee performed a bare metal visual examination in accordance with Code Case N-729-1 on October 15, 2014. The NRC staff requested a discussion of the examination results, and the licensee responded in a supplemental letter dated February 4, 2015 (ADAMS Accession Number ML ML15036A252) that the examination showed no indications. No residual boric acid was identified and no items were noted that would be considered adverse to quality. The licensee stated that no alternative examination processes are proposed to those required by ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The visual examinations (VT-2) and acceptance criteria, as required by item B4.30 of Table 1 of ASME Code Case N-729-1, are not affected by this request and will continue to be performed on a frequency not to exceed every 5 calendar years.

Third Basis

In addressing its third basis for use of the proposed alternative, the licensee developed a plant-specific calculation of the required factor of improvement in the crack growth rate of Alloy 690/52/152 as compared to the crack growth rate of Alloy 600/82/182. In this calculation, the licensee used the actual temperature of the RVCHs and conservatively assumed calendar years to be equivalent to effective full power years. Based on this calculation, the licensee determined that an improvement factor of 7.8 was required to meet the proposed and desired inspection interval of 15 calendar years. The licensee proposed that because the required factor of improvement (7.8) was smaller than the factor of improvement of 10, which bounded most of the MRP-375 data for Alloy 690/52/152, use of a 7.8 factor of improvement would not result in a reduction in safety and provided further justification for an increased inspection frequency.

The licensee stated that their analysis showed significant margin to ensure that Alloy 690 nozzle base and Alloy 52/152 weld materials used in the replacement RVCHs at Prairie Island provide for a reactor coolant system pressure boundary where the potential for PWSCC has been shown by analysis and years of positive industry experience to be remote. As such, the licensee determined a technical basis sufficient to ensure public health and safety by extending the inspection frequency of the RVCH nozzle at Prairie Island, Units 1 and 2, from 10 calendar years to a new maximum of 15 calendar years.

3.6 Duration of Proposed Alternative

The licensee requested relief for the fifth 10-year inspection interval of the ISI Program for Prairie Island, Units 1 and 2. The fifth interval is effective for Units 1 and 2 from December 21, 2014, through December 20, 2024.

As discussed in Section 3.2 of this safety evaluation, the licensee indicated that it would revise the duration of its 1-RR-5-7 and 2-RR-5-7 to a nominal 15-year inspection interval from the time each reactor vessel head was placed in service. As such, the corresponding vessel head examinations are projected to be performed in the fall of 2019 (for Unit 2) and the fall of 2020 (for Unit 1). In no case would the reactor vessel head examinations of either unit be started after December 20, 2020.

3.7 NRC Staff's Evaluation

In evaluating the technical sufficiency of the licensee's proposed alternative, i.e., a one-time extension of the volumetric/surface examination interval contained in ASME Code Case N-729-1 from 10 years to not longer than 15 years, the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative. The NRC staff found that the technical basis included by the licensee provided sufficient information for the NRC staff to review the proposed alternative.

Due to concerns about PWSCC, many PWR plants in the United Stated and overseas have replaced RVCHs containing Alloy 600/182/82 nozzles with heads containing Alloy 690/152/52 nozzles. The inspection frequencies developed in Code Case N-729-1 for RVCH penetration nozzles using Alloy 600/182/82 were developed based, in part, on those material's crack growth rate equations documented in MRP-55 and MRP-115. The licensee's primary technical basis is to present crack growth rate data for the new more crack resistant materials, Alloy 690/152/52, and demonstrate an improvement factor (IF) of these materials compared to the older Alloy 600/82/182 materials. This IF would then provide the basis for the extension of the inservice inspection frequency requested by the licensee in its proposed alternative.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes that the licensee's use of MRP-375. This document, in part, summarizes numerous Alloy 690/152/52 crack growth rate data from various sources to develop IFs for the crack growth rate equations provided in MRP-55 and MRP-115. While the staff finds the licensee's assertions and/or interpretations to be reasonable, MRP-375 is not an NRC-approved document. The NRC staff does not have sufficient time or resources to validate all of the data used by this document, and the staff does not consider it appropriate to use all of the data from this document to review the licensee's relief request. A more detailed review of the data provided in MRP-375 will be performed by an international group of experts as part of an Alloy 690 Expert Panel, which is currently scheduled to complete its review in the 2016-2017 timeframe. In the interim, the NRC staff's review will rely upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data is documented in a data summary report and can be found under ADAMS Accession No. ML14322A587. The NRC's confirmatory research generally supports the contention that the crack growth rate of Alloy 690/52/152 is more crack resistant, but differs from the MRP-375 data in some respects.

The PNNL and ANL data summary report includes crack growth rate data up to approximately 20 percent cold work based on the observation of local strains in welds and weld dilution zone data. However, the NRC staff did not consider weld dilution zone data in its assessment. This is because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for Alloy 690/152/52 material. The high crack growth rates in weld dilution zones may be due to reduced chromium present in these areas. The staff chose to exclude the weld dilution zone data from this analysis due to the limited number of data points available, the variability in results, and due to the limited area of continuous weld dilution for flaws to grow through. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a J-groove weld along the low alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in very small areas of weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. Exclusion of these data may be reevaluated as additional data become available; a better understanding of the existing data is obtained; or if a longer extension of the inspection interval is requested. Therefore, the staff finds that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency, as requested by the licensee's proposed alternative, is not considered to be relevant for this specific relief request.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that the past bare metal visual examination on the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle/J-groove weld prior to the time the examination was conducted. The staff also finds that performance of future bare metal visual examinations in accordance with the code case is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the staff finds that the proposed alternative's frequency for bare metal visual examinations in conjunction with the new frequency of volumetric examinations is sufficient to provide reasonable assurance of the structural integrity of the RVCH.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff finds that the licensee's calculated improvement factor of 7.8, to support an extension of the ASME Code Case N-729-1 inspection frequency of 2.25 RIY to 15 calendar years, was acceptable by NRC staff calculation. The staff also finds that application of an IF of 7.8 to the 75th percentile curves in MRP-55 and MRP-115 bounded essentially all of the NRC data included in the PNNL and ANL data summary reports. Therefore, the staff finds this analysis supports the concept that a volumetric inspection interval for the RVCH of up to 15 calendar years does not pose a higher risk than that associated with an Alloy 600/182/82 RVCH inspected at an interval of 2.25 RIY. Therefore, the staff finds the licensee's technical basis to be acceptable.

3.8 Summary

Based on the above evaluation, the NRC staff finds that the proposed alternative of relief requests 1-RR-5-7 and 2-RR-5-7 will continue to provide an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff finds that the alternative method proposed by the licensee in relief requests 1-RR-5-7 and 2-RR-5-7 will provide an acceptable level of quality and safety for the examination frequency requirements of the RVCH. Accordingly, the staff concludes 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, ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D), for which relief was not requested.

Based on the above evaluation, the NRC staff authorizes the one-time use of relief requests 1-RR-5-7 and 2-RR-5-7 at the Prairie Island Nuclear Generating Plant, Units 1 and 2, respectively, for the duration up to December 20, 2020.

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 In-service Inspector.

Principal Contributor: Steven Vitto, NRR/DE/EPNB

Date: June 4, 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

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