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

February 25, 2019

Mr. Scott Sharp
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 FROM THE REQUIREMENTS OF THE ASME CODE (EPID: L-2018-LLR-0023)

Dear Mr. Sharp:

By letter dated March 6, 2018, Northern States Power Company (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, requirements at Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the alternative method proposed by the Northern States Power Company in alternative relief request Nos. 1-RR-5-9 and 2-RR-5-9 provides an acceptable level of quality and safety for the examination frequency requirements of the reactor pressure vessel closure heads at PINGP, Units 1 and 2. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes 1-RR-5-9 and 2-RR-5-9 until June 6, 2026, for PINGP, Unit 1, and June 10, 2025, for PINGP, Unit 2.

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

If you have any questions, please contact the Project Manager, Robert Kuntz at 301-415-3733 or via e-mail at Robert.Kuntz@nrc.gov.

Sincerely,

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NOS. 1-RR-5-9 AND 2-RR-5-9

REGARDING REACTOR PRESSURE VESSEL CLOSURE HEAD NOZZLES

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated March 6, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18065A583), Northern States Power Company (the licensee), submitted alternative relief request Nos. 1-RR-5-9 and 2-RR-5-9 which provides an alternative for the volumetric examination frequency requirements of the American Society of Mechanical Engineer's Boiler and Pressure Vessel (ASME) Code Case N-729-4 at Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. ASME Code Case N-729-4 requires volumetric or surface examinations of all primary water stress corrosion cracking (PWSCC) resistant reactor pressure vessel closure head (RVCH) nozzles every inservice inspection (ISI) interval and direct visual examinations of the upper head outer surface every third refueling outage or 5 years, whichever is less. The licensee proposed to increase the volumetric and surface examination interval from 10 years to approximately 20 years.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative would provide an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The ISI of ASME Code Class 1, 2, and 3 components is to be performed in accordance with ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," and applicable editions and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the U.S. Nuclear Regulatory Commission (NRC).

Pursuant to 10 CFR 50.55a(g)(6)(ii), the NRC may require the licensee to follow an augmented ISI program for systems and components for which the NRC deems that added assurance of structural reliability is necessary. The regulations in 10 CFR 50.55a(g)(6)(ii)(D) require, in part, "[a]ll licensees of pressurized water reactors must augment their ISI program with ASME Code Case N-729-4, subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) ..."

The regulations in 10 CFR 50.55a(z) state that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used when authorized by the NRC if the licensee demonstrates that: (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, and the NRC to authorize, the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Components Affected

The affected components Nos. 157-051 and 257-051 are ASME Class 1 pressurized-water reactor (PWR) RVCH nozzles and associated J-groove attachment partial-penetration welds fabricated with Alloy 690/52/152 materials. Each of these nozzles and associated welds are categorized as Item B4.40 in ASME Code Case N-729-4, Table 1, to identify the volumetric inspection frequency requirement for these components.

The licensee replaced the upper head for the PINGP, Units 1 and 2, reactor pressure vessel in May 2006 and May 2005, respectively.

3.2 Inservice Inspection Interval

PINGP, Units 1 and 2, are currently in the fifth 10-year ISI interval, which began December 21, 2014. The ASME Code of Record for the fifth 10-year ISI interval is the 2007 Edition with 2008 Addenda.

3.3 Code Requirement for Which Relief is Requested

Pursuant to 10 CFR 50.55a(g)(6)(ii)(D)(1), the NRC requires that licensees augment their ISI program in accordance with ASME Code Case N-729-4, subject to the conditions specified in paragraphs (2) through (4) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-4, Table 1, Inspection Item B4.40, requires volumetric or surface examinations be performed within one inspection interval (nominally 10 calendar years) for a replaced RVCH with PWSCC resistant nozzles and weld materials.

3.4 Proposed Alternative

The licensee requests to extend the frequency of the volumetric/surface examination of the RVCH specified in Table 1, Item B4.40, of ASME Code Case N-729-4 for a nominal 10 year period beyond the one inspection interval (nominally 10 calendar years) from installation of the PINGP, Units 1 and 2, replacement RVCHs. The end of the alternative date would be June 6, 2026, for PINGP Unit 1 and June 10, 2025, for PINPG Unit 2.

3.5 Licensee's Basis for Proposed Alternative

The licensee noted that the PINGP RVCH penetration nozzles and associated welds at both units are made from Alloys 690/52/152. The licensee explained that the Electric Power Research Institute (EPRI) published topical report, "[Materials Reliability Program] MRP-375, Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," dated February 2014 (ADAMS Accession No. ML14283A046), provided a technical justification to extend the volumetric/surface examination interval of the RVCH nozzle penetrations from 10 years to 20 years. The licensee also stated that the ASME Code Case Committee adopted the revised volumetric/surface examination of two inspection intervals (20 years) in ASME Code Case N-729-5. In summary, the licensee proposed to extend the inspection interval from once each interval (nominally 10 calendar years) by 10 additional years for a total of 20 calendar years based on plant service experience and factor of improvement (FOI) studies using laboratory data.

The licensee also provided a summary of EPRI report, MRP-386, "Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386)," dated December 2017. This report documented the activities of an expert panel to review crack growth rate data for Alloys 690/52/152. The report also provides a recommended FOI for the crack growth rates of these alloys compared to Alloys 600/82/182 materials. The report concluded that the lower bound FOI for the base metal Alloy 690 compared to Alloy 600 is 25, while the more realistic and recommended FOI is 38. Therefore, the report finds that under the same operating conditions, a similar hypothetical primary water stress corrosion crack in Alloy 690 material would grow on average 38 times slower than it would grow in an Alloy 600 material.

The licensee found that a minimum FOI of 7.8 was necessary for its replacement RVCH with Alloy 690/52/152 materials to support extending the inspection interval to 20 calendar years. The licensee stated that the PWSCC crack growth rates for Alloy 690/52/152 materials are significantly lower than those of Alloy 600/82/182 materials and, therefore, merit a much longer inspection interval than required by ASME Code Case N-729-4. In order to show that the inspection interval extension provides reasonable assurance of structural integrity, the licensee showed that a minimum FOI of 7.8 in the crack growth rate was acceptable by comparing the available crack growth rate curves of Alloy 600 materials to the available crack growth rate data for Alloy 690 materials. The licensee concluded that the proposed alternative revised volumetric/surface examination interval provides an acceptable level of quality and safety as conditioned by 10 CFR 50.55a(z)(1).

3.6. Duration of Proposed Alternative

The licensee proposed the alternative until June 6, 2026, for Unit 1, and June 10, 2025, for Unit 2.

3.7 NRC Staff Evaluation

In evaluating the technical sufficiency of the licensee's proposed alternative to defer the PINGP, Units 1 and 2, RVCH nozzle penetration and associated J-groove weld volumetric/surface examination interval to 20 calendar years, the NRC staff considered the licensee's basis for use of the proposed alternative in accordance with 10 CFR 50.55a(z)(1), on the basis that the alternative examination frequency provides an acceptable level of quality and safety.

The NRC staff notes that the inspection frequencies developed in Code Case N-729-4 for RVCH penetration nozzles made of Alloy 690/52/152 were developed based, in part, on a conservative assessment of the limited crack growth rate data and operating experience of these materials. The licensee's primary technical basis is that the available crack growth rate data is now sufficient to justify a longer inspection interval and demonstrate a sufficient FOI of these materials as compared to the Alloy 600/82/182 materials. Since the technical basis for the inspection frequency of nozzles and welds using Alloy 600/82/182 materials is based, in part, on the time necessary for a postulated flaw to cause leakage through the reactor coolant pressure boundary, a correlation between the crack growth rates of the Alloy 690 to Alloy 600 materials would allow a similar correlation to the time-to-leakage. Hence, a FOI between the two alloys would then provide the basis for equivalent safety of the extension of the ISI frequency requested by the licensee in its proposed alternative to the required inspection frequency of nozzles and welds using Alloy 600 materials.

The licensee calculated that it needed a FOI of 7.8 in order to have an equivalent safety factor from Alloy 690 to Alloy 600 materials. The NRC staff independently verified that the licensee's requested alternate inspection interval of 20 calendar years is reasonably bounded by the licensee's calculated FOI by using the parameters defined by ASME Code Case N-729-4 and using PINGP, Units 1 and 2, upper head estimated operating temperature.

The licensee then showed that the necessary FOI of 7.8 was less than the FOI obtained through crack growth rate testing of the Alloy 690 materials in laboratory testing. The licensee used several data sets to show that the FOI of 7.8 was bounded by the available data.

In evaluating the licensee's technical basis for the proposed alternative, the NRC staff notes that the licensee uses the crack growth rate data in MRP-375. MRP-375, in part, summarizes numerous Alloy 690/52/152 crack growth rate data from various sources to develop FOIs for the crack growth rate equations provided in MRP-55 and MRP-115 for Alloys 600/82/182. While the NRC staff finds the licensee's assertions and/or interpretations to be reasonable, MRP-375 is not an NRC-approved document. Additionally, the NRC staff has not validated all of the data reported in MRP-375. Therefore, the NRC staff does not consider it appropriate to use all of the crack growth data from MRP-375 in its review of the licensee's relief request. A more detailed review of the data provided in MRP-375 has been performed by an international group of experts as part of an Alloy 690 Expert Panel. This report entitled, "Materials Reliability Program: Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP 386)," has not been submitted to the NRC for formal generic review. The NRC staff has noted some limitations to the MRP-386 report for plant specific relief requests. The use of a restricted data set and the lack of a crack growth rate curve based on the material testing for Alloys 690/152/52 themselves prevents the full NRC endorsement of the report and its FOI. Therefore, the NRC finds that the licensee's specific FOI cannot be justified by these reports alone.

The NRC staff's review of the licensee's proposed alternative also relied upon Alloy 690/52/152 crack growth rate data from two NRC contractors, Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). The data is documented in the PNNL and ANL summary report found in ADAMS Accession No. ML14322A587. The majority of the data from PNNL and ANL for Alloy 690 test samples were generally consistent with the overall data presented in MRP-375, and also support the FOI value need to support the requested relief for the relevant conditions of the RVCHs at PINGP, Units 1 and 2. Therefore, the NRC staff finds that the licensee's proposed alternative is justified and bounded by the relevant available crack growth rate data included in the PNNL and ANL report, thus providing an equivalent or acceptable level of quality and safety.

The NRC staff finds that the licensee's analyses provided sufficient technical justification to support the proposed alternative of extending the volumetric/surface inspection interval for PINGP, Units 1 and 2, replacement RVCHs to 20 calendar years. The NRC staff finds that the proposed alternative does not pose a higher risk than the inspection frequency associated with a RVCH with Alloy 600/82/182 nozzles and associated J-groove welds that are inspected at intervals as specified in 10 CFR 50.55a(g)(6)(ii)(D). Hence, the NRC staff finds the licensee's technical basis to be acceptable. Therefore, based on the above evaluation, the NRC staff finds that the proposed alternative provides 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 has determined that the alternative method proposed by the licensee in alternative request Nos. 1-RR-5-9 and 2-RR-5-9 provides an acceptable level of quality and safety for the examination frequency requirements of the RVCHs at PINGP. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes 1-RR-5-9 and 2-RR-5-9 until June 6, 2026, for Unit 1, and June 10, 2025, for Unit 2.

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

Principal Contributor: J. Collins, NRR

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