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

March 27, 2024

Thomas A. Conboy
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Prairie Island Nuclear Generating Plant
1717 Wakonade Drive East
Welch, MN 55089

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - REVISION 1 TO REQUEST 1-RR-5-10 AND 2-RR-5-10 REGARDING REACTOR PRESSURE VESSEL WELDS AND NOZZLE WELDS (EPID L-2023-LLR-0052)

Dear Thomas Conboy**:**

By letter dated October 3, 2023, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (NSPM, the licensee) submitted a request to the Nuclear Regulatory Commission (NRC) for the use of an alternative 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 (Revision 1 of requests 1-RR-5-10 and 2-RR-5-10) to extend the fifth inservice inspection (ISI) interval at PINGP, Units 1 and 2, for Categories B-A and B-D examinations so that the ASME Code required examination in the fifth ISI interval can be performed in 2033 for Unit 1 and 2034 for Unit 2 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 NSPM has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the staff authorizes the use of alternative requests 1-RR-5-10, Revision 1, and 2-RR-5-10, Revision 1, at PINGP, Units 1 and 2, for the extended fifth ISI interval for ASME Categories B-A and B-D items until December 20, 2034.

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

If you have any questions, please contact the Project Manager, Brent Ballard, at 301-415-0680 or by e-mail to Brent.Ballard@nrc.gov.

Sincerely,

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2024.03.27 14:22:04 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVISION 1 TO RELIEF REQUEST NOS. 1-RR-5-10 AND 2-RR-5-10

REGARDING REACTOR PRESSURE VESSEL WELDS AND NOZZLE WELDS

NORTHERN STATES POWER COMPANY

PRARIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By letter dated October 3, 2023 (Agencywide Document Access and Management System (ADAMS) Accession No. ML23276B462), Northern States Power Company, a Minesota corporation, doing business as Xcel Energy (NSPM, the licensee), requested the use of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), section XI, table IWB-2500-1, for Categories B-A and B-D examinations of reactor pressure vessel (RPV) welds and nozzle welds 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 (Revision 1 of requests 1-RR-5-10 and 2-RR-5-10) to extend the fifth inservice inspection (ISI) interval at PINGP, Units 1 and 2, for Categories B-A and B-D examinations so that the ASME Code required examination in the fifth ISI interval can be performed in 2033 for Unit 1 and 2034 for Unit 2 on the basis that the alternative provides an acceptable level of quality and safety.

By letter dated June 13, 2019 (ML19164A166), NSPM submitted Revision 0 of 10 CFR 50.55a requests 1-RR-5-10 and 2-RR-5-10. By letter dated November 5, 2019 (ML19282A541), the U.S. Nuclear Regulatory Commission (NRC) authorized Revision 0 of the requests.

Recently, the licensee identified an error in Revision 0 of requests 1-RR-5-10 and 2-RR-5-10. Specifically, the licensee noted that an error with respect to the neutron fluence value reported for the nozzle shell-to-intermediate shell circumferential weld (W2) in tables 3a and 3b of Revision 0 that documents the calculation of through-wall cracking frequency (TWCF) for Unit 1 and Unit 2. This revised information comes from the correction of an error of the weld W2 location in the RPV shell and new reactor vessel surveillance capsule results.

The licensee noted that the error results in a change to the selection of the limiting circumferential weld but does not change the final calculated TWCF. In the previous submittal, tables 3a and 3b, the limiting circumferential weld was determined to be the intermediate shell-

Enclosure

to-lower shell circumferential weld (W3). However, when the neutron fluence information for weld W2 is corrected, weld W2 becomes the limiting circumferential weld for Unit 2. The licensee stated that despite the impact of the error on the calculation results, the TWCF contribution of the limiting circumferential weld ($TWCF_{95\text{-}CW}$) remains unchanged from the previous submittal. The calculation of $TWCF_{95\text{-}CW}$ includes a subtraction of a constant value from the reference temperature for the limiting circumferential weld ($RT_{MAX\text{-}CW}$). The licensee noted that because $RT_{MAX\text{-}CW}$ for both welds W2 and W3 is less than the constant, the subtraction results in a negative number and the calculation method sets the $TWCF_{95\text{-}CW}$ equal to zero. Therefore, the effect of the corrected neutron fluence on the limiting circumferential weld does not change the TWCF result.

2.0 REGULATORY EVALUATION

The regulations of 10 CFR 50.55a(g)(4), states, in part, that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in section XI of the ASME Code.

The regulations of 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," provide a methodology to calculate fracture toughness of reactor pressure vessel shell materials to ensure that the reactor vessel materials have sufficient fracture toughness to safeguard against potential pressurized thermal shock events.

The regulations of 10 CFR 50.55a(z) states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof 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 that: (1) the proposed alternatives 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 the use of and the NRC to authorize the proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 Background

The NRC staff's review of this proposed alternatives assesses the consistency of the licensee's proposal with topical report WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" (ML11306A084). Henceforth, WCAP-16168-NP-A, Revision 3, will be referred to as WCAP-A. WCAP-A provides a basis for the acceptability of the proposed inspection intervals for Category B-A and B-D components at U.S. pressurized-water reactors (PWRs) designed by Westinghouse, Combustion Engineering and Babcock and Wilcox (B&W) through the use of risk-informed analyses and probabilistic fracture mechanics for a pilot plant of each design. WCAP-A also contains the NRC staff's safety evaluation (SE) of the Westinghouse proposal. The SE finds the proposal acceptable for use based on consistency with the principles contained in Regulatory Guide (RG) 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ML023240437). However, the SE imposes a condition that requires licensees to provide plant-specific information in six areas to demonstrate the applicability of WCAP-A to the licensee's plant. The plant-specific information required by the condition is:

(1) Licensees must provide the 95th percentile total through-wall cracking frequency ($TWCF_{TOTAL}$) and its supporting material properties at the end of the proposed 20-year ISI interval. The 95th percentile $TWCF_{TOTAL}$ must be calculated using the methodology in NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)" (ML070860156), which is frequently referred to as "the NRC PTS Risk Study." The RT_{MAX-X} and the shift in the Charpy transition temperature produced by irradiation defined at the 30 ft-lb energy level, ΔT_{30}, must be calculated using the latest revision of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," or other NRC-approved methodology.

(2) Licensees must report whether the frequency of the limiting design basis transients during prior plant operation are less than the frequency of the design basis transients identified in the Pressurized-Water Reactor Owners Group (PWROG) fatigue analysis as significant contributors to fatigue crack growth.

(3) Licensees must report the results of prior ISI of RPV welds and the proposed schedule for the next 20-year ISI interval. Each licensee shall identify the years in which future inspections will be performed, and the dates provided must be within plus or minus one refueling cycle of the dates identified in the implementation plan provided to the NRC in PWROG letter OG-10-238 (ML11153A033).

(4) Licensees with B&W plants must (a) verify that the fatigue crack growth of 12 heat-up/cool-down transients per year that was used in the PWROG fatigue analysis bounds the fatigue crack growth for all of its design basis transients, and (b) identify the design bases transients that contribute to significant fatigue crack growth.

(5) Licensees with RPVs having forgings that are susceptible to underclad cracking and with RT_{MAX-FO} values exceeding 240 °F (degree Fahrenheit) must submit a plant-specific evaluation because the analyses performed in the WCAP-A are not applicable.

(6) Licensees seeking second or additional interval extensions shall provide the information and analyses requested in section (e) of 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

3.2 ASME Code Component Affected

The affected components are the subject plant RPV welds and full penetration nozzle welds. The following examination categories and item numbers from IWB-2500 and Table IWB-2500-1 of the ASME Code, section XI, are listed in alternative requests 1-RR-5-10, Revision 1, and 2-RR-5-10, Revision 1:

Exam Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Section

3.3 Applicable Code Edition and Addenda

The code of record is the ASME Code, section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition through 2008 Addenda. The licensee stated that the PINGP, Unit 1 and Unit 2, fifth 10-year ISI interval is scheduled to end on December 20, 2024.

3.4 Applicable Code Requirements

ASME Code, section XI, paragraph IWB-2411, "Inspection Program," requires volumetric examination of essentially 100 percent of the RPV pressure-retaining welds identified in table IWB-2500-1, once each 10-year interval.

3.5 Licensee's Proposed Alternative

In lieu of performing the required volumetric inspection for the subject Examination Categories B-A and B-D components during the fifth 10-year ISI interval, the licensee proposed to perform the required examination in 2033 for PINGP, Unit 1, and 2034 for PINGP, Unit 2, during the extended fifth ISI interval. The licensee stated that the proposed inspection dates for the two units is consistent with the schedule proposed in the PWROG letter OG-10-238.

3.6 Licensee's Basis for Alternative

The licensee stated that the alternative is based on a negligible change in risk, satisfying the risk criteria specified in RG 1.174. The licensee further states that the methodology used to conduct this analysis is based on the study defined in WCAP-A. This study focuses on risk assessments of materials within the beltline region of the RPV wall. Appendix A of the WCAP-A identifies the parameters to be compared between an applicant's plant and the appropriate pilot plant. These items include:

- Dominant PTS (pressurized thermal shock) Transients in the NRC PTS Risk Study
- TWCF
- Frequency and Severity of Design Basis Transients
- Cladding Layers (single/multiple).

Tables 1a and 1b of alternative requests 1-RR-5-10, Revision 1, and 2-RR-5-10, Revision 1, provide the above parameters for PINGP, Units 1 and 2, and the Westinghouse pilot plant. Based on this information, the licensee concludes that the parameters for Units 1 and 2 are bounded by the results of the Westinghouse pilot plant and states that Units 1 and 2 are qualified for the ISI interval extension.

For the most important parameter, TWCF, the licensee's calculated value is 4.64E-13 events per year for Unit 1, and 1.83E-13 events per year for Unit 2. These TWCF values are within the WCAP-A TWCF of 1.76E-08 events per year for the Westinghouse pilot plant. The details of the TWCF calculation are presented in tables 3a and 3b of the alternative requests.

Tables 2a and 2b of the submittal also contain inspection results for Units 1 and 2, showing that RPV examinations have been performed with satisfactory results.

3.7 Duration of Alternative

The licensee stated that in an NRC letter dated November 5, 2019 (ML19282A541), the NRC staff authorized requests Nos. 1-RR-5-10, Revision 0, and 2-RR-5-10, Revision 0, until December 20, 2034. The licensee further stated that the proposed revision (i.e., 1-RR-5-10, Revision 1, and 2-RR-5-10, Revision 1) does not request a change to this duration.

3.8 NRC Staff Evaluation

Since the WCAP-A methodology has already been accepted by the NRC staff, the current evaluation focused on the manner in which the licensee addresses the four critical parameters in table A-1 of WCAP-A, appendix A, and the six plant-specific information items specified in the NRC's SE for WCAP-A as discussed above.

The NRC staff reviewed the licensee's evaluation of the four critical parameters in section 5 of alternative requests 1-RR-5-10, Revision 1, and 2-RR-5-10, Revision 1. Regarding the PTS transients, the licensee used the NRC letter report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants" (ML042880482), as its plant-specific basis. The NRC staff finds this acceptable because the NRC's SE for WCAP-A concludes that based on this letter report the PTS transient characteristics are generally applicable for plants from the same reactor vendor. Regarding the cladding layers, the licensee reports "single layer" for both PINGP units. This is also acceptable because it is consistent with the Westinghouse pilot plant.

The remaining two critical parameters are among the six plant-specific information items discussed below.

3.8.1 Plant-Specific Information Item (1)

Plant-specific information item 1 addresses TWCFs and tables 3a and 3b of the submittal pertain to this item. As contained in the guidance provided in appendix A in WCAP-A, tables 3a and 3b of the submittal contain a summary of the input parameters for all Units 1 and 2, RPV materials and the resulting TWCFs for the controlling materials, respectively. The alternative proposed that the negligible changes in risk contained in tables 3a and 3b demonstrate that Units 1 and 2 are bounded by WCAP-A and are, therefore, acceptable. Specifically, tables 3a and 3b of the submittal provide updated input chemistry data, unirradiated nil-ductility transition reference temperature (RT_{NDT}), neutron fluence values for all RPV materials, and output shifts and TWCFs for controlling RPV materials of each unit.

By letter dated December 2, 2022 (ML22343A257), the licensee submitted a license amendment request (LAR) to revise the plant technical specifications related to pressure-temperature curves in response to a RPV weld location error identified in a neutron fluence evaluation which resulted in a higher neutron fluence value for the RPV weld than originally calculated. By letter dated January 17, 2024 (ML23356A003), the NRC staff approved the LAR. Therein, the NRC staff approved the revised neutron fluence values that were found to be in error and approved the fluence evaluation methodologies used in the LAR.

The NRC staff compared the information contained in tables 3a and 3b of the submittal with that in the license renewal application (LRA) for PINGP, Units 1 and 2, because these LRA values were accepted in NUREG-1960, "Safety Evaluation Report Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 and 2," August 2011 (ML11235A622), and are considered as the current licensing basis values for 54 effective full power years (EFPYs). The NRC staff found that the updated and approved fluence values and the updated chemistry data in the current submittal are still within a sizeable margin of the values approved in NUREG-1960. The NRC staff finds in the proposed alternative that despite the changes in fluence values and chemistry data for both units and factoring the change in limiting circumferential weld in Unit 2, the end result of the $TWCF_{95\text{-}CW}$ remains unchanged from the previous submittal.

Tables 3a and 3b of the submittal show that the calculated total TWCF is 4.64E-13 events per year for Unit 1 and 1.83E-13 for Unit 2. The licensee calculated the TWCF values using the WCAP-A methodology. Tables 3a and 3b used RG 1.99, Revision 2 (ML003740284), Position 1.1 (without surveillance data), or Position 2.1 (with surveillance data), to calculate RT_{MAX} (ΔT_{30} + unirradiated RT_{NDT} + 460 °F) for 54 EFPYs for all RPV beltline materials for Units 1 and 2. Using tables 3a and 3b input values, the NRC staff has verified that the licensee's calculated ΔT_{30} values, RT_{MAX} values, and the resulting TWCFs are acceptable.

The NRC staff determined that the TWCFs for the limiting RPV weld at Units 1 and 2 can support alternative requests 1-RR-5-10 and 2-RR-5-10 because they are several orders of magnitude lower than the value of 1.76E-08 for the Westinghouse pilot plant in the WCAP-A. Hence, the staff concludes that the licensee has addressed plant-specific information item (1) satisfactorily and that the embrittlement of the PINGP RPVs is within the envelope used in the Westinghouse pilot plant analysis and determined by the NRC to be acceptable in its review of WCAP-A.

3.8.2 Plant-Specific Information Item (2)

The NRC staff reviewed plant-specific information item (2) regarding the frequency of the limiting design basis transients. Tables 1a and 1b of the submittal state that the heatup/cooldown cycles per year for Units 1 and 2 are bounded by the heatup/cooldown cycles (7 per year) for the Westinghouse pilot plant. The NRC staff examined the heatup/cooldown design cycles for 60 years of operation in table 4.1-8 of PINGP Updated Final Safety Analysis Report (UFSAR) (ML18155A448) and verified that the above tables 1a and 1b statement are consistent with the PINGP UFSAR report. Therefore, the NRC staff found that the licensee has addressed plant-specific information item (2) satisfactorily.

3.8.3 Plant-Specific Information Item (3)

The NRC staff reviewed plant-specific information item (3) regarding the results of prior ISI of RPV welds and the proposed inspection schedule for the subject welds. Tables 2a and 2b in the submittal contain additional information pertaining to previous RPV inspections and the schedule for the future inspection. Specifically, tables 2a and 2b indicated that four 10-year ISIs have been performed for the subject RPV welds at Units 1 and 2. There were indications identified in welds and forgings in the RPV beltline region for both units during the last ISI. Some were accepted because they meet the acceptance criterion in table IWB-3510-1 of section XI of the ASME Code. The remaining indications are within the inner $1/10^{th}$ or one inch of the vessel thickness and need further evaluation. Tables 2a and 2b show that the number of flaws depthwise for welds and forgings are within the limits of the scaled maximum number of flaws depthwise based on the 10 CFR 50.61a table. Further, the response to the NRC's request for additional information dated September 16, 2019 (ML19259A020), for the original relief request clarified how the maximum number of flaws depthwise were scaled from the 10 CFR 50.61a Table, using plant-specific weld length and forging area. Therefore, the staff determined that the licensee has addressed the first part of plant-specific information item

(3) satisfactorily.

The licensee proposed to conduct the next RPV weld inspection in 2033 for Unit 1, and 2034 for Unit 2, before the ending date of the extended fifth 20-year ISI interval of December 20, 2034. Further, the NRC staff found that this date is consistent with the RPV inspection proposed in the PWROG letter OG-10-238 as stated in tables 2a and 2b of the submittal. Therefore, the NRC staff determines that the licensee has addressed the second part of plant-specific information item (3) satisfactorily.

3.8.4 Plant-Specific Information Items (4), (5), and (6)

The request did not address plant-specific information items (4), (5), and (6). The NRC staff examined the specifics in each of these three plant-specific information items and confirmed that these information requirements are not applicable to PINGP, Units 1 and 2.

3.8.5 Clarification on Duration of Alternative

The NRC staff recognizes that the request is applicable to the PINGP, Unit 1 and Unit 2, ISI program for the fifth and sixth 10-year ISI intervals. The NRC staff clarifies that once the Revision 1 of requests 1-RR-5-10 and 2-RR-5-10 are authorized, the ending date of the duration of alternative would be December 20, 2034. Since this date is the same as the ending date for the sixth 10-year ISI interval, the licensee's statement regarding the duration of alternative is acceptable.

3.8.6 Summary

The NRC staff has reviewed the licensee's submittal and determined that it has satisfied all applicable plant-specific information items specified in the SE for WCAP-A, Revision 3. For the risk-informed parameter, $TWCF_{95\text{-}TOTAL}$, the staff performed independent calculations to verify the input data and output results in tables 3a and 3b of the submittal. The difference between the licensee's and staff's calculated $TWCF_{95\text{-}TOTAL}$ is insignificant. With the above information, the NRC staff determined that the proposed alternative is based on the WCAP-A methodology and the $TWCF_{95\text{-}TOTAL}$ values in tables 3a and 3b of the submittal are bounded by the corresponding pilot plant parameter in the WCAP-A. Consequently, the licensee has demonstrated that the proposed alternative meets the guidance provided by RG 1.174, Revision 1, for risk-informed decisions and, therefore, will provide an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee has demonstrated that the proposed alternative provides an acceptable 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(z)(1). Therefore, the staff authorizes the use of alternative requests 1-RR-5-10, Revision 1, and 2-RR-5-10, Revision 1, at PINGP, Units 1 and 2, for the extended fifth ISI interval for ASME Code Categories B-A and B-D items until December 20, 2034.

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

Principal Contributors: E. Palmer, NRR
J. Tsao, NRR

Date: March 27, 2024

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 - REVISION 1 TO REQUEST 1-RR-5-10 AND 2-RR-5-10 REGARDING REACTOR PRESSURE VESSEL WELDS AND NOZZLE WELDS (EPID L-2023-LLR-0052) DATED: MARCH 27, 2024

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