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

December 21, 2020

Mr. Thomas A. Conboy
Site Vice President
Northern States Power Company - Minnesota
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – REQUEST FOR ALTERNATIVE FOR EXAMINATION OF REACTOR PRESSURE VESSEL THREADS IN FLANGE (EPID L-2020-LLR-0013)

Dear Mr. Conboy:

By letter dated January 31, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20031E432), Northern States Power Company, a Minnesota corporation doing business as Xcel Energy (NSPM, the licensee), requested relief from the requirements of the American Society of the Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the Monticello Nuclear Generating Plant (Monticello). The proposed alternative, RR-015, requests to eliminate the volumetric examination of the reactor pressure vessel (RPV) threads in flange during the fifth inservice inspection (ISI) interval at Monticello.

As set forth in the enclosed safety evaluation, the U.S. Nuclear Regulatory Commission (NRC) staff determines that proposed alternative RR-015 provides an acceptable level of quality and safety. Accordingly, the staff concludes that the regulatory requirements set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1) are adequately addressed. Therefore, the NRC staff authorizes proposed alternative RR-015 at Monticello, for remainder of the fifth ISI interval which is scheduled to end on May 31, 2022.

All other requirements of Section XI of the ASME Code for which an alternative 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 Robert Kuntz at 301-415-3733 or via e-mail at Robert.Kuntz@nrc.gov.

Sincerely,

Scott P. Wall

Digitally signed by Scott P. Wall
Date: 2020.12.21 13:51:20 -05'00'

Nancy L. Salgado, Chief (S. Wall for)
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure:
Safety Evaluation

cc: Listserv

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – REQUEST FOR ALTERNATIVE FOR EXAMINATION OF REACTOR PRESSURE VESSEL THREADS IN FLANGE (EPID L-2020-LLR-0013) DATED DECEMBER 21, 2020

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUESTS FOR ALTERNATIVE NO. RR-015

REACTOR PRESSURE VESSEL THREADS IN FLANGE

FIFTH TEN-YEAR INSERVICE INSPECTION INTERVAL

NORTHERN STATES POWER COMPANY – MINNESOTA

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated January 31, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20031E432), Northern States Power Company, a Minnesota corporation doing business as Xcel Energy (NSPM, the licensee), requested relief from the requirements of the American Society of the Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) Code for the Monticello Nuclear Generating Plant (Monticello). The licensee proposed alternative, RR-015, requests to eliminate the volumetric examination of the reactor pressure vessel (RPV) threads in flange during the fifth inservice inspection (ISI) interval at Monticello.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 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.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(4) state, in part, that 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 Section XI of the applicable editions and addenda of the ASME B&PV Code to the extent practical within the limitations of design, geometry, and materials of construction of the components. The threads in the RPV flange are categorized as ASME B&PV Code Class 1 components. Therefore, per 10 CFR 50.55a(g)(4), the ISI of these threads must be performed in accordance with Section XI of the applicable edition and addenda of the ASME B&PV Code.

Enclosure

The regulations in 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," state:

> "Alternatives to the requirements of paragraphs (b) through (h) of this section [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 applicant or licensee must demonstrate that:
>
> (1) *Acceptable Level of Quality and Safety*. The proposed alternative would provide an acceptable level of quality and safety; or
>
> (2) *Hardship without a Compensating Increase in Quality and Safety*. Compliance with the specified requirements of this section [50.55a] 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 the licensee may propose an alternative to ASME B&PV Code, Section XI, and the NRC staff has the regulatory authority to authorize the licensee's proposed alternative.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request for Alternative

3.1.1 ASME B&PV Code Components Affected

Proposed alternative RR-015 applies to the RPV threads in flange Examination Category B-G-1, Item No. B6.40, in Section XI of the ASME B&PV Code.

Examination Category	**Item No.**	**Examination Method**	**Description**	**Code Class**
B-G-1	B6.40	Volumetric	RPV Threads in Flange	1

3.1.2 Applicable ASME B&PV Code Edition and Addenda

For the fifth 10-year ISI interval at Monticello, the Code of Record for the inspection of ASME B&PV Code Class 1, 2, and 3 components is the 2007 Edition with the 2008 Addenda of the ASME B&PV Code, Section XI.

3.1.3 Applicable ASME B&PV Code Requirements and Proposed Alternative

The applicable inspection requirement for this component class is contained in Examination Category B-G-1, Item No. B6.40, which is listed in Table IWB-2500-1, "Examination Categories" of the ASME B&PV Code, Section XI. This item requires volumetric examination, every ISI interval, of all the threads in RPV flange stud holes, as indicated in Figure IWB 2500 12 "Closure Stud and Threads in Flange Stud Hole" of the ASME B&PV Code, Section XI. The proposed alternative would eliminate the ASME B&PV Code requirement to volumetrically examine the threads in the RPV flange stud holes.

3.1.4 Technical Basis for Proposed Alternative

The request stated that the technical basis for eliminating the RPV threads in flange volumetric examinations is provided in Electric Power Research Institute (EPRI) report No. 3002010354 (non-public), hereinafter referred to as the EPRI report). The request discussed the potential degradation mechanisms, bounding stress analysis, flaw tolerance evaluation, and operating experience that were included in the EPRI report and concludes that these justify the elimination of volumetric examination of RPV threads. The request provided a stress analysis specific to Monticello and compared the plant specific preload stress to the bounding preload stress provided in the EPRI report. The request demonstrated that the preload stress for Monticello (stated in the request to be 36,589 per square inch (psi) was bounded by the preload stress in the EPRI report (calculated as 42,338 psi. The request noted that the conclusion from the EPRI evaluation "was that these ASME Code Section XI examinations had not been identifying any service-induced degradation and the associated impact on worker exposure, personnel safety, critical path time, and additional time at reduced water inventory was not commensurate with performance."

Additionally, the request stated that detailed procedures are used for RPV disassembly and reassembly at Monticello to ensure protection and care of the studs and flange, including threads in flange every time the RPV head is removed. The request described the maintenance activities and inspections that will be performed on the RPV threads in flange and studs each time the RPV head is removed during the fifth ISI interval. The request stated that these controlled maintenance activities provide assurance that any degradation is mitigated and detected prior to returning the reactor to service.

3.1.5 Duration of Proposed Alternative

Proposed alternative RR-015 is requested for the duration of the remaining fifth ISI interval for Monticello. The fifth ISI interval for Monticello, began on September 1, 2012, and is scheduled to end on May 31, 2022.

3.2 NRC Staff Evaluation

The request relied on the EPRI report for the technical basis for the proposed alternative to eliminate examination of threads in the RPV flange. The NRC staff focused its evaluation of the proposed alternative on the deterministic stress analyses and flaw tolerance evaluation in the EPRI report, but also considered operating experience and potential degradation mechanisms. Each of these topics was discussed in the EPRI report and in the licensee's submittal.

The NRC notes that staff has evaluated proposed alternatives from other licensees based on a previous version of the EPRI report (dated March 2016, ADAMS Accession No. ML16221A068). For example, by letter dated January 26, 2017 (ADAMS Accession No. ML17006A109), the NRC staff authorized Southern Nuclear Operating Company, Inc. to use a similar alternative at Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1. The January 26, 2017, safety evaluation (SE) documents the NRC staff's evaluation of the 2016 EPRI report and concludes that EPRI's generic stress analysis and flaw tolerance evaluation are acceptable and the results can be used to support the elimination of the threads in RPV flange examination. To the extent possible, based on similarities between the 2016 and 2017 versions of the EPRI report, the NRC staff evaluated the request for Monticello consistent with previous SEs. However, where there are differences between the 2016 and 2017 versions of the EPRI report, such as in the calculation of KI used in the flaw tolerance evaluation, the NRC staff evaluation of this proposed alternative may not be consistent with previous evaluations performed for other requests.

3.2.1 Operating Experience

The EPRI report included the results of a survey of U.S. nuclear reactors taken in 2015 and early 2016 of the volumetric examination results for threads in the RPV flange (Table 3 of the submittal). The survey included 33 boiling-water reactor (BWR) units and 61 pressurized-water reactor (PWR) units. The total number of examinations for all 94 units was 10,662, with no reportable indications. The NRC staff finds that these survey results offer ample supporting evidence that the threads in the RPV flange are performing their function without a credible threat to the structural integrity of the RPV flange.

3.2.2 Potential Degradation Mechanisms

Section 5, "Evaluation of Potential Degradation Mechanisms," of the EPRI report provides an evaluation of the susceptibility of the threads in the RPV flange to the following degradation mechanisms: pitting, intergranular attack, corrosion, fatigue, stress corrosion cracking, crevice corrosion, velocity phenomena, de alloying corrosion and general corrosion, stress relaxation, creep, mechanical wear and mechanical/thermal fatigue. The EPRI report concluded that the only potential degradation mechanisms applicable to the threads in the RPV flange are mechanical and thermal fatigue. To address the potential for mechanical or thermal fatigue, the licensee referred to the generic stress analysis and flaw tolerance analysis in the EPRI report.

The NRC finds that mechanical and thermal fatigue are the only potential degradation mechanisms for the threads in RPV flanges of Monticello. The other degradation mechanisms listed in the EPRI report (e.g., stress corrosion cracking and creep) are not credible degradation mechanisms for the threads in the RPV flange because they are not in contact with the reactor coolant, and they are not in the operating temperature range where metal creep can occur.

3.2.3 Stress Analysis

Section 6.1, "Stress Analysis," of the EPRI report describes the determination of stresses at the critical location in the threads in the RPV flange. These stresses were used as input into the flaw tolerance evaluation, which is discussed in Section 3.2.4 of this safety evaluation. The stress analysis was performed using a three-dimensional, symmetric finite element model (FEM) of a portion of the threads in the RPV flange, RPV shell immediately below the flange, and a symmetric half of an RPV stud. Geometric parameters, such as number of RPV studs, stud diameter, RPV inside diameter, and flange thickness at the threads, were used to create the FEM. The loads applied in the FEM were the preload on the RPV studs, internal pressure, and thermal loads due to heatup and cooldown.

The NRC staff concluded that the generic EPRI stress analysis is acceptable and that the resulting stresses can be used in the flaw tolerance evaluation for Monticello.

Finite Element Model

As discussed in the EPRI report, bounding geometric parameters were used to create an FEM. The EPRI report states that the PWR design was used as a representative geometry for the FEM because of its higher design pressure and temperature. Tables 1 and 2 of the submittal showed the Monticello geometric parameters along with those used in the bounding analysis in the EPRI report. The NRC finds the selection of the PWR geometric parameters in the EPRI report acceptable because they bound the geometric parameters of Monticello.

Applied Loads

The request stated the preload stress for Monticello was calculated to be 36,589 psi. The NRC staff performed a confirmatory calculation and based on the request as confirmed by the NRC staff, the preload stress at Monticello is bounded by the preload stress in the EPRI report (42,338 psi).

The stress analysis in the EPRI report evaluated reactor heatup, but not a reactor cooldown. The NRC staff found that the use of cooldown instead of heatup would have the same effect on the fatigue crack growth calculation (evaluated in Section 3.2.4 of this SE) because it would produce the same stress range in the calculation. The EPRI thermal transient analysis assumed a 100 degrees Fahrenheit per hour heatup rate for the reactor coolant until the operating temperature was reached. The EPRI report heatup rate is equivalent to the maximum allowed reactor coolant heatup rate specified in the Monticello Updated Safety Analysis Report.

Based on the above, the NRC staff concludes that the applied loads used in the EPRI stress analysis are acceptable for Monticello.

3.2.4 Flaw Tolerance Evaluation

Section 6.2, "Flaw Tolerance Evaluation," of the EPRI report describes how the crack driving force, or stress intensity factor, K_I, due to the applied loads was determined. The flaw tolerance analysis, including the crack growth analysis, was based on the principles of linear elastic fracture mechanics. The stresses in the region of the root of the threads in the FEM were used to determine the critical location based on the largest tensile axial stress. A flaw was simulated by inserting crack tip elements in the FEM originating from this critical location, which enabled KI to be determined. The flaw was modeled around the critical thread and orientated such that the axial stresses act normal to the face of the flaw. Four flaw depths were modeled to determine the variation of KI with flaw depth, and the maximum applied KI was compared to the maximum value allowed by Appendix G of the ASME B&PV Code, Section XI. A flaw growth evaluation was then performed with a postulated initial flaw size at the root of the critical thread to show that the structural integrity of the threads in the RPV flange was not compromised for 80 years of plant life. In the generic EPRI flaw tolerance evaluation, the deepest flaw postulated was 0.77 a/t (depth-to-thickness ratio).

The generic EPRI flaw tolerance evaluation included simulations of a postulated flaw of four flaw depths inserted into the FEM to determine KI due to preload, internal pressure, and heat-up transient. The maximum applied KI around the postulated flaw was determined for each flaw depth for two load cases: (1) preload only and (2) preload with heat-up and pressure. The first case occurs during tensioning of the RPV bolts, and the second case occurs during reactor heatup to operating temperature and pressure. The EPRI report identified a maximum applied KI of 17.4 kilopounds per square inch square root inch (ksi√in) for the preload only and 19.8 ksi√in for preload with heat-up pressure. In the EPRI report, the maximum applied KI corresponded to a flaw depth of 0.29 a/t.

Although the EPRI report identified two load cases as stated above, the licensee stated the preload only case was limiting for Monticello, and derived a corresponding value of fracture toughness, K_{IC}, value of 89.6 ksi√in based on Paragraph A-4200 of the ASME B&PV Code, Section XI, Appendix A. The request analysis used a structural factor of 2, consistent with Appendix G of the ASME B&PV Code, Section XI, to derive an allowable KI value of 44.8 ksi√in at Monticello. The NRC staff independently verified the KIC values using the equations found in Appendices A and G of the ASME B&PV Code, Section XI. The NRC staff also verified that the preload only condition was limiting, based on the temperature dependence of the Appendix A equation for K_{IC}. Since the maximum applied KI value of 17.4 ksi√in is less than the allowable value of 44.8 ksi√in, the NRC staff concludes that the threads in RPV flange are reasonably flaw tolerant at preload conditions. Since the preload condition is bounding, the NRC staff concludes that the threads in RPV flange are also reasonably flaw tolerant at heatup and operating conditions.

The request, consistent with the EPRI report, stated that for a postulated flaw of 0.2 inches from the root of thread, the resulting crack growth was determined to be negligible due to the small delta K and the relatively low number of cycles associated with the transients evaluated. The request also stated that the allowable flaw size will not be reached, and the integrity of the component is not challenged for at least 80 years (original 40-year design life plus additional 40 years of plant life extension). In this evaluation and consistent with previous evaluations, the NRC staff concludes that this determination was acceptable.

3.2.5 Technical Conclusion

The NRC staff determined that the request has demonstrated that the deterministic stress analysis and flaw tolerance evaluation in the EPRI report are bounding for the threads in RPV flanges of Monticello. Therefore, the NRC staff determined that elimination of the ASME B&PV Code required examination of threads in the RPV flanges of Monticello is acceptable, because the request has provided reasonable assurance of structural integrity of the threads in RPV flanges without this examination for the duration of the fifth 10 year ISI interval at Monticello.

4.0 CONCLUSION

As set forth above, the NRC staff determines that proposed alternative RR-015 provides an acceptable level of quality and safety. Accordingly, the staff concludes that the regulatory requirements set forth in 10 CFR 50.55a(z)(1) are adequately addressed. Therefore, the NRC staff authorizes proposed alternative RR-015 at Monticello, for remainder of the fifth ISI interval which is scheduled to end on May 31, 2022.

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.

Principle Contributor: Joel Jenkins, NRR

Date: December 21, 2020