December 11, 2018

Mr. Joel P. Gebbie
Senior Vice President and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – PROPOSED ALTERNATIVE REQUEST FOR ELIMINATION OF THE REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION (EPID L-2018-LLR-0084)

Dear Mr. Gebbie:

By application dated June 14, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18169A148), Indiana Michigan Power Company (the licensee) submitted a request in accordance with Paragraph 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR) for a proposed alternative to the requirements of 10 CFR 50.55a, "Codes and standards," for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. By letter dated October 2, 2018 (ADAMS Accession No. ML18277A154), the licensee provided supplemental information to support the NRC review. By letter dated November 20, 2018 (ADAMS Accession No. ML18331A163), the licensee provided additional information in response to NRC staff questions.

The proposed alternative would allow the licensee to eliminate the examination of threads in the reactor pressure vessel flange required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), for CNP, Units 1 and 2. Specifically, pursuant to 10 CFR 50.55a(z)(1), the licensee requested to use the alternative on the basis that it will provide an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's proposed alternative and concludes, as set forth in the enclosed safety evaluation (SE), 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 the licensee use of the proposed alternative at CNP, Units 1 and 2 as requested in the licensee's application, for the remainder of the fourth 10-year inservice inspection (ISI) interval scheduled to end on February 29, 2020.

All other ASME Code 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 Mr. 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. 50-315 and 50-316

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE REQUEST (ISIR-4-07) FOR ELIMINATION OF THE REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION FOR

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

INDIANA MICHIGAN POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

EPID NO. L-2018-LLR-0084

1.0 INTRODUCTION

By letter dated June 14, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18169A148), as supplemented by letters dated October 2, 2018 and November 20, 2018 (ADAMS Accession Nos. ML18277A154 and ML18331A163, respectively), Indiana Michigan Power Company (the licensee) proposed an alternative from the examination requirements of the American Society of the Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The licensee's proposed alternative requests to eliminate the volumetric examination of the reactor pressure vessel (RPV) threads in flange during the fourth inservice inspection (ISI) interval at CNP, Units 1 and 2. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that it 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 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 Code Class 1 components. Therefore, per 10 CFR 50.55a(g)(4), ISI of these threads must be performed in accordance with Section XI of the applicable edition and addenda of the ASME Code.

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 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

3.1.1 ASME Code Components Affected

The proposed alternative applies to threads in the RPV flange subject to ASME Code, Section XI, Examination Category B-G-1, Item No. B6.40.

3.1.2 Applicable ASME Code Edition and Addenda

For the fourth 10-year ISI interval at CNP, Units 1 and 2, the Code of Record for the inspection of ASME Code Class 1, 2, and 3 components is the 2004 Edition of the ASME Code, Section XI. The fourth 10-year ISI interval for CNP, Units 1 and 2 is scheduled to end on February 29, 2020.

3.1.3 Applicable ASME Code Requirement

The licensee has requested an alternative to the examination requirements in Examination Category B-G-1, Item No. B6.40, Threads in Flange, which is listed in Table IWB-2500-1, "Examination Categories" of the ASME Code, Section XI. This item requires volumetric examination during every ISI interval on 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 Code, Section XI.

3.1.4 Licensee's Proposed Alternative and Basis for Use

The licensee is proposing to eliminate the examination of the threads in the RPV flange, as required by Examination Category B-G-1, Item No. B6.40, of the ASME Code, Section XI, for the fourth 10-year ISI interval at CNP, Units 1 and 2. The licensee's request is based on an evaluation by the Electric Power Research Institute (EPRI) documented in EPRI Technical Report No. 3002010354 (EPRI report), "Reactor Pressure Vessel (RPV) Threads in Flange Examination Requirements," dated December 2017 (ADAMS Accession No. ML18277A154). The licensee's submittal included information from the 2017 EPRI report regarding the generic stress analysis and flaw tolerance evaluation, with additional plant-specific information to demonstrate applicability of the EPRI report results. The submittal also included information from the 2017 EPRI report regarding operating experience and potential degradation mechanisms for the threads in the RPV flange.

3.2 NRC Staff's Evaluation

The licensee 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. Additionally, the staff 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 staff has evaluated proposed alternatives from other licensees based on a previous version of the EPRI report (ADAMS Accession No. ML16221A068), dated March 2016. 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, 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 licensee's proposed alternative for CNP, Units 1 and 2 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 the applied stress intensity factor (K_I) [as measured by kilopounds square root inch ($ksi\sqrt{in}$)], used in the flaw tolerance evaluation. The NRC staff evaluation of this proposed alternative may not be consistent with previous evaluations performed for other licensees due to the revised calculations used in the EPRI report.

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 licensee's submittal dated June 14, 2018). 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, 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 CNP, Units 1 and 2. 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 SE. 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.

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. In Table 1 of the submittal, the licensee showed the CNP, Units 1 and 2 geometric parameters along with those used in the bounding analysis in the EPRI report. The NRC finds the selection of a PWR design acceptable for CNP, Units 1 and 2 because both units are PWRs. Additionally, the NRC staff finds the PWR geometric parameters in the EPRI report acceptable because they bound the geometric parameters of CNP, Units 1 and 2.

Applied Loads

With respect to preload stress, from Table 1 of the submittal dated June 14, 2018, the licensee listed geometric parameters of CNP, Units 1 and 2 and compared them to the bounding values used in the EPRI calculation of preload stress on the RPV studs. The NRC staff verified the geometric parameters the licensee provided using the CNP, Units 1 and 2 updated final safety analysis report (UFSAR) and the corresponding value of the calculated preload stress for each unit. The NRC staff determined that the 42,338 pounds per square inch (psi) preload stress used in the EPRI analysis bounds the calculated preload stress for CNP, Units 1 and 2.

The stress analysis in the EPRI report evaluated reactor heatup, but not a reactor cooldown. 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 heatup rate is acceptable because it is greater than or equal to the maximum allowed reactor coolant heatup rate specified in the CNP, Units 1 and 2 UFSAR.

Based on the above, the NRC staff concludes that the applied loads used in the EPRI stress analysis are acceptable for CNP, Units 1 and 2. The NRC staff concluded that the generic EPRI report stress analysis is acceptable and that the resulting stresses can be used in the licensee's flaw tolerance evaluation for CNP, Units 1 and 2.

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 K_I 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 K_I with flaw depth, and the maximum applied K_I was compared to the maximum value allowed by Appendix G of the ASME 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-wall thickness ratio).

The generic EPRI flaw tolerance evaluation included simulations of a postulated flaw of four flaw depths inserted into the FEM to determine K_I due to preload, internal pressure, and heat-up transient. The maximum applied K_I 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 K_I of 17.4 ksi$\sqrt{in}$ for the preload only and 19.8 ksi$\sqrt{in}$ for preload with heat-up pressure. In the EPRI report, the maximum applied K_I corresponded to a flaw depth of 0.29 a/t.

For preload only, the licensee relied on Appendix B of the 2017 EPRI report and the equations in Figure G-2210-1 of Appendix G of the ASME Code, Section XI, to derive a K_{IC} [lower bound fracture toughness at operating temperature] value of 53.9 ksi$\sqrt{in}$. The licensee used a structural factor of 2, consistent with Appendix G, to derive an allowable K_I value of 27.0 ksi$\sqrt{in}$ for preload only at CNP, Units 1 and 2. Since the maximum applied K_I value of 17.4 ksi$\sqrt{in}$ is less than the allowable value of 27.0 ksi$\sqrt{in}$, the NRC staff concludes that the threads in RPV flange are reasonably flaw tolerant at preload temperatures.

For preload with heat-up and pressure, the licensee relied on Appendix B of the 2017 EPRI report, and Figure G-2210-1 of Appendix G of the ASME Code, Section XI, to derive a K_{IC} value of 220 ksi$\sqrt{in}$. The licensee used a structural factor of 2, consistent with Appendix G, to derive an allowable K_I value of 110 ksi$\sqrt{in}$ for preload with heat-up and pressure at CNP, Units 1 and 2.

Since the maximum applied K_I value of 19.8 ksi$\sqrt{in}$ is less than the allowable value of 110 ksi$\sqrt{in}$, the NRC staff concludes that the threads in the RPV flange are reasonably flaw tolerant at operating temperatures.

The licensee, consistent with the EPRI report, stated that for a postulated flaw of 0.2 inches from the root of thread, the crack would grow by 0.005 inch over 80 years of reactor operation. In this evaluation and consistent with previous evaluations, the NRC staff concludes that this amount of crack growth is acceptable. Additionally, the NRC staff determined this crack growth length is bounding using the fatigue crack growth curves in Figure A-4300-1 in ASME Code, Section XI, Appendix A. The crack growth evaluation in the EPRI report also assumed 50 reactor heatup/cooldown cycles per year and 5 bolt preloads per year. The NRC staff confirmed that these assumptions are conservative for CNP, Units 1 and 2.

3.2.5 Technical Conclusion

The NRC staff determined that the licensee has demonstrated that the deterministic stress analysis and flaw tolerance evaluation in the EPRI report are bounding for the threads in RPV flanges of CNP, Units 1 and 2. Therefore, the NRC staff determined that elimination of the ASME Code required examination of threads in the RPV flanges of CNP, Units 1 and 2 is acceptable. The NRC staff have concluded the licensee has provided reasonable assurance of structural integrity of the threads in RPV flanges without this examination for the duration of the fourth 10-year ISI interval at CNP, Units 1 and 2.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee's 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 NRC staff authorizes the use of the proposed alternative in request number ISIR-4-07 for CNP, Units 1 and 2 for the duration of the fourth 10-year ISI interval scheduled to end on February 29, 2020.

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

Principal Contributor: Joel Jenkins

Date: December 11, 2018

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – PROPOSED ALTERNATIVE REQUEST FOR ELIMINATION OF THE REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION (EPID L-2018-LLR-0084) DATED DECEMBER 11, 2018

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