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

June 11, 2015

Mr. Lawrence J. Weber
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 – REQUEST FOR USE OF ALTERNATIVE ISIR 04-02 ASSOCIATED WITH REACTOR VESSEL CLOSURE HEAD VOLUMETRIC/SURFACE EXAMINATION FREQUENCY REQUIREMENTS FOR THE INSERVICE INSPECTION PROGRAM (TAC NOS. MF5606 AND MF5607)

Dear Mr. Weber:

By letter dated January 20, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15023A038), Indiana Michigan Power Company (the licensee) submitted inservice inspection request (ISIR) 04-02 to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Code Case N-729-1 at Donald C. Cook Nuclear Plant (CNP), Units 1 and 2.

Specifically, pursuant to paragraph 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. ASME Code Case N-729-1 specifies that reactor vessel welded Alloy 690 head penetration nozzles shall undergo volumetric/surface examinations on a frequency of all nozzles in a nominal 10-year inspection interval. The proposed alternative would allow deferral of the volumetric/surface examinations of each unit's replacement reactor vessel closure head (RVCH) for two fuel cycles beyond the nominal 10-year inservice inspection (ISI) interval.

By *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014, the paragraph headings in 10 CFR 50.55a were revised. Accordingly, requests that were previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1), and requests that were previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

The NRC staff concludes, as set forth in the enclosed safety evaluation, that the alternative method proposed by the licensee in ISIR 04-02 will provide an acceptable level of quality and safety for the examination frequency requirements of the RVCH. 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 one-time use of alternative ISIR 04-02 at CNP, Units 1 and 2 for the duration up to and including the

29th refueling outage for Unit 1, and the 25th refueling outage for Unit 2, that are scheduled to occur in 2019 during the fourth 10-year ISI interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Allison Dietrich at 301-415-2846, or via e-mail at Allison.Dietrich@nrc.gov.

Sincerely,

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure:
Safety Evaluation

cc: Distribution via ListServ

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

INSERVICE INSPECTION REQUEST 04-02

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated January 20, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15023A038), Indiana Michigan Power Company (the licensee) submitted inservice inspection request (ISIR) 04-02 to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 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," at Donald C. Cook Nuclear Plant (CNP), Units 1 and 2.

Specifically, pursuant to paragraph 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), 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)(6)(ii) state, in part, that "[t]he Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary."

The regulations in 10 CFR 50.55a(g)(6)(ii)(D) state, in part, that "[a]ll licensees of pressurized water reactors must augment their ISI program with ASME Code Case N-729-1, subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section."

By *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014, the paragraph headings in 10 CFR 50.55a were revised. Accordingly, requests that were previously covered by 10 CFR 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR 50.55a(z)(1), and requests that were previously covered by 10 CFR 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR 50.55a(z)(2).

Paragraph (a)(z) of 10 CFR 50.55a states, in part, 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 (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 concludes that regulatory authority exists for the licensee to request and the Commission to authorize the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Components Affected

The affected components are ASME Code 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 RVCHs, which contained penetration nozzles manufactured with Alloy 600/82/182 material, were replaced with new RVCHs containing penetration nozzles manufactured with Alloy 690/52/152 material. These replacements occurred during the refueling outages (RFOs) in fall 2006 for Unit 1 and in fall 2007 for Unit 2.

3.2 Inservice Inspection Interval

The proposed duration for this use of the alternative for Unit 1 is up to and including the Cycle 29 RFO that is scheduled to commence in spring 2019, and will occur in the fourth 10-year inservice inspection (ISI) interval. The proposed duration for this use of the alternative for Unit 2 is up to and including the Cycle 25 RFO that is scheduled to commence in fall 2019, and will occur in the fourth 10-year ISI interval.

3.3 ASME Code of Record

The Code of record for the fourth 10-year ISI interval is the ASME Code, Section XI, 2004 Edition with no Addenda.

3.4 ASME Code and/or Regulatory Requirements

Paragraph 50.55a(g)(6)(ii)(D) of 10 CFR requires, in part, that licensees shall augment their ISI 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). ASME Code Case N-729-1, Table 1, Inspection Item B4.40 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/surface examinations would thus have to be completed by fall 2016 for Unit 1 and by fall 2017 for Unit 2, in order to fulfill the requirements of N-729-1.

3.5 Proposed Alternative

The licensee proposes to delay the next required inspection for a period of approximately 2.4 years for Unit 1, and 1.9 years for Unit 2. The licensee proposes to accomplish the

inspection in accordance with ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) during RFO 29 for Unit 1 and RFO 25 for Unit 2, which will occur in 2019.

3.6 Licensee's Evaluation of Proposed Alternative

The licensee's request to use the proposed alternative is based primarily on three topics of consideration. The first topic addresses the concept that the inspection interval in ASME 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 a bare metal visual examination conducted on the licensee's replacement RVCHs in 2011 and 2012. The third topic addresses a plant-specific factor of improvement (FOI) analysis conducted by the licensee.

3.6.1 PWSCC Crack Growth Rates

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 equal to 2.25, and that this value is based on PWSCC crack growth rates. These PWSCC crack growth rates are provided in the 75th percentile curves contained in Electric Power Research Institute (EPRI) Materials Reliability Program (MRP)-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials," dated July 18, 2002 (non-proprietary version available at ADAMS Accession No. ML023010510), and MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," dated November 2004 (non-proprietary version available at ADAMS Accession No. ML051450555). 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. To support this assertion, the licensee provided the following evidence. Firstly, there has been a record of resistance to PWSCC in Alloy 690 components such as steam generators, pressurizers, and RVCHs during the approximate 25 years that Alloy 690 has been used in these components. Secondly, there has been no observed PWSCC during inspections already performed on 13 of the 40 replacement RVCHs currently operating in the United States, including RVCHs that operate at higher temperatures than the RVCHs under consideration. Thirdly, laboratory test data for Alloy 690/52/152 as contained in MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," dated February 2014 (publicly available at http://www.epri.com; Product ID 3002002441), shows a substantially improved resistance to PWSCC for Alloy 690 material and Alloy 52/152 weld materials.

3.6.2 Bare Metal Visual Examinations

Bare metal visual examinations were performed on the replacement RVCHs in 2011 for Unit 1, and in 2012 for Unit 2, in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. These visual examinations were performed by visual testing (VT)-2 qualified examiners on the outer surface of the RVCHs including the annulus area of the penetration nozzles. These examinations did not reveal any indications of nozzle leakage, such as boric acid deposits, on the surface or near nozzle penetrations. Also, the licensee stated that no alternative examination processes are proposed to those required by ASME Code Case N-729-1, Table 1, Item B4.30, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The VT-2 examinations 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.

3.6.3 FOI Analysis

The licensee performed a plant-specific calculation of the required FOI to support the requested extension periods. The FOI models the increased resistance of Alloy 690/52/152 compared to that of Alloy 600/82/182. In making this calculation, the licensee used the actual temperature of the head and conservatively assumed that calendar years were equal to effective full power years. Based on this calculation, the licensee determined that an FOI of 5.9 was required for both CNP units to meet the proposed and desired inspection interval of 12.4 calendar years for Unit 1 and 11.9 calendar years for Unit 2. The licensee then proposed that because the required FOI of 5.9 was smaller than the FOI of 10 that statistically bounded the MRP-375 data for Alloy 690/52/152, the use of an FOI of 5.9 would not result in a reduction in safety and was, therefore, justified.

The licensee concluded that the Alloy 690 nozzle base and Alloy 52/152 weld materials used in the replacement RVCHs have a high resistance to PWSCC, based on analysis and years of positive industry experience. As such, the licensee found the technical basis sufficient to extend the inspection frequency of the RVCH nozzles from a maximum of 10 years to a new maximum of 12.4 years for Unit 1 and 11.9 years for Unit 2.

3.7 <u>NRC Staff Evaluation</u>

In evaluating the technical sufficiency of the licensee's proposed one-time extension of the volumetric/surface examination interval contained in ASME Code Case N-729-1 from 10 years to not longer than 12.4 and 11.9 years, for CNP Units 1 and 2, respectively, the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative.

3.7.1 PWSCC Crack Growth Rates

Due to concerns about PWSCC, many pressurized-water reactor plants in the United States and overseas have replaced RVCHs containing Alloy 600/182/82 nozzles with RVCHs containing Alloy 690/152/52 nozzles. The inspection frequencies developed in ASME Code Case N-729-1 for RVCH penetration nozzles using Alloy 600/182/82 were developed based, in part, on the materials' 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 more crack-resistant materials, Alloy 690/152/52, and demonstrate an FOI of these materials versus the older Alloy 600/82/182 materials. This FOI would then provide the basis for the extension of the ISI frequency requested by the licensee in its proposed alternative.

The NRC staff's review relied 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 dated October 30, 2014, and is publicly available in ADAMS at Accession No. ML14322A587. The NRC's confirmatory research generally supports the assertion 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 the 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 the reduced chromium present in these areas. The NRC 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 does not appear that 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 NRC staff concludes 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 request.

3.7.2 Bare Metal Visual Examinations

The NRC staff concludes that referencing the past bare metal visual examinations on the RVCHs 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 NRC staff also concludes 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 NRC staff concludes 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.

3.7.3 FOI Analysis

The NRC staff concludes that the licensee's calculated FOI of 5.9, to support an extension of the ASME Code Case N-729-1 inspection frequency of 2.25 re-inspection years to 12.4 and 11.9 calendar years for CNP, Units 1 and 2, respectively, was acceptable by the NRC staff's calculation. The NRC staff also concludes that the application of an FOI of 5.9 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 report. Therefore, the NRC staff concludes that this analysis supports the assertion that volumetric inspection intervals for the CNP, Units 1 and 2 RVCHs of not more than 12.4 and 11.9 calendar years, respectively, does not pose a higher risk than that associated with an Alloy 600/182/82 RVCH inspected at intervals of 2.25 re-inspection years. Based on the above, the NRC staff concludes that the licensee's technical basis for the proposed alternative is acceptable and 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 RR ISIR 04-02 provides an acceptable level of quality and safety for the examination frequency requirements of the RVCHs. 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 one-time use of ISIR 04-02 at CNP, Units 1 and 2, for the duration up to and including RFO 29 for Unit 1, and the RFO 25 for Unit 2, that are scheduled to occur in 2019, during the fourth 10-year ISI interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Margaret T Audrain

Date: June 11, 2015

29th refueling outage for Unit 1, and the 25th refueling outage for Unit 2, that are scheduled to occur in 2019 during the fourth 10-year ISI interval.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third-party review by the Authorized Nuclear Inservice Inspector.

If you have any questions, please contact Allison Dietrich at 301-415-2846, or via e-mail at Allison.Dietrich@nrc.gov.

Sincerely,

/RA/

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure:
Safety Evaluation

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