June 5, 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, UNIT NO. 2 – REVIEW OF SUPPLEMENTAL INFORMATION REGARDING THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM (EPID NO. L-2017-LRO-0068)

Dear Mr. Gebbie:

By letter dated December 8, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17346A683), Indiana Michigan Power Company (I&M, the licensee) submitted supplemental information regarding the reactor vessel internals aging management program for the Donald C. Cook Nuclear Plant, Unit No. 2. The submittal consisted of a proprietary failure modes and effects analysis (FMEA) of the baffle-edge bolts submitted in accordance with Condition 3 of the U.S. Nuclear Regulatory Commission (NRC) staff's final safety evaluation (SE) of WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria, Methodology and Data Requirements" (ADAMS Accession No. ML16061A187).

The NRC staff has reviewed the subject submittal and determined that the licensee has complied with Condition 3 from the NRC staff's final SE of WCAP-17096-NP, Rev. 2, which states that the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of the applicability of generic FMEA analyses, if a generic FMEA is referenced, within 1 year after any inspection that detects relevant conditions as defined in MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (ADAMS Accession No. ML120170453), Tables 5-2 or 5-3.

Please feel free to contact me at (301) 415-3826 if you have any additional questions or concerns.

Sincerely,

Allison W. Dietrich, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-316

Enclosure:
Review of Supplemental Information

cc: ListServ

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

REVIEW OF SUPPLEMENTAL INFORMATION REGARDING

REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated December 8, 2017 (Reference 1), Indiana Michigan Power Company (I&M, the licensee) submitted supplemental information regarding the reactor vessel internals (RVI) aging management program for the Donald C. Cook Nuclear Plant (CNP), Unit No. 2. The submittal consisted of a proprietary failure modes and effects analysis (FMEA) of the baffle-edge bolts submitted in accordance with Condition 3 of the U.S. Nuclear Regulatory Commission (NRC) staff's final safety evaluation (SE) of WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria, Methodology and Data Requirements" (Reference 2).

2.0 BACKGROUND

Baffle-edge bolts are part of the baffle-former assembly. The baffle-former assembly consists of baffle plates, which closely follow the outline of the reactor core. The function of the baffle plates is mainly to concentrate and direct flow through the reactor core. The baffle plates are secured to horizontal support plates called formers by the baffle-former bolts. The formers are in turn secured to the core barrel by barrel-former bolts. The baffle-edge bolts' function is to secure the edges of adjacent baffle plates, thereby preventing gaps between the baffle plates. Gaps between adjacent baffle plates could allow fuel cladding damage due to coolant leakage through the gaps impinging on peripheral fuel assemblies (baffle jetting). Baffle-edge bolts are generally not credited in structural analyses of the baffle-former assembly (unlike baffle-former bolts), so degradation of the baffle-edge bolts is considered to be of lower safety significance than degradation of baffle-former bolts. More detail on the design and function of baffle-former bolts and baffle-edge bolts can be found in NRC letter dated October 20, 2016 (Reference 3). During the 2016 refueling outage, the licensee found 170 baffle-former bolts and 5 baffle-edge bolts were degraded during its RVI examinations (Reference 4) that were conducted in accordance with MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (Reference 5).

By letter dated September 8, 2016, the NRC staff approved the RVI Inspection Program for CNP, Unit Nos. 1 and 2, which follows the guidance of MRP-227-A, via a staff assessment (Reference 6). For baffle-edge bolts (a subcomponent of the baffle-former assembly), MRP-227-A, Table 4-3, specifies a baseline visual (VT-3) examination be performed between 20 and 40 effective full power years, and requires a follow-on exam every 10 years thereafter, of 100 percent of bolts and locking devices on high-fluence seams, which are accessible from the core side.

Enclosure

The examination is conducted to detect the aging effect of cracking due to either fatigue or irradiation assisted stress corrosion cracking that results in lost or broken locking devices, failed or missing bolts, or protrusion of bolt heads. MRP-227-A, Table 5-3, "Westinghouse plants examination acceptance and expansion criteria," indicates that the specific relevant conditions for the baffle edge bolts are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads. Therefore, the examination results of the baffle-edge bolts at CNP, Unit 2, did not meet the acceptance criteria of MRP-227.

The WCAP-17096-NP-A, Revision 2 (Reference 7), is a topical report approved by the NRC for engineering evaluations of RVI components that do not meet the acceptance criteria of MRP-227-A. For baffle-edge bolts, WCAP-17096-NP-A specifies that the observed degradation be evaluated using a FMEA. The details of the recommended analysis procedure are provided on pages E-39 through E-41 of Appendix E to WCAP-17096-NP-A, Revision 2.

The baffle-edge bolts are identified in the SE of WCAP-17096-NP-A, Revision 2, as a Condition 3 Group 1 RVI component; therefore, the plant-specific FMEA must be submitted to the NRC within 1 year after any inspection that detects relevant conditions as defined in MRP-227-A.

3.0 NRC STAFF REVIEW

In Enclosure 2 of its letter dated December 8, 2017, the licensee provided the sections relevant to baffle-edge bolts from WCAP-18131-P, Revision 1, "Background and Technical Basis Supporting Engineering Flaw Acceptance Criteria for D.C. Cook Unit 2 Reactor Vessel Internals MRP-227-A Primary and Expansion Components." The NRC staff notes that the information provided was from a pre-outage contingency report prepared for the licensee. The information provided describes the process for the baffle-edge bolt FMEA and the generic results of the FMEA, but does not take into account the specific "as-found" condition of the baffle-edge bolts at CNP, Unit 2. Under "Methodology: Analysis," in the procedure for analysis of baffle-edge bolt degradation in WCAP-17096-NP-A, Revision 2, it states, in part, that observation of relevant conditions in the baffle-edge bolts or locking bars would require a FMEA on the observed condition.

The NRC staff reviewed the licensee's evaluation described in Enclosure 2 to its December 8, 2017, letter, and finds that the process is consistent with the guidance of WCAP-17096-NP-A, Revision 2. However, since the information submitted by the licensee did not provide any results or disposition specific to the "as-found" condition at CNP, Unit 2, the NRC staff requested the licensee submit baffle-edge bolt FMEA considering the "as-found" condition (Reference 8).

By letter dated March 28, 2018 (Reference 9), the licensee provided the requested information. The licensee stated that the plant-specific FMEA documented in WCAP-18131-P, Revision 1, was prepared prior to the baffle-edge bolt inspection and included a full range of potential conditions that could be observed. The response discusses the as-found condition. Indications of degradation were observed in five baffle-edge bolts on Column 7 (joint between baffle plates 6 and 7) during the VT-3 examination. The licensee indicated that the indications were considered patterned/clustered because the affected bolts were all on Column 7 between former levels D and F. The licensee also stated that patterned/clustered baffle-edge bolt indications was an analyzed condition in WCAP-18131-P.

The licensee stated that the baffle-edge bolts are not considered structural components that need to meet requirements for safe shutdown, and that Westinghouse performed a loose parts assessment, which determined that if lockbars or pieces of edge bolts become loose, safe operation of the unit will not be affected. However, to mitigate the risk of fuel defects due to flow induced vibration, debris (from potentially broken bolts) and baffle jetting, the fuel assemblies at core locations R11 and P12 were armored with stainless steel pins in the areas adjacent to the baffle-edge bolt heads and the seam between the baffle plates to which the baffle-edge bolts were fastened.

The licensee also indicated that the baffle-edge bolts with indications, along with all the baffle-edge bolts in the area of the baffle assembly that had significant clustered baffle-former bolt failures, were re-examined using VT-3 during the Unit 2 Cycle 24 refueling outage which began on March 1, 2018. The examination concluded that the previously recorded indications on the five baffle-edge bolts were non-relevant indications and were likely due to shadowing. The licensee attributed the difference in the examination results between 2016 and 2018 to the use of a higher resolution camera in 2018.

4.0 CONCLUSION

Since the baffle-edge bolt indications were determined by the licensee to be non-relevant as a result of the second examination, a FMEA is no longer necessary for the baffle-edge bolts. The NRC staff finds that the licensee has complied with Condition 3 from the NRC staff's final SE of WCAP-17096-NP, Revision 2, which states that (for Group 1 RVI Component Items, including the Westinghouse Baffle-Former Assembly Baffle-Edge Bolts), the licensee shall submit the plant-specific or generic FMEA analysis and confirmation of the applicability of generic FMEA analyses, if a generic FMEA is referenced, within 1 year after any inspection that detects relevant conditions as defined in MRP-227-A, Tables 5-2 or 5-3.

5.0 REFERENCES

1. Donald C. Cook Nuclear Plant Unit 2, Supplemental Information Regarding the Reactor Vessel Internals Aging Management Program, December 8, 2017 (ADAMS Accession No. ML17346A683).

2. Letter from K. Hsueh, NRC to A. Demma, Electric Power Research Institute, "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements' (TAC No. ME4200)," dated May 3, 2016 (ADAMS Accession No. ML16061A187).

3. Degradation of Baffle-Former Bolts in Pressurized Water Reactors - Documentation of Integrated Risk-Informed Decision Making Process in Accordance with NRR Office Instruction LIC-504, October 20, 2016 (ADAMS Accession No. ML16225A341).

4. Donald C. Cook Nuclear Power Plant, Units 1 And 2 - NRC Integrated Inspection Report 05000315/2016004; 05000316/2016004; 05000315/2016501; 05000316/2016501, February 13, 2017 (ADAMS Accession No. ML17044A405).

5. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) 1022863 Final Report, December 2011 – Transmitted to NRC by MRP letter MRP-2011-036 dated January 9, 2012 (ADAMS Accession No. ML120170453).

6. Letter from D.J. Wrona, NRC to J.P. Gebbie, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Staff Assessment Regarding Program Plan for Aging Management for Reactor Vessel Internals (CAC Nos. MF0050 and MF0051)," dated September 8, 2016 (ADAMS Accession No. ML 16063A434).

7. WCAP-17096-NP-A, Rev 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 31, 2016 (ADAMS Accession No. ML16279A320).

8. 2018/02/09 NRR E-mail Capture - DC Cook, Unit 2 - Request for Additional Information Regarding Reactor Vessel Internals Aging Management, February 9, 2018 (ADAMS Accession No. ML18043A009).

9. Donald C. Cook Nuclear Plant Unit 2, Response to Request for Additional Information Regarding Supplemental Information Regarding the Reactor Vessel Internals Aging Management Program, March 28, 2018 (ADAMS Accession No. ML18092A084).

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