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Docket No.: 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 2
Response to Request for Additional Information Regarding Supplemental Information Regarding the Reactor Vessel Internals Aging Management Program

References:

1. Letter from J. P. Gebbie, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC), "Donald C. Cook Nuclear Plant Units 1 and 2, Transmittal of Reactor Vessel Internals Aging Management Program," dated October 1, 2012, Agencywide Documents Access and Management System (ADAMS) Accession No. ML12284A320.

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. 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. ML16063A434.

4. Letter from Q. S. Lies, I&M, to NRC, "Donald C. Cook Nuclear Plant Unit 2, Supplemental Information Regarding the Reactor Vessel Internals Aging Management Program," dated December 8, 2017, ADAMS Accession No. ML17346A683.

5. Electronic Mail from J. K. Rankin, NRC, to H. L. Levendosky, I&M, "DC Cook, Unit 2 – Request for Additional Information Regarding Reactor Vessel Internals Again Management Program (EPID L-2017-LRO-0068)," dated February 9, 2018, ADAMS Accession No. ML18043A009.

This letter provides the Response to a Request for Additional Information Regarding Supplemental Information for the Donald C. Cook Nuclear Plant (CNP) Unit 2, pertaining to the CNP Reactor Vessel Internals Aging Management Program.

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By Reference 1, Indiana Michigan Power Company (I&M) transmitted the CNP Reactor Vessel Internals Aging Management Program which implemented guidance from the Electric Power Research Institute provided in MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," (MRP-227-A). By Reference 2, the U. S. Nuclear Regulatory Commission (NRC) transmitted a Final Safety Evaluation of WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." Section 4 of Reference 2 requires licensees to submit plant-specific and/or generic analyses to the NRC within one (1) year after any inspection that detects relevant conditions as defined in MRP-227-A. By Reference 3, the NRC transmitted a Final Safety Evaluation of the CNP Reactor Vessel Internals Aging Management Program which approved the program. By Reference 4, I&M provided supplemental information regarding the Reactor Vessel Internals Aging Management Program pursuant to Section 4 of Reference 2. By Reference 5, the NRC requested additional information in order to complete its review of Reference 4.

Enclosure 1 to this letter is an Affirmation Statement. Enclosure 2 to this letter provides I&M's response to the follow-up RAI contained in Reference 5. Enclosure 3 to this letter provides WCAP-18133-NP, a Non-Proprietary version of Section 3.6 of WCAP-18131-P, which was submitted by Reference 4.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Q. Shane Lies
Site Vice President

DB/mll

Enclosures:
1. Affirmation
2. Response to Request for Additional Information Regarding Supplemental Information Regarding the Reactor Vessel Internals Aging Management Program
3. Westinghouse WCAP-18133-NP, Revision 0, "Inspection Flaw Acceptance Criteria for D.C. Cook Unit 2 Reactor Vessel Internals MRP-227-A Primary and Expansion Components," Section 3.6

c: R. J. Ancona - MPSC
MDEQ- RMD/RPS
NRC Resident Inspector
J. K. Rankin, NRC Washington, D.C.
K. S. West, NRC Region III
A. J. Williamson - AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2018-22

AFFIRMATION

I, Q. Shane Lies, being duly sworn, state that I am the Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Q. Shane Lies
Site Vice President

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2024
Acting in the County of Berrien

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 28 DAY OF March, 2018

Danielle Burgoyne
Notary Public

My Commission Expires 04-04-2024

Enclosure 2 to AEP-NRC-2018-22

Response to Request for Additional Information Regarding Supplemental Information Regarding the Reactor Vessel Internals Aging Management Program

By letter dated December 8, 2017, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17346A683), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, submitted supplemental information regarding the Reactor Vessel Internals Aging Management Program for CNP Unit 2.

The U. S. Nuclear Regulatory Commission (NRC) staff is currently reviewing the submittal, and has determined that additional information is needed in order to complete the review. The request for additional information (RAI) and I&M's response is provided below.

RAI-1

Condition 3 of the 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) requires licensees to submit to the NRC the plant specific failure modes and effects analysis (FMEA) within one year after the inspection that detects relevant conditions. In Enclosure 2 of the December 8, 2017, submittal, the licensee provided a proprietary FMEA of the baffle-edge bolts for CNP, Unit 2. Specifically, the licensee provided 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 methodology and analysis section for baffle-edge bolts as described in WCAP-17096-NP-A (ADAMS Accession No. ML16279A320) states, in part, "[o]bservation of relevant conditions in the baffle edge bolts or locking bars would require a FMEA on the observed condition."

The NRC staff notes that the information provided in the December 8, 2017, submittal describes the process for baffle-edge bolt FMEA and the generic results of the FMEA, but does not take into account the specific "observed condition" of the baffle-edge bolts at CNP, Unit No. 2 as described in WCAP-17096-NP-A. Please provide the results of the baffle-edge bolt FMEA considering the as-found condition at CNP, Unit No. 2.

I&M Response to RAI-1

The plant-specific failure modes and effects analysis (FMEA) documented in 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," was prepared prior to the baffle-edge bolt inspection and included a full range of potential conditions that could be observed. The observed condition of baffle-edge bolts at CNP Unit 2 was considered patterned/clustered which was an analyzed condition in the plant-specific FMEA documented in WCAP-18131-P. The condition was considered patterned/clustered since the five (5) baffle-edge bolts with relevant indications were located on the same column (Column 7 – see Figure 1) and were all between Former Level D and Former Level F (See Figure 2). Column 7 baffle-edge bolts fasten Plate 7 to Plate 6 as shown in Figure 1. The five baffle-edge bolts are locations Y, AB, AD, AE, and AF shown on Figure 2.

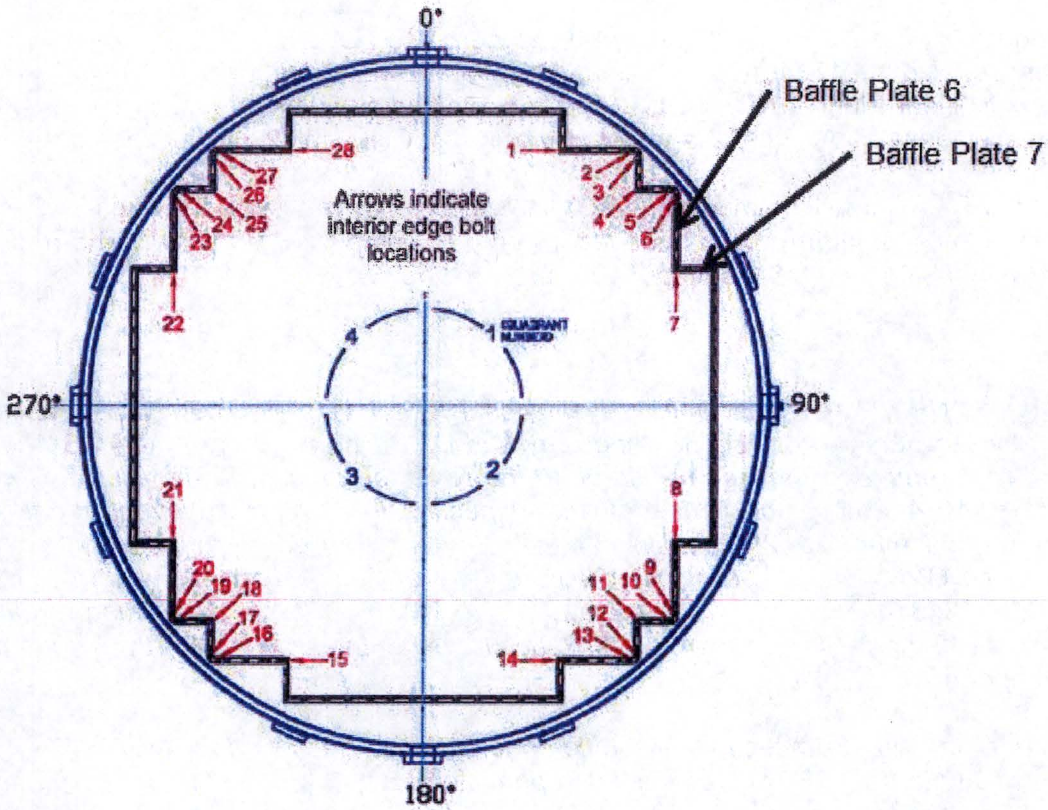

Figure 1 – Baffle-Former Assembly with Columns Labeled (looking down from above)

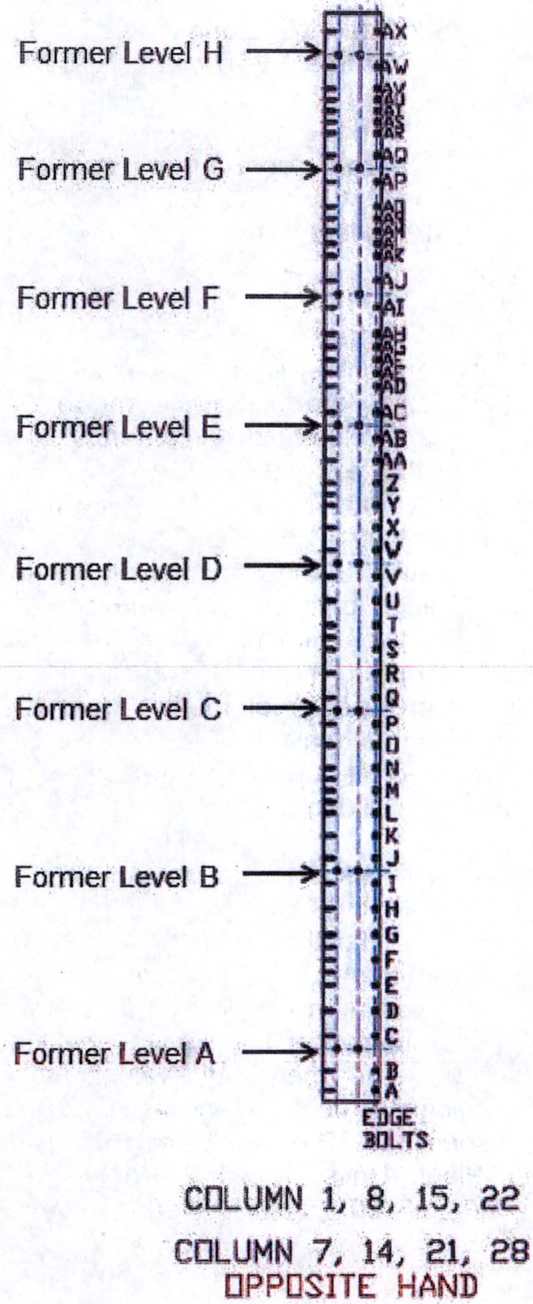

Figure 2 – Column 7 with Baffle-Edge Bolts and Former Levels Labeled

Baffle-edge bolts are not considered structural components that need to meet requirements for safe shutdown. Based on the decision to leave the baffle edge bolts as-is, informed by replacement feasibility and potential consequence of localized fuel damage due to jetting through vacant baffle-edge bolt counter bores, a loose parts assessment was prepared by Westinghouse. The loose parts assessment concluded that if the lockbars or pieces of edge bolts become loose parts during future operating cycles, safe operation of the unit would 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 apparent cause of the baffle-edge bolt degradation was that the large clustered region of failed baffle-former bolts surrounding Column 7 most likely led to increased stress on the baffle-edge bolts on Column 7 where the five baffle-edge bolts were found with cracked heads. Since the apparent cause of the edge bolt cracking was due to clustered baffle-former bolt failures, the baffle-edge bolts in the area of the baffle assembly that experienced significant baffle-former bolt clustering were inspected again during the current Unit 2 Cycle 24 refueling outage which began on March 1, 2018. The inspection included Columns 1 through 14 (shown in Figure 1). The inspection performed was a visual (VT-3) examination. The examination concluded that the previously recorded indications on the five baffle-edge bolts are non-relevant indications that were likely shadowing. The discrepancy between the two examinations can be explained due to factors such as the 2018 examination being performed using the Ahlberg Mega Rad camera which is a higher resolution camera. The Ahlberg camera provides greater latitude in camera angles and lighting as well as a higher resolution image which can be qualified for a VT-1 examination. In conclusion, the five baffle-edge bolts previously identified as containing relevant indications do not contain indications of cracking or any other age-related degradation.

During the Unit 2 Cycle 23 refueling outage, the NRC evaluated CNP's baffle-edge bolt inspection in accordance with Inspection Procedure 71111.08. The baffle-edge bolt inspections were included in the scope of this NRC review. After identification of the degraded baffle-edge bolts, an operability evaluation was prepared to determine if CNP Unit 2 was operable considering the identified degradation. The operability evaluation determined that the affected support structure was operable but degraded pending completion of additional inspections and corrective actions during the subsequent refueling outage. The operability evaluation was reviewed by the NRC inspection team. There were no findings or observations associated with the actions taken in response to the baffle-edge bolt inspection. The results of the inspection are summarized in "Donald C. Cook Nuclear Power Plant, Units 1 and 2—NRC Integrated Inspection Report 05000315/2016004; 05000316/2016004; 05000315/2016501; 05000316/2016501."

Enclosure 2 to AEP-NRC-2018-22

Westinghouse WCAP-18133-NP, Revision 0, “Inspection Flaw Acceptance Criteria for D.C. Cook Unit 2 Reactor Vessel Internals MRP-227-A Primary and Expansion Components,” Section 3.6