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

May 5, 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 – FOLLOW-UP REQUEST FOR ADDITIONAL INFORMATION CONCERNING THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM SUBMITTAL (TAC NOS. MF0050 AND MF0051)

Dear Mr. Weber:

By letter dated October 1, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12284A320), Indiana Michigan Power (I&M, the licensee) submitted an aging management program (AMP) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, reactor vessel internals (RVI). By letter dated June 6, 2014 (ADAMS Accession No. ML14135A320), the U.S. Nuclear Regulatory Commission (NRC) staff issued a request for additional information (RAI) for the CNP RVI AMP. The licensee submitted its response to the staff's RAI by letters dated July 30, 2014, September 4, 2014, and October 22, 2014 (ADAMS Accession Nos. ML14216A497, ML14253A316, and ML14316A449, respectively).

The NRC staff has reviewed the subject submittal, as supplemented, and determined that additional information is needed to complete the review, as described in the enclosed RAI. The draft RAI was sent to I&M via electronic mail on March 23, 2015. Clarification telephone conferences were held on April 8, 2015, and April 22, 2015. Based on our discussions, we understand that a response to the RAI will be provided by August 7, 2015.

Please feel free to contact me at (301) 415-2846 if you have any additional questions or concerns.

Sincerely,

AW Dietrich

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure:
Request for Additional Information

cc: Distribution via ListServ

FOLLOW-UP REQUEST FOR ADDITIONAL INFORMATION REGARDING

THE AGING MANAGEMENT PROGRAM FOR REACTOR VESSEL INTERNALS

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

TAC NOS. MF0050 AND MF0051

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the reactor vessel internals (RVI) aging management program (AMP) for Donald C. Cook Nuclear Plant, Units 1 and 2 (DC Cook 1 and 2) based on the MRP-227-A inspection guidelines (ADAMS Accession No. ML120170453) and the plant-specific action items (AIs) established in the safety evaluation for MRP-227-A. By letter dated June 6, 2014 (ADAMS Accession No. ML14135A320), the staff issued a request for additional information (RAI) for the DC Cook 1 and 2 RVI AMP. The licensee submitted its responses to the RAI by letters dated July 30, 2014, September 4, 2014, and October 22, 2014 (ADAMS Accession Nos. ML14216A497, ML14253A316, and ML14316A449, respectively).

The NRC staff has determined that additional information is required in order to complete its review of the licensee's responses to RAI-3, Part (b) and RAI-6, provided by letter dated October 22, 2014, as well as the responses to RAI-5 and RAI-7, provided by letter dated July 30, 2014. The follow-up RAI is provided below.

Follow-Up RAI-1 – Response to AI 1 and RAI-3, Part (b) Regarding Differences in Plant-Specific Fuel Design and Management Relative to MRP-227-A Assumptions

In its response to RAI-3, Part (b), and in accordance with AI 1, the licensee addressed the differences in plant-specific fuel design and management for DC Cook 1 and 2 relative to those that form the basis for the MRP-227-A guidelines. The licensee found that a projection of future operation for DC Cook 1 with the current fuel management strategy shows that the MRP-227-A applicability guideline for the core heat generation rate figure of merit (HGR-FOM), as established in Electric Power Research Institute Letter MRP 2013-025, Attachment 1, "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," dated October 14, 2013 (ADAMS Accession No. ML13322A454), will be exceeded in the future. Specifically, the licensee noted that the MRP applicability guidelines indicate that the HGR-FOM may not exceed 68 Watts per cubic centimeter (W/cm^3) for more than 2 years after the first 30 years of plant operation. The licensee determined that Unit 1 will exceed the 68 W/cm^3 applicability limit for more than 2 years after the first 30 years of plant operation.

Additional information regarding the fuel design and fuel management assessments for DC Cook 1 and 2 was provided by the licensee in the Westinghouse report PWROG-14049-P, "DC Cook Units 1 and 2 Summary Report for the Fuel Design / Fuel Management Assessments for Reactor Internals Aging Management MRP-227-A Applicability," dated October 13, 2014. The non-proprietary version of this report, PWROG-14049-NP, is available at ADAMS Accession No. ML14316A450.

(a) There is a discrepancy between the response to RAI-3, Part (b) in the third paragraph of Page 5 of the RAI response and PWROG-14049-NP regarding the operating period when the HGR-FOM exceeds the MRP-227-A applicability limit of 68 W/cm^3. Specifically, the response to RAI 3, Part (b) indicates that the DC Cook 1 HGR-FOM exceeded 68 W/cm^3 during Cycle 25 and will exceed this limit during Cycle 26, resulting in more than 2 years of operation outside the HGR-FOM limit by the end of Cycle 26. However, Section 1.1.1 of PWROG-14049-NP indicates a different operating cycle during which the HGR-FOM is exceeded at DC Cook 1, and it states that DC Cook 1 has used approximately 1.5 years of the allowable 2 years' time for exceeding the HGR-FOM limit.

    Resolve this discrepancy by stating the actual operating period(s) during which the DC Cook 1 HGR-FOM exceeds or will exceed the MRP-227-A applicability limit.

(b) Discuss any fuel management strategies that will be implemented for DC Cook 1 to ensure that the HGR-FOM will not exceed 68 W/cm^3 during future operation beyond Cycle 26. If there are no fuel management strategies that would bring the DC Cook 1 HGR-FOM to within this limit, then submit a plant-specific evaluation to demonstrate that the MRP-227-A guidelines and MRP-191 failure modes, effects, and criticality analysis (FMECA) inputs are applicable to DC Cook 1 relative to fuel design and fuel management, specifically taking into consideration the out-of-limit HGR-FOM and the projected operating period for the out-of-limit HGR-FOM.

Follow-Up RAI-2 – Response to AI 7 and RAI-6 Regarding CASS Components

Background

In accordance with AI 7 of the MRP-227-A SE, the licensee provided its evaluation of the cast austenitic stainless steel (CASS) RVI components for DC Cook 1 and 2 in Attachments 3 and 4, respectively, of Westinghouse LTR-RIAM-14-24, Revision (Rev.) 1, "Reports for D.C. Cook, Units 1 and 2 for PWROG PA-MSC-0983 Cafeteria Tasks 3, 4, and 5 Deliverables (Non-Proprietary)" – heretofore referred to as the AI 7 reports. These reports were included as Enclosure 6 of the licensee's final RAI response, dated October 22, 2014 (ADAMS Accession No. ML14316A449).

Additional information is required in order to demonstrate that the MRP-227-A guidelines are adequate for aging management of the CASS components identified in Table 1 below. Table 1 summarizes the licensee's determination of the components' susceptibility to thermal embrittlement (TE) and irradiation embrittlement (IE) in the AI 7 reports, the generic material and FMECA classification from MRP-191, and the additional information required for each CASS component. The licensee determined susceptibility to TE based on the screening criteria established for non-irradiated CASS in the U.S. NRC letter from C. Grimes, Office of Nuclear Reactor Regulation, dated May 19, 2000, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" (ADAMS Accession No. ML003717179, the C. Grimes Letter). The licensee determined susceptibility to IE based on the MRP-191 generic IE threshold for CASS components. Screening of CASS components for the synergistic effects of IE and TE using more recent IE and TE thresholds for irradiated CASS promulgated by staff in a June 2014 white paper (ADAMS Accession No. ML14163A112) has not yet been addressed for DC Cook 1 and 2.

Table 1 – DC Cook CASS RVI Components Requiring Further Evaluation

CASS Component	Plant-Specific TE Susceptibility Based on May 2000 C. Grimes Letter	Generic IE Susceptibility Based on MRP-191	Generic MRP-191 Material and FMECA[3] Group	Additional Information Needed
		DC Cook 1		
CRGT[1] Assembly – Guide Plates/Cards	YES[2]	NO	304 SS Group 3	Plant-Specific Evaluation and Inspection Criteria for Cracking
CRGT[1] Assembly – Housing Plates	YES[2]	NO	304 SS Group 0	Justification for AI 1 Resolution[4]
Upper Instrumentation Conduit and Supports – Brackets, Clamps, Terminal Blocks, Conduit Straps	YES[2]	NO	304 SS Group 0	Justification for AI 1 Resolution[4]
Lower Support Column Assemblies – Lower Support Column Bodies	YES[2]	YES	CF8 CASS Group 1	Plant-Specific Functionality Analysis or Inspection Plan
		DC Cook 2		
Upper Instrumentation Conduit and Supports – Brackets, Clamps, Terminal Blocks, Conduit Straps	YES[2]	NO	304 SS Group 0	Justification for AI 1 Resolution[4]
Upper Support Plate Assembly – Plate, Flange, and Upper Support Ring or Skirt	NO	NO	304 SS Group 2 – Ring or Skirt Group 0 – Plate and Flange	Justification for AI 1 Resolution[4]
Lower Support Column Assemblies – Lower Support Column Bodies	NO	YES	CF8 CASS Group 1	Plant-Specific Functionality Analysis or Inspection Plan[4]

Notes:

(1) Control rod guide tube (CRGT) components are assumed to be CASS by the licensee since documentation of constructional materials was not located.

(2) Susceptibility to TE is due to lack of CMTR data and assumed delta ferrite greater than 20 percent, as indicated in Tables 3.1-1 and 4.1-1 of the licensee's AI 7 reports in Attachments 3 and 4 of LTR-RIAM-14-24 .

(3) FMECA – Failure Modes, Effects, and Criticality Analysis. Generic FMECA results for Westinghouse RVI are provided in MRP-191, Table 6-5.

(4) The recent IE and TE screening criteria for irradiated CASS were provided in a June 2014 white paper. Synergistic effects of IE and TE based on these 2014 screening criteria should be considered for these components, as part of the justification, evaluation, or functionality analysis.

For those components identified in Table 1 as needing "Justification for AI 1 Resolution", the licensee's reports for AI 1 (Attachments 1 and 2 of LTR-RIAM-14-24) state that "[a] FMECA expert panel review applying the same methodology as used in the development of MRP-191 was conducted for these components ... [and] concluded that the aging management strategies of MRP-227-A were still applicable based on a consideration of the likelihood of failure and the likelihood of damage and the resulting classification of components." No additional details were provided regarding this determination.

Request Part (a) for Follow-Up RAI-2

For those confirmed CASS components designated in Table 1 as needing "Justification for AI 1 Resolution," provide justification for the determination that the MRP-227-A guidelines are still applicable to the above components based on the MRP-191 methodology, in accordance with AI 1. This justification should include the plant-specific screening results for all aging mechanisms, explanation of the likelihood of component failure, the likelihood of core damage, the resulting FMECA group for the components, the categorization and ranking of the components, and a discussion of how the final aging management strategy was determined. The justification must account for the additional embrittlement mechanisms (IE and/or TE) for CASS that were not generically considered for these components due to their treatment as non-CASS in MRP-191. This justification must take into consideration the potential synergistic effects of IE and TE for the CASS components. Screening criteria for IE and TE of irradiated CASS that are acceptable are detailed in a June 2014 white paper (ADAMS Accession No. ML14163A112).

Request Part (b) for Follow-Up RAI-2

The CRGT assembly guide plates/cards are generically analyzed as 304 stainless steel in MRP-191 and assigned to FMECA Group 3 in that report. MRP-227-A specifies that the Guide Plates/Cards are to be inspected as primary components for loss of material (wear) using the VT-3 visual examination method on the 10-year inservice inspection (ISI) interval. At DC Cook 1, the CRGT guide plates/cards are assumed to be CASS and susceptible to TE. Additionally, MRP-191 indicates that the guide cards screened in for cracking due to SCC (welds) and fatigue, for the generic guide card material. Given that the susceptibility of the CASS guide plates/cards to TE would make these components more likely to fail if cracks were present, provide an evaluation of the susceptibility of the guide cards to cracking. If the guide cards are susceptible to cracking, propose plant-specific inspection criteria for these components that would be sufficient for detecting cracking, or provide an evaluation of the components justifying that no additional inspections, other than the MRP-227-A inspection criteria, are necessary for these CASS components considering their susceptibility to cracking.

Request Part (c) for Follow-Up RAI-2

The DC Cook 1 lower support column (LSC) bodies are susceptible to TE and IE and therefore require an analysis to demonstrate their functionality during the period of extended operation, considering aging degradation of the LSC bodies due to TE and IE. The DC Cook 2 LSC bodies screened as not susceptible TE based on the criteria of the May 2000 C. Grimes Letter for non-irradiated CASS, but are susceptible to IE based on MRP-191. However, if the DC Cook 2 LSC bodies have delta ferrite greater than 15 percent, synergistic effects of both TE and IE are applicable, based on the more recent IE and TE thresholds for irradiated CASS established in the June 2014 white paper.

The licensee indicated in response to RAI-6(b) that the methodology for demonstrating the functionality of the LSC bodies is currently under development by the PWROG. An alternative to the functionality analysis could involve a plant-specific change to the inspection criteria for these expansion components. Any plant-specific inspection criteria should take into account the following: The LSC bodies are categorized as "Expansion" components in MRP-227-A for cracking due to IASCC and IE. However, the "Primary" linked component in MRP-227-A for the LSC bodies, the CRGT lower flange welds, is not a good predictor for either IASCC or IE because of the low neutron fluence exposure for the CRGT lower flange welds.

Submit the analysis to demonstrate the functionality of the DC Cook 1 and 2 LSC bodies during the period of extended operation, considering aging degradation due to the synergistic effects of IE and TE, or propose a plant-specific change to the inspection criteria for these components.

Follow-Up RAI-3 – CRGT Support Pins (Split Pins) Replacement Material and Subsequent VT-3 Inspections

In its response to RAI-5, the licensee provided a schedule and regulatory commitments related to the replacement of CRGT support pins (split pins) at DC Cook 1 and 2; the replacement of the split pins is scheduled to occur during the fall of 2017 at Unit 1 and the fall of 2016 at Unit 2.

(a) The current split pins are Alloy X-750 with a modified heat treatment. Indicate whether a more cracking-resistant type of material, such as 316 SS, will be selected for the replacement split pins.

(b) In its response to RAI-5 regarding AI 3, the licensee stated that no specific inspections are performed for the split pins at DC Cook 1 and 2. However, given the previous operating experience with split pin failures and the fact that these components are identified for plant-specific aging management in AI 3, some visual inspection of the replacement split pins may be necessary to verify that these components are maintaining their functionality during the period of extended operation. Therefore, address whether visual examinations of accessible portions of the spit pins will be performed during the period of extended operation (after split pin replacement) at DC Cook 1 and 2 concurrent with the 10-year ISI interval ASME Code, Section XI, Category B-N-3 inspections.

Follow-Up RAI-4 – Barrel-Former Bolts and Baffle-Former Bolts Aging Management

In its response to RAI-7, the licensee provided information concerning the root cause of the barrel-former and baffle-former bolt failures at DC Cook 1 and 2, respectively, and justification for the adequacy of the MRP-227-A guidelines for inspection of these components during the period of extended operation. Additional information is required concerning these bolt failures as discussed below:

DC Cook 1 Barrel-Former Bolt Failures:

(a) The licensee stated that that no single root cause was identified for the three barrel-former bolt failures discovered in 1994/1995. However, the licensee also stated that elevated stress near the thermal shield support block and bending stress on the bolts during normal steady-state operation were contributing causes and that SCC was not a factor. Therefore, the cause of failure appears to have been abnormal loading conditions on the bolts beyond their design criteria. Discuss whether any actions were taken to resolve the external causes of the abnormal loading on these bolts.

(b) The licensee stated that a total of three bolts were replaced with oversized bolts. Confirm whether these were the only three barrel-former bolts with indications of looseness or failure at DC Cook 1. Also, identify the original bolt material and the replacement bolt material (e.g. 316 SS, 347 SS, or other).

(c) The licensee stated that it was appropriate to return the system to its former monitoring requirements following bolt replacement in 1997 and that the barrel-former bolts are adequately managed by the existing monitoring and aging management programs already in place. Other than the MRP-227-A expansion component inspection criteria described in Appendix B of the DC Cook RVI AMP, list the existing monitoring and aging management programs currently in place that are applicable to the barrel-former bolts.

DC Cook 2 Baffle-Former Bolt Failures:

(d) The licensee stated that a total of 52 baffle-former bolts were replaced at DC Cook 2 in 2010, which includes the 18 failed bolts and the bolts in the adjacent rows and columns, with two locations left vacant. Elaborate on the reason for leaving two locations vacant, and briefly discuss whether any analysis was performed to ensure continued functionality of the baffle-former assembly with the two vacancies.

(e) State the total number of baffle-former bolts at DC Cook 2, and the total number of baffle-former bolts examined at symmetrical locations in the other three baffle plates in 2010. Also, state the method of examination that was used for the sampled bolts at the symmetrical locations.

(f) Indicate whether the cracked bolts conformed to any pattern related to neutron exposure.

(g) Identify the original baffle-former bolt material and the replacement bolt material (e.g. 316 SS, 347 SS, or other).

(h) Although the baffle-former bolt failures at D.C. Cook, Unit 2 appear to be limited to the large south baffle plate, it does not appear that the reason that 18 baffle-former bolts failed in this localized area is well understood. Therefore, the baffle-former bolts at D.C. Cook, Unit 2 may have greater susceptibility to cracking than those in other Westinghouse-design RVI. Provide the expected inspection date and projected EFPY for the initial UT examination of the D.C. Cook, Unit 2 baffle-former bolts. Justify the adequacy of this schedule considering the experience with baffle-former bolt failures at D.C. Cook, Unit 2. Describe any other actions that are planned prior to the initial UT inspection (for example additional visual examinations), to ensure the integrity of the baffle-former bolts.

Follow-Up RAI-5 – Schedule for Initial Primary Component Inspections

Table 4-3 of MRP-227-A (included in Appendix A of the DC Cook RVI AMP) specifies requirements for the initial (baseline) primary component inspections. For the baffle-former assembly components, the initial inspection schedule requirements are specified in terms of effective full power years (EFPY) of facility operation. For all other primary components, the initial inspection schedule requirements are specified in terms of the number of refueling cycles from the beginning of the license renewal period.

Provide the plant-specific schedule (calendar year and refueling outage) for the initial primary component inspections at DC Cook 1 and 2, including the baffle-former bolt baseline UT examinations.

Please feel free to contact me at (301) 415-2846 if you have any additional questions or concerns.

Sincerely,

/RA/

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure:
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