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

Donald C. Cook Nuclear Plant Units 1 and 2
RESPONSE TO FOLLOW-UP REQUEST FOR ADDITIONAL INFORMATION CONCERNING 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 T. J. Wengert, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program Submittal (TAC Nos. MF0050 and MF0051)," dated June 6, 2014, ADAMS Accession No. ML14135A320.

3) Letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2 – First Response to Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," dated July 30, 2014, ADAMS Accession No. ML14216A497.

4) Letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2 – Second Response to Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," dated September 4, 2014, ADAMS Accession No. ML14253A316.

5) Letter from J. P. Gebbie, I&M, to NRC, "Donald C. Cook Nuclear Plant Units 1 and 2 – Final Response to Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," dated October 22, 2014, ADAMS Accession No. ML14316A449.

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6) Letter from A. W. Dietrich, NRC, to L. J. Weber, I&M, "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)," dated May 5, 2015, ADAMS Accession No. ML15119A339.

This letter provides Indiana Michigan Power Company's (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, response to the Follow-Up Request for Additional Information (RAI) by the U. S. Nuclear Regulatory Commission (NRC) regarding CNP's Reactor Vessel Internals (RVI) Aging Management Program (AMP).

By Reference 1, I&M submitted the CNP RVI AMP. By Reference 2, the NRC transmitted RAIs regarding the program. References 3, 4, and 5 provided I&M's responses to Reference 2. By Reference 6, the NRC requested follow-up information for the RAI.

Enclosure 1 to this letter provides a response to Reference 6. Enclosure 2 provides Westinghouse Report PWROG-15066-NP, Revision 0, which provides further discussion to Reference 5, RAI 2. Enclosure 3 provides a commitment with a revised due date.

Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P. Gebbie
Site Vice President

DMB/ams

Enclosures:

1. Donald C. Cook Nuclear Plant Response to Follow-Up Request for Additional Information Regarding the Reactor Vessel Internals Aging Management Program
2. Pressurized Water Reactor Owners Group (PWROG)-15066-NP, Revision 0, Responses to Follow-Up NRC RAI 2 on the DC Cook Units 1 and 2 Reactor Internals Aging Management Program
3. Regulatory Commitment

c: A. W. Dietrich, NRC Washington, D.C.
J. T. King - MPSC
MDEQ- RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
A. J. Williamson – AEP Ft. Wayne, *w/o enclosures*

ENCLOSURE 1 TO AEP-NRC-2015-69

RESPONSE TO FOLLOW-UP REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

List of Acronyms:

ADAMS	Agencywide Documents Access and Management System
AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
CASS	Cast Austenitic Stainless Steel
CMTR	Certified Material Test Report
CNP	Donald C. Cook Nuclear Plant
CRGT	Control Rod Guide Tube
DMIMS	Digital Metal Impact Monitoring System
EFPY	Effective Full Power Years
EPRI	Electric Power Research Institute
FMECA	Failure Modes Effects and Criticality Analysis
HGR-FOM	Heat Generation Rate – Figure of Merit
I&M	Indiana Michigan Power
IE	Irradiation Embrittlement
ISI	In-Service Inspection
LSC	Lower Support Column
MRP	EPRI Materials Reliability Program
MSC	PWROG Materials Subcommittee
NEI	Nuclear Energy Institute
NRC	U. S. Nuclear Regulatory Commission
PWROG	Pressurized Water Reactor Owners Group
PWSCC	Primary Water Stress Corrosion Cracking
RAI	Request for Additional Information
RCS	Reactor Coolant System
RVI	Reactor Vessel Internals
SCC	Stress Corrosion Cracking
TE	Thermal Embrittlement
W/cm^3	Watts per Cubic Centimeter

By letter dated October 1, 2012 (ADAMS Accession No. ML12284A320), I&M, the licensee for CNP, submitted an AMP for CNP, Units 1 and 2, RVI to the NRC. By letter dated June 6, 2014 (ADAMS Accession No. ML14135A320), the NRC staff reviewed the submittal and requested additional information to complete its review. By letter dated July 30, 2014, the responses to RAI-1, RAI-5, and RAI-7 were provided to the NRC (ADAMS Accession No. ML14216A497). By letter dated September 4, 2014, the response to RAI-8 was provided to the NRC (ADAMS Accession Nos. ML14253A316, ML14253A317, and ML14253A318). By letter dated October 22, 2014, the responses to RAI-2, RAI-3, RAI-4, and RAI-6 were provided to the NRC (ADAMS Accession No. ML 14316A449). By letter dated May 5, 2015, the NRC staff requested follow-up additional information to complete its review. Responses to the follow-up RAI are provided in this enclosure.

Follow-Up RAI-1

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.*

Response to Follow-Up RAI-1

Issues related to fuel design and management for CNP Units 1 and 2 are discussed below.

(a) CNP Unit 1 exceeded the HGR-FOM screening limit established in the EPRI letter MRP 2013-025, "MRP-227-A Applicability Template Guideline," during Unit 1 Cycle 24 (U1C24). CNP Unit 1 is exceeding the HGR-FOM screening limit during the current operating Cycle 26 (U1C26). No other cycles of operation have exceeded the screening limit after the first 30 years of operation. The CNP Unit 1 HGR-FOM will exceed 68 W/cm^3 for more than two years after the first 30 years of plant operation during the current operating cycle.

Unit 1 Cycle 24 ran October 25, 2011, to March 27, 2013, for a total of 1.4 EFPY. Unit 1 Cycle 26 started October 23, 2014, and is scheduled to complete March 22, 2016. Unit 1 Cycle 26 is only expected to operate for 1.2 EFPY due to a forced maintenance outage.

(b) A fuel management strategy has been implemented for CNP Unit 1 and Unit 2 to ensure that the HGR-FOM will not exceed the 68 W/cm^3 screening limit beyond U1C26. The reactor core design procedure, EHI-4300, "Reactor Core Design," has been revised to include a requirement to observe screening limits outlined in MRP 2013-025, including the 68 W/cm^3 HGR-FOM screening limit, and to include a specific engineering stakeholder responsibility for reactor vessel internals fluence.

Follow-Up RAI-2

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 – CNP CASS RVI Components Requiring Further Evaluation from the Licensee

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
CNP 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
CNP 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.

<u>Request Part (a) for Follow-Up RAI-2</u>

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).

<u>Request Part (b) for Follow-Up RAI-2</u>

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 [to] 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.

Response to Follow-Up RAI-2

Items related to CASS issues are discussed below.

(a) The components in Follow-Up RAI-2, Table 1, "CNP CASS RVI Components Requiring Further Evaluation from the Licensee," identified as needing "Justification for AI [Action Item] 1 Resolution" are discussed below.

A plant-specific expert panel was held for these CASS components in CNP Units 1 and 2 using guidance presented in MRP-191, "Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design," Section 6, "Failure Modes, Effects and Criticality Analysis." The panel observed that these CASS and/or potentially CASS components screened in as potentially susceptible to TE, but did not change the generic MRP-191 classification of "Category A," low susceptibility components, for any of these items. Therefore, no changes were made to the program. The panel concluded that even with the addition of the additional degradation mechanism, the existing MRP-227-A aging management strategy is adequate and appropriate for these components with fabrication material different from the original MRP-191 assumptions.

The expert panel concluded that these components are potentially susceptible to TE, but receive irradiation exposure below the screening limit for IE for both the IE Screening process identified in MRP-191 and the NRC proposed IE screening criteria contained in "NRC position on Aging Management of CASS Reactor Vessel Internals Components," dated June 2014. Therefore, no changes are required for the RVI AMP.

Enclosure 2, PWROG-15066-NP, Revision 0, "Responses to Follow-Up NRC RAI 2 on the Donald C. Cook Nuclear Plant Units 1 and 2 Reactor Internals Aging Management Program," under the section titled "Response to Request Part (a)," provides further discussion regarding the components in Follow-Up RAI-2, Table 1, "CNP CASS RVI Components Requiring Further Evaluation from the Licensee," identified as needing "Justification for AI 1 Resolution."

(b) The CNP Unit 1 CRGT guide cards are potentially fabricated from CASS material. A plant-specific expert panel was held for this set of components using guidance presented in MRP-191, Section 6. The panel observed that this set of potentially CASS components screened in as potentially susceptible to TE, which results in these components considered as potentially susceptible to cracking. The panel concluded that even with the addition TE, the existing MRP-227-A aging management strategy is adequate and appropriate for these components with fabrication material different from the original MRP-191 assumptions.

The expert panel concluded that the CNP Unit 1 CRGT guide cards are potentially susceptible to TE, but receive irradiation exposure below the screening limit for IE for both the IE screening process identified in MRP-191 and the NRC proposed IE screening criteria contained in "NRC position on Aging Management of CASS Reactor Vessel Internals Components," dated June 2014.

Cracking effects are not a greater concern for CNP Unit 1 CRGT guide cards fabricated from CASS than they are for those fabricated from type 304 stainless steel as assumed in MRP-191. The stress, function, and geometry of the part remain the same regardless of material. Welds are similar between the chemically equivalent CASS CF8 and type 304 stainless steel. TE does not result in a complete loss of fracture toughness.

The expert panel conclusions regarding the potentially CASS CNP Unit 1 CRGT guide cards were based on the following items. Each item from the plant specific FMECA results was considered during the disposition process. The current program includes this set of components as a primary inspection item using visual inspection techniques which would detect gross failures. This set of components has redundancy from the perspective that failures at multiple cards within a single CRGT assembly would be required to prevent a control rod from inserting, and failure of one control rod to insert would not preclude safe shutdown. CNP procedures perform periodic monitoring of control rod functionality. Therefore, no changes are required for the RVI AMP.

Enclosure 2, PWROG-15066-NP, under the section titled "Response to Request Part (b)," provides further discussion on the CRGT guide cards.

(c) The aging management strategy of the CNP Unit 1 and 2 CASS LSCs is in accordance with MRP-227-A. CNP Unit 1 CMTRs were not located; therefore, they are conservatively assumed to be potentially susceptible to TE. Evaluation of the CNP Unit 2 CMTRs shows that the calculated ferrite contents range from 9.8 to 14.3 percent; therefore, they are not susceptible to TE using either the criteria in the Grimes letter or the NRC proposed criteria dated June 2014. The CNP Unit 1 and 2 LSCs both screen as potentially susceptible to IE.

I&M is participating in a PWROG project which is interfacing with the NRC to develop and provide a generic functionality evaluation of the lower support structure. The initial approach has been presented to the NRC, and the summary report for the first phase of the project has been submitted to the NRC for information. The project will develop a methodology which will form the basis for generic or plant-specific functionality analysis of the lower support structure. This project has a projected completion date of 2017. I&M will continue to participate in the PWROG project for LSCs. I&M will provide a supplemental response to the NRC on this RAI when an acceptable methodology is developed by the PWROG project.

Enclosure 2, PWROG-15066-NP, under the section titled "Response to Request Part (c)," provides further discussion on the LSCs.

The CNP RVI AMP follows the guidance of MRP-227-A, Table 4-6, "Westinghouse plants Expansion components," which states that the LSC bodies are an expansion inspection component linked to the CRGT lower flange welds primary inspection component. I&M recognizes that MRP-191, Table 4-6, "Screening Input Parameters for Westinghouse-Designed Plants," indicates that the CRGT lower flanges experience lower neutron fluence than the LSC bodies.

To address this concern, the industry is updating MRP-227 to include updated logic with an alternate primary component as the link for the LSC bodies. I&M is required to implement the revised industry guidance as prescribed by the EPRI MRP through the NEI 03-08 protocol.

Follow-Up RAI-3

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.

Response to Follow-Up RAI-3

Items related to CRGT support (split) pins are discussed below.

(a) I&M will replace the existing CNP Unit 1 Alloy X-750 split pins, with the more cracking resistant material of type 316 stainless steel, in the fall of 2017.

I&M will replace the existing CNP Unit 2 Alloy X-750 split pins, with the more cracking resistant material of type 316 stainless steel, in the fall of 2016.

The replacement split pins are fabricated from an improved material which is not susceptible to PWSCC, the primary failure mechanism for Alloy X-750 split pins.

(b) I&M does not plan to perform visual examinations of accessible portions of the split pins during the period of extended operation concurrent with the 10-year ISI interval ASME Code, Section XI, Category B-N-3 inspections.

The MRP-227-A, Section 4.4.3, "Westinghouse Components," guidance for CRGT split pins is limited to plant specific recommendations which instruct owners to review and follow the supplier recommendations for aging management and subsequent performance monitoring. MRP-227-A, Table 3-3, "Final disposition of Westinghouse internals," does not identify type 316 stainless steel split pins as requiring aging management. MRP-227-A, Table 4-9, "Westinghouse plants Existing Programs components," does not include type 316 stainless steel split pins as a line item. I&M is proactively replacing this set of components at CNP Units 1 and 2, which is in accordance with the supplier instructions by replacing the existing Alloy X-750 split pins as described in part (a) of this response.

The replacement type 316 stainless steel split pin supplier, Westinghouse, has not recommended inspection following installation and return to service. The design of the replacement type 316 stainless steel split pins for CNP Units 1 and 2 will be based on a program developed through the legacy Westinghouse Owner's Group, now the PWROG, which assessed the effects of wear, fatigue, stress relaxation, creep, SCC, swelling, and embrittlement. The PWROG program bounds the degradation mechanisms identified in MRP-191, Table 5-1, "Screening Table for Westinghouse Reactor Internals," which indicate that type 316 stainless steel split pins are screened in for wear, fatigue, and irradiation stress relaxation/irradiation creep, and classified as a Category A, "No Additional Measures" component. This design first entered service in 1997 and it has since been installed in the majority of applicable domestic plants. No failures or adverse operating experience have been observed to date in the type 316 stainless steel split pin design. Therefore, CNP existing programs comply with the supplier recommendations and MRP-227-A requirements for aging management of type 316 stainless steel CRGT split pins. No inspections are required.

Follow-Up RAI-4

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:*

<u>DC Cook 1 Barrel-Former Bolt Failures</u>:

(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.*

<u>DC Cook 2 Baffle-Former Bolt Failures</u>:

(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.

Response to Follow-Up RAI-4

Items related to the CNP Unit 1 barrel-former bolt issue are described below in parts (a) through (c).

(a) CNP Unit 1 is a 4-loop Westinghouse reactor with a thermal shield attached to the core barrel by supports at the top and flexures at the bottom. Differential thermal expansion between the core barrel and former plates resulted in bending loads in the three failed barrel-former bolts which could have affected preload. This bending load combined with elevated vibration loads associated with the bolts' proximity to a thermal shield support block may have provided the driver for wear by fretting on the bottom (relative to the core barrel) of the bolt threads. This wear led to the eventual thread disengagement of one of the three bolts and looseness in the remaining two of three bolts.

The replacement bolts addressed these concerns by selecting a higher yield strength material. This allowed more elastic strain to be stored in the bolt in the form of preload. This higher preload reduces bolt loads, particularly for small high cycle loading, such as flow induced vibration loads.

(b) The three barrel-former bolts were the only locations with indications and/or observed to be loose. The original bolt material is type 347 stainless steel. The replacement bolt material is type 316 stainless steel.

(c) The CNP Unit 1 barrel-former bolts are included as an expansion inspection in Appendix B of the CNP RVI AMP. In addition, CNP Unit 1 has a DMIMS which may detect loose parts in the RCS in the form of free/unrestrained barrel-former bolts or bolt fragments.

Neutron noise monitoring was used periodically during cycle U1C15 as prescribed in AEP:NRC:1239, "Donald C. Cook Nuclear Plant Unit 1: Commitments Regarding Barrel Former Bolt Issue," dated October 9, 1995, Item 4. However, neutron noise monitoring was discontinued following bolt replacement and it is not currently in use at CNP Unit 1.

The failures were first identified by observation of a bolt fragment on the lower core plate during a visual inspection of the lower core plate following core unload. A comprehensive foreign material inspection is performed during every outage at CNP between core unload and core reload, including the lower core plate and a sample of accessible areas below the lower core plate. These inspections could detect barrel-former bolts or bolt fragments displaced from their installed location.

Access was gained for replacement of the three CNP Unit 1 barrel-former bolts by machining a hole in the thermal shield at each bolt location which provide limited accessibility for continued monitoring. A visual inspection was performed on these three bolts concurrent with the 10-year ISI interval ASME Code, Section XI, Category B-N-3 inspections conducted in 2010 which confirmed that the replacement bolts were still in place with no indications.

Items related to the CNP Unit 2 baffle-former bolt issue are described below in parts (d) through (g).

(d) An evaluation was performed to reduce the number of required baffle-former bolts to exclude credit for the most eastern and western edge of the degradation zone on the large south baffle plate. Items considered included baffle jetting, core bypass flow, fatigue, loss of coolant accident and seismic loads, and loose parts. The evaluation concluded that the reactor internals would continue to perform their design basis function. No credit is taken for the four bolts on each side of the degradation zone, for a total of eight uncredited bolts.

Two additional baffle-former bolts with visual indications were identified during a post-maintenance visual inspection following bolt replacement. These bolts were located in the population along the edges of the degradation zone where no credit is taken. Therefore, the two baffle-former bolts were removed and the holes abandoned.

A specific evaluation was performed for the two vacant baffle-former bolt locations. Items considered included localized thermal stress of the baffle plate, radiation assessment of reactor internals and reactor vessel components, thermal-hydraulic assessment of core bypass flow through the open bolt holes, and evaluation of flow velocities through the open bolt holes for the potential of erosion on the baffle and former plates. The evaluation concluded that acceptable margins existed to resume operation with the two hole vacancies.

(e) The total original design number of baffle-former bolts at CNP Unit 2 was 832. The total number of installed baffle-former bolts at CNP Unit 2 is 830 due to the two vacant locations.

A visual inspection was performed on all baffle-former bolts on the four large baffle plates at CNP Unit 2 in 2010. Indications were found only on the large south baffle plate.

One bolt removed from each of the north, east, and west large baffle plates was tested. Each bolt was tensile load tested to a maximum load which bounded design loads. Non-destructive evaluations were performed on each bolt pre- and post-tensile test which included macro imaging, fluorescent dye penetrant testing, and ultrasonic testing from the shank end of the bolt. Each of the bolts was tested for chemical composition. The bolt sample from the north wall was sectioned, cross-sectional metallography was performed, and hardness testing was performed.

Higher chromium content was observed in the bolts from the east and west wall, which was consistent with the chemistry test of one bolt from the south wall. This is considered beneficial with respect to SCC resistance of austenitic stainless steels in a pressurized water reactor RCS environment, and it is; therefore, not an indication of increased susceptibility.

No other anomalies or indications were identified in any of the experimental results.

(f) The pattern of failed baffle-former bolts in Unit 2 did not strictly follow the pattern of highest neutron exposure. Neutron exposure is highest on the re-entrant corners of the baffle-former assembly where no failures were observed. The pattern of failed bolts did not conform to the areas of highest neutron exposure within the large baffle where the degradation area was observed. The degradation area was in the upper middle area of the baffle plate while the area of highest neutron exposure was in the lower middle area of the baffle plate. It should be noted that all of the baffle-former bolts on the large plates exceeded three displacements per atom making them susceptible to irradiation assisted SCC.

(g) The original CNP Unit 2 baffle-former bolt material is type 347 stainless steel. The replacement bolt material is type 316 stainless steel.

(h) The CNP Unit 2 baffle-former bolts will be volumetrically inspected during the cycle 25 refueling outage (U2C25) scheduled for 2019, with a projected 29.7 EFPY. This inspection is scheduled to be performed prior to 30 EFPY, which is in the first half of the initial inspection window of 25-35 EFPY for baffle-former bolts. The volumetric inspection will be performed approximately 10 years after discovery, investigation, and repair of the CNP Unit 2 baffle-former bolt degradation performed during the cycle 19 refueling outage (U2C19) in 2010, which is the inspection frequency prescribed in the CNP RVI AMP.

I&M originally scheduled the CNP Unit 2 baffle-former bolt volumetric inspection during the cycle 27 refueling outage (U2C27) scheduled for 2022, with a projected 32.4 EFPY. However, based upon the Unit 2 specific operating experience the inspection was rescheduled for an earlier refueling outage, currently scheduled for U2C25. This schedule is considered adequate based upon the justification below.

The observed degradation was isolated to one of the large CNP Unit 2 baffle plates. A sampling approach was used to bound the degradation area on the large south baffle plate and confirm that similar degradation was not occurring on the remaining three large baffle plates. The degradation was addressed by replacing bolts in the degradation area with bolts that were an improved material and geometry. Bolt replacement addresses a number of possible causes as described in the list below:

- ***Stress Corrosion Cracking*** – The replacement bolts are an improved material and geometry which are less susceptible to SCC; replacement restarted the initiation time for this failure mechanism in the degradation area.
- ***Loss of Preload*** – The replacement bolts are an improved material and geometry, installed with a higher preload than the original bolts. The replacement bolts are susceptible to stress relaxation or loss of preload similar to the original bolts, but this is a symmetric degradation mechanism which is not a plausible root cause for the isolated degradation observed on the large south baffle plate.
- ***Embrittlement*** – The virgin replacement bolts restarted material property changes caused by this degradation mechanism in the degradation area. The replacement bolts are susceptible to irradiation embrittlement similar to the original bolts, but this is a symmetric degradation mechanism which is not a plausible root cause for the isolated degradation observed on the large south baffle plate
- ***Fatigue*** – The replacement bolts are an improved material and geometry which are installed to a higher preload. The reduced stress concentration in the head-to-shank radius results in a lower fatigue usage factor at this location.

- ***Un-zippering*** – Bolt sampling was performed to bound the degradation area and bolt replacement ensured that there was not a large area of unsecured baffle plate which could lead to un-zippering.
- ***Manufacturing, fabrication, installation, and construction items related directly to the original bolts*** – I&M selected a qualified vendor to perform baffle-former bolt replacement through the modification process. The replacement bolts were designed and manufactured in accordance with applicable codes, standards, and requirements. Installation was performed by the vendor using approved procedures and under an ASME Section XI Repair and Replacement plan. This approach ensured quality and completeness through the design, fabrication, and installation process.

The possible causes reduced, reset, or mitigated by bolt replacement address all of the root and contributing causes identified by I&M for the isolated degradation on the CNP Unit 2 large south baffle plate, except for the steady-state pressure gradient across the baffle plates. The steady-state pressure gradient is a symmetric condition which is not a plausible root cause for the isolated degradation observed on the large south baffle plate. Therefore, this isolated instance of degradation has been addressed by permanent repair through bolt replacement.

No additional component specific CNP Unit 2 baffle-former bolt inspections are currently planned between the voluntary visual inspection performed during the cycle 20 refueling outage (U2C20) in 2012 and the first volumetric inspection scheduled for U2C25 in 2019. I&M continues to monitor the CNP Unit 2 DMIMS during operation and perform comprehensive foreign material inspections of the reactor vessel during each refueling outage.

<u>Follow-Up RAI-5</u>

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.

<u>Response to Follow-Up RAI-5</u>

Table 5-1, "Reactor Vessel Internals Aging Management Program Initial Inspection Schedule" lists the initial primary component inspection schedule, which is in accordance with the CNP RVI AMP. The outage dates and EFPY are estimated based on the projected generation schedule.

Table 5-1: Reactor Vessel Internals Aging Management Program Initial Inspection Schedule

	CNP Unit 1			CNP Unit 2		
Primary Inspection Component	**Outage**	**Year**	**EFPY**	**Outage**	**Year**	**EFPY**
CRGT Assembly Guide plates (cards)	U1C28	2017	29.3	U2C23	2016	26.9
CRGT Assembly Lower flange welds	U1C28	2017	29.3	U2C23	2016	26.9
Core Barrel Assembly Upper core barrel flange weld	U1C28	2017	29.3	U2C25	2019	29.7
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	U1C28	2017	29.3	U2C25	2019	29.7
Core Barrel Assembly Lower core barrel flange weld	U1C28	2017	29.3	U2C25	2019	29.7
Baffle-Former Assembly Baffle-edge bolts	U1C28	2017	29.3	U2C25	2019	29.7
Baffle-Former Assembly Baffle-former bolts	U1C31	2022	33.6	U2C25	2019	29.7
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	U1C28	2017	29.3	U2C25	2019	29.7
Alignment and Interfacing Components Internals hold down spring	U1C28	2017	29.3	U2C26	2021	31.1
Thermal Shield Assembly Thermal shield flexures	U1C28	2017	29.3	U2C25	2019	29.7