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October 30, 2015

AEP-NRC-2015-98
10 CFR 50.4

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant Units 1 and 2
SUPPLEMENTAL 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.

7) Letter from J. P. Gebbie, I&M, to NRC, "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," dated August 6, 2015, ADAMS Accession Nos. ML15223A435 and ML15233A436.

8) Letter from A. W. Dietrich, NRC, to I&M, "Summary of September 23, 2015, Public Meeting with Indiana Michigan Power Company Regarding Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC NOS. MF0050 and MF0051)," dated October 5, 2015, ADAMS Accession No. ML15271A046.

This letter provides Indiana Michigan Power Company's (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, supplemental 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 References 2 and 6, the NRC transmitted RAIs regarding the program. References 3, 4, 5, and 7 provided I&M's responses to References 2 and 6. By Reference 8, the NRC requested clarification of the follow-up RAI response for RAI-2(b) of Reference 6.

Enclosure 1 to this letter provides a response to Reference 8. Enclosure 2 provides PWROG-15066-NP, Revision 1, "Responses to Follow-Up NRC RAI 2 on the DC Cook Units 1 and 2 Reactor Internals Aging Management Program," dated October 15, 2015.

There are no new commitments in this letter. 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. Supplemental Response to Follow-Up Request for Additional Information Regarding the Reactor Vessel Internals Aging Management Program
2. PWROG-15066-NP, Revision 1, Responses to Follow-Up NRC RAI 2 on the DC Cook Units 1 and 2 Reactor Internals Aging Management Program

c: A. W. Dietrich, NRC Washington, D.C.
J. T. King - MPSC
MDEQ- RMD/RPS
NRC Resident Inspector
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SUPPLEMENTAL RESPONSE TO FOLLOW-UP REQUEST FOR ADDITIONAL INFORMATION REGARDING THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM

List of Acronyms and Abbreviations:

-A	NRC Approved (by Safety Evaluation)
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 Power Plant
CRGT	Control Rod Guide Tube
EPRI	Electric Power Research Institute
I&M	Indiana Michigan Power Company
kJ/m^2	Kilojoule per Meter Squared
MRP	EPRI Materials Reliability Program
MSC	PWROG Materials Committee (legacy Materials Subcommittee)
NEI	Nuclear Energy Institute
-NP	Non-Proprietary
NRC	U. S. Nuclear Regulatory Commission
-P	Proprietary
PA-	PWROG Project Authorization
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
RAI	Request for Additional Information
RCCA	Rod Control Cluster Assembly
RVI	Reactor Vessel Internals
SR	Surveillance Requirement
TS	Technical Specification
VT	Visual Testing
WAAP	Westinghouse Association Activities Paper
WCAP	Westinghouse Commercial Atomic Power

Introduction

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 (ADAMS Accession No. ML14216A497), the responses to RAI-1, RAI-5, and RAI-7 were provided to the NRC. By letter dated September 4, 2014 (ADAMS Accession Nos. ML14253A316, ML14253A317, and ML14253A318), the response to RAI-8 was provided to the NRC. By letter dated October 22, 2014 (ADAMS Accession No. ML 14316A449), the responses to RAI-2, RAI-3, RAI-4, and RAI-6 were provided to the NRC. By letter dated May 5, 2015 (ADAMS Accession No. ML15119A339), the NRC staff transmitted a follow-up RAI to complete its review. By letter dated August 6, 2015

(ADAMS Accession Nos. ML15223A435 & ML15223A436), a response to the follow-up RAI was provided. By letter dated September 11, 2015 (ADAMS Accession No. ML15257A289), the NRC scheduled a public teleconference where the staff requested clarification and a supplemental submittal for four specific items in the response to follow-up RAI provided by I&M. By meeting summary dated October 5, 2015 (ADAMS Accession No. ML15271A046), the NRC documented the request for elaboration on the four statements made in I&M's response to follow-up RAI-2(b). Clarification of these four items from the response to the follow-up RAI is provided in this enclosure with additional information provided in Enclosure 2.

Requested Clarification

A detailed response to follow-up RAI-2(b) was provided by letter dated August 6, 2015, "Response to Follow-Up Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," in Enclosure 2, 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." The NRC requested that I&M elaborate on four items from the response in this document during a public meeting held on September 23, 2015. These four items are listed below with additional detail following each item.

A detailed records search revealed that the CNP Unit 1 CRGT guide cards were specified with a primary material of wrought Type 304 stainless steel, with an alternate material of CASS CF8 allowed. No CMTR or other manufacturing record was located which indicated the fabrication material. It is expected that the components are fabricated from wrought Type 304 stainless steel. However, in the absence of confirmatory records, these components are conservatively assumed to be fabricated from CASS CF8.

Item #1: Requested Clarification from Response to Follow-Up RAI-2(b), CASS CRGT Guide Card Cracking Effects Observations

Thermal embrittlement of CASS does not result in a complete loss of fracture toughness. Even with complete thermal embrittlement, a significant amount of fracture toughness would remain in the card.

Additional Detail on Item #1:

A meeting was held between the PWROG and NRC on September 16, 2015, to discuss a statistical approach to evaluating CASS. Presentation WAAP-9551, "PA-MSC-1288R0: PWR Materials Assessment Results," was provided and discussed with the NRC staff. The results of the PWROG work identified that RVI materials fabricated from CF8 material show saturation fracture toughness has a large margin above the 255 kJ/m^2 screening criteria provided in the Grimes Letter (ADAMS Accession No. ML003717179). The minimum fracture toughness of CASS CF8 was 100 kJ/m^2 over the screening criteria.

In lieu of plant specific information on the potentially CASS guide cards in CNP Unit 1, I&M proposes to use the industry-wide distribution discussed above to identify the fracture toughness of these components. The industry-wide distribution has a non-significant reduction in fracture toughness due to thermal embrittlement. Therefore, this set of components is considered to have a non-significant reduction in fracture toughness and does not require additional aging management requirements due to thermal embrittlement.

Enclosure 2 to this letter, PWROG-15066-NP, Revision 1, "Responses to Follow-Up NRC RAI 2 on the DC Cook Units 1 and 2 Reactor Internals Aging Management Program," under the section title "Response to Request Part (b)," has been enhanced to provide further discussion and clarification of Item #1.

Item #2: Requested Clarification from Response to Follow-Up RAI-2(b), CASS CRGT Guide Card Conclusion Considerations

The scope of the current VT-3 Primary Component inspection for wear required under MRP-227-A [3], which would detect gross failures.

Additional Detail to Item #2:

The requirements of VT-3 inspections are provided in ASME Boiler & Pressure Vessel Code, Section XI, with additional details specific to RVI inspections under MRP-227-A provided in MRP-228, Revision 1, "Materials Reliability Program: Inspection Standard for PWR Internals – 2012 Update."

In addition to the guidance provided in EPRI report MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," as published in December 2011, CNP will also follow the interim guidance provided in EPRI letter MRP 2014-006 (ADAMS Accession No. ML14274A372), "*Materials Reliability Program*: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI, Palo Alto, CA: 2011. 1022863, Transmittal of Interim Guidance," published February 2014. This letter provides interim guidance related to MRP-227-A, Table 4-3, "Westinghouse plants Primary Components," and Table 5-3, "Westinghouse plants examination acceptance and expansion criteria." The interim guidance issued by EPRI directs utilities to follow guidance provided in WCAP-17451-P (ADAMS Accession No. ML15041A106), Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections." I&M is following the six NEI 03-08 "needed" requirements described in Section 6, "Summary Conclusions and Actions."

I&M plans to inspect, at a minimum, all active rodded CRGTs as recommended in WCAP-17451-P, Section 5.5.2, "Extent of Wear Inspections or Measurements," which exceeds the existing requirements of "20% of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined," from MRP-227-A. The actual inspection coverage may vary depending upon accessibility and field conditions.

This inspection strategy is adequate to detect cracking prior to loss of function of the guide cards. While the inspection will be primarily used to evaluate wear, it will also detect damage such as deformed guide card sections and missing card ligaments. The VT-3 inspection is not intended to detect fine cracking. However, the components remain functional until cracking becomes extensive enough that it is detectable by VT-3 inspection.

Therefore, the existing inspection strategy is adequate and does not require additional aging management requirements.

Enclosure 2 to this letter, PWROG-15066-NP, Revision 1, under the section title "Response to Request Part (b)," has been enhanced to provide further discussion and clarification of Item #2.

Item #3: Requested Clarification from Response to Follow-Up RAI-2(b), CASS CRGT Guide Card Conclusion Considerations

The redundancy of the component – failures at multiple cards would be required for the control rods to slip out of place and the failure of the control rods in one CRGT assembly to insert would not preclude safe shutdown.

Additional Detail on Item #3:

A discussion of the redundancy of the CRGT guide cards can be found in WCAP-17451-P. CNP Unit 1 is a 4-loop Westinghouse design. Table 4-1, "Plant Groupings," identifies the CNP Unit 1 guide tube design and Figure 2-21, "Guide Tube Design Layouts," describes the CNP Unit 1 general geometry. This information is used in Table 5-10, "Summary of Allowable Number of Worn-Through (Open) Guide Cards," to find the number of allowable consecutive open guide cards. Section 5.2.1, "Guide Card Wear Criteria," provides a discussion and further context of the allowable consecutive open guide cards.

CNP Unit 1 maintains shutdown margin as described in TS, Section 1.1 "Definitions," item "Shutdown Margin," which assumes that all RCCAs are fully inserted, except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn (failed to insert). Therefore, in addition to the redundancy of the CRGT guide cards, there is redundancy in the CRGT assemblies with respect to shutdown margin.

Enclosure 2 to this letter, PWROG-15066-NP, Revision 1, under the section title "Response to Request Part (b)," has been enhanced to provide further discussion and clarification of Item #3.

Item #4: Requested Clarification from Response to Follow-Up RAI-2(b), CASS CRGT Guide Card Conclusion Considerations

The periodic monitoring of control rod functionality under current plant procedures.

Additional Detail to Item #4:

Control rod functionality and operability is monitored through testing, which provides evidence of the absence of damage impacting the function of the guide cards:

1. Control Rod Drag Testing
   Control rod drag testing is performed every time a RCCA is latched to a drive shaft. RCCAs are latched to their respective drive shaft each time the reactor is reassembled, including following each refueling outage. The drag testing demonstrates that predetermined resistance load acceptance criteria are met.

2. Control Rod Drop Testing
Control rod drop testing is performed, at a minimum, each time the reactor vessel closure head is removed, prior to reactor criticality, in accordance with TS SR 3.1.4.3. The drop testing demonstrates rod drop times are within predetermined acceptance criteria following maintenance and/or refueling, and prior to reactor criticality.

3. Quarterly Operability Surveillance
Control rod operability testing is performed quarterly in accordance with TS SR 3.1.4.2. The operability testing moves each rod which is not fully inserted by a minimum number of steps in either direction to validate that behavior is within predetermined acceptance criteria.

Enclosure 2 to this letter, PWROG-15066-NP, Revision 1, under the section title "Response to Request Part (b)," has been enhanced to provide further discussion and clarification of Item #4.