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INDIANA MICHIGAN POWER

April 23, 2004

AEP:NRC:4034-02
10 CFR 54

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Units 1 and 2
Docket No. 50-315 and 50-316
Supplemental Information for the Donald C. Cook Nuclear Plant License Renewal Application – Aging Management Program Audit (TAC Nos. MC 1202 and MC 1203)

Dear Sir or Madam:

By letter dated October 31, 2003, Indiana Michigan Power Company (I&M) submitted an application to renew the operating licenses for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2.

The Nuclear Regulatory Commission (NRC) review process includes audits of the aging management programs (AMPs) credited in the CNP license renewal application. During the conduct of these audits and subsequent to conversations between I&M and NRC Staff, it has become necessary to provide additional information and to revise certain information regarding the license renewal application. This letter provides the supplemental information requested during the AMP audits.

The enclosure to this letter provides an affirmation pertaining to the statements made in this letter. The attachment to this letter provides the supplemental information requested by NRC Staff. There are no new commitments contained in this submittal.

Should you have any questions, please contact Mr. Richard J. Grumbir, Project Manager, License Renewal, at (269) 697-5141.

Sincerely,

M. K. Nazar

M. K. Nazar
Senior Vice President and Chief Nuclear Officer

NH/rdw

A104

Enclosure: Affirmation

Attachment: Donald C. Cook Nuclear Plant License Renewal Application – Supplemental Information Identified during the Aging Management Program Audit

c: J. L. Caldwell, NRC Region III
K. D. Curry, AEP Ft. Wayne
J. T. King, MPSC
J. G. Lamb, NRC Washington DC
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NRC Resident Inspector
J. G. Rowley, NRC Washington DC

Enclosure to AEP:NRC:4034-02

AFFIRMATION

I, Mano K. Nazar, being duly sworn, state that I am Senior Vice President and Chief Nuclear Officer of American Electric Power Service Corporation and Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the 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.

American Electric Power Service Corporation

M. K. Nazar

M. K. Nazar
Senior Vice President and Chief Nuclear Officer

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 23 DAY OF April, 2004

Bridget Taylor
Notary Public

My Commission Expires 6/10/2007

BRIDGET TAYLOR
Notary Public, Berrien County, MI
My Commission Expires Jun. 10, 2007

Donald C. Cook Nuclear Plant License Renewal Application
Supplemental Information Identified during the Aging Management Program Audit

The Donald C. Cook Nuclear Plant (CNP) aging management programs (AMPs) discussed in this attachment were the subject of a Nuclear Regulatory Commission license renewal application (LRA) review audit performed on March 2, through March 4, 2004. In response to questions raised during this audit, the following LRA clarifications are provided.

<u>Clarification to LRA Table 3.1.1 – Thermocouple Nozzle Assembly Head Port Adapter and LRA Table 3.1.2-1, Reactor Vessel and CRDM Pressure Boundary</u>

Based on a review of LRA Table 3.1.1, Item 3.1.1-36, Pages 3.1-29 and 3.1-30, and the component type "CRDM housing" listed in LRA Table 3.1.2-1, Page 3.1-44, it was determined that a separate line item for the thermocouple nozzle assembly head port adapter would provide additional clarity. The core exit thermocouple assembly is bolted to the stainless steel control rod drive mechanism (CRDM) head nozzle adapter, which is attached to the nickel-based alloy CRDM penetration on the reactor vessel head.

The thermocouple nozzle assembly pressure-retaining items include the head port adapter (pressure housing), holddown nut, compression collar, Grafoil® seals, seal carrier assembly, and lockwasher. The head port adapter, holddown nut, and compression collar are fabricated from stainless steel and are subject to aging management review. The short-lived Grafoil seals, seal carrier assembly, and lockwasher are replaced each refueling outage and are not subject to aging management review.

The aging management review determined that the compression collar and holddown nut are part of the bolted connection used to connect the head port adapter to the core exit thermocouple column, which is part of the reactor vessel internals. The stainless steel holddown nut and compression collar are exposed to an external-ambient environment in which there are no aging effects that require management.

The stainless steel thermocouple nozzle assembly head port adapter is exposed to an internal environment of borated water and an external-ambient environment. The applicable aging effects include cracking and loss of material, which will be managed by the Inservice Inspection Program and Water Chemistry Control Program throughout the period of extended operation.

The thermocouple nozzle assembly head port adapter is attached to a CRDM head nozzle adapter by means of a threaded connection and a canopy seal weld. The canopy seal weld is exposed to an external-ambient environment. The applicable aging effect of the canopy seal weld is cracking, which is managed by the Boric Acid Corrosion Prevention Program.

The current listing for the CRDM housing will be retained and is consistent with NUREG-1801, *Generic Aging Lessons Learned (GALL) Report.*

Clarification to LRA Table 3.1.2-5 – Steam Generator Tubesheet

The secondary face of the steam generator tubesheet is not clad with nickel-based alloy as listed in the component type "Tubesheet" in LRA Table 3.1.2-5, Page 3.1-84. The Steam Generator Integrity Program includes visual secondary side inspections prior to and following sludge lancing. The Steam Generator Integrity Program and Water Chemistry Control Program will manage loss of material at the tubesheet surface exposed to secondary treated water throughout the period of extended operation.

Clarification to LRA Table 3.1.2-5 – Steam Flow Restrictors

LRA Table 3.1.2-5, Page 3.1-90, credits the Water Chemistry Control Program and Inservice Inspection Program for managing aging of the Unit 1 stainless steel steam flow restrictors. While the main steam nozzle receives volumetric examination each inspection interval, the internal flow restrictor is not inspected. Therefore, inservice inspection should not be credited for managing cracking of the stainless steel steam flow restrictors. Water chemistry control will manage cracking and loss of material at the stainless steel steam flow restrictors.

Clarifications to LRA Section A.2.1.7, Cast Austenitic Stainless Steel Evaluation Program, and LRA Section B.1.7, Cast Austenitic Stainless Steel Evaluation

In LRA Sections A.2.1.7 and B.1.7, Indiana Michigan Power Company (I&M) committed to develop and implement a Cast Austenitic Stainless Steel (CASS) Evaluation Program. Both sections discuss that the CASS Evaluation Program will use "additional inspections and a component-specific flaw tolerance evaluation" to manage aging of potentially susceptible components. LRA Section B.1.7 states that the CASS Evaluation Program will be comparable to the program described in NUREG-1801, Section XI.M12, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS).

NUREG-1801, Section XI.M12, states, regarding detection of aging effects, that:

> "For 'potentially susceptible' piping, because the base metal does not receive periodic inspection per ASME Section XI, the CASS AMP provides for volumetric examination of the base metal, with the scope of the inspection covering the portions determined to be limiting from the standpoint of applied stress, operating time, and environmental considerations. Examination methods that meet the criteria of the ASME Section XI, Appendix VIII, are acceptable. Alternatively, a plant- or component-specific flaw tolerance evaluation, using specific geometry and stress information, can be used to demonstrate that the thermally-embrittled material has adequate toughness."

Clarification to the I&M commitment is provided herein that CNP will retain the option of evaluating components that are potentially susceptible to thermal aging embrittlement utilizing either volumetric examinations or a flaw tolerance evaluation. LRA Sections A.2.1.7 and B.1.7 are revised for clarity by changing the words "additional inspections and a component-specific flaw tolerance evaluation" to "volumetric inspections or a component-specific flaw tolerance evaluation" for consistency with NUREG-1801. If a flaw tolerance evaluation is used, the appropriate method for flaw tolerance will be chosen when the evaluation is initiated, prior to the end of the current operating period (i.e., October 25, 2014, for Unit 1 and December 23, 2017, for Unit 2).

Clarification to LRA Section A.2.1.10, Diesel Fuel Monitoring

For completeness, to reflect that diesel fuel oil monitoring is performed per American Society for Testing and Materials (ASTM) Standards specified in the CNP Technical Specifications, LRA Section A.2.1.10 is revised to clarify that:

> "The program monitors fuel oil quality and the contaminant concentrations in the fuel oil *using ASTM Standards specified in the technical specifications*."
> (NOTE: The text added for clarification is in *italics*.)

Clarification to LRA Section A.2.1.15, Flow-Accelerated Corrosion Program, and LRA Section B.1.12, Flow-Accelerated Corrosion

For consistency with the listing for component type "Main steam nozzles" in LRA Table 3.1.2-5, LRA Sections A.2.1.15 and B.1.12 are revised to clarify that:

> "The Flow-Accelerated Corrosion (FAC) Program assures that the structural integrity of carbon steel pipes containing high-energy fluids *and the low alloy steel steam generator main steam nozzles are* maintained."
> (NOTE: The text added for clarification is in *italics*.)

Clarification to LRA Section A.2.1.30, Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program and LRA Section B.1.27, Reactor Vessel Internals Plates, Forgings, Welds, and Bolting

LRA Table 3.1.1, Item 3.1.1-42, Page 3.1-33, credits the Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program for managing loss of pre-load of the reactor vessel internals. The component type "Holddown spring" listed in LRA Table 3.1.2-2, Page 3.1-57, shows loss of preload is managed by this program. This new aging management program will supplement the reactor vessel internals inspections required by the American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection Program to assure the aging effects will not result in loss of the intended functions of the reactor vessel internals during the period of extended operation. The program will utilize acceptable inspection methods for bolted joints, including those necessary to demonstrate that loss of closure integrity due to stress relaxation (i.e., loss of pre-load) is addressed. For consistency with the tables above, the program descriptions in LRA Sections A.2.1.30 and B.1.27 are revised to clarify that:

> "The program will manage the following aging effects ...

- *Loss of bolted closure integrity due to stress relaxation (loss of preload)*"
  (NOTE: The text added for clarification is in *italics*.)

Clarification to LRA Section B.1.28, Reactor Vessel Internals CASS

The following statements in LRA Section B.1.28, Program Description, were intended to show how the Reactor Vessel Internal CASS Program will be consistent with NUREG-1801, Section XI.M13, Thermal Aging and Neutron Irradiation Embrittlement of CASS, with respect to disposition of cracks.

The last sentence in the second paragraph states that "The program will detect and manage cracking, reduction of fracture toughness, and dimensional changes." Management of cracking includes evaluation and disposition of identified cracks.

The third paragraph discusses monitoring propagation of existing flaws since reduction of fracture toughness affects flaw growth versus flaw detection.

Clarification to LRA Section B.1.34, Structures Monitoring – Divider Barrier Seal Inspection

Clarification is provided to address the method used to monitor a change in material properties of the elastomeric pressure seals through visual examinations. The phrase "change in material properties" was intended to convey a visual inspection to ensure the absence of apparent deterioration (i.e., cracks or defects in the sealing surfaces) as discussed in the implementing procedures. Changes in other material properties are not monitored or inspected.