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Indiana Michigan Power
One Cook Place
Bridgman, MI 49106
IndianaMichiganPower.com

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

Subject: Donald C. Cook Nuclear Plant Units 1 and 2
Docket No. 50-315 and 50-316
Response to Request for Additional Information Regarding Fourth Ten-Year ISI Program Plan

References:

1. Letter from R. A. Hruby, Indiana Michigan Power Company (I&M), to Nuclear Regulatory Commission (NRC) Document Control Desk, "Fourth Ten-Year Interval Inservice Inspection Program Plan," AEP-NRC-2010-21, Accession Number ML100750680, dated March 12, 2010.
2. Electronic Communication from P. S. Tam, NRC, to H. L. Etheridge, I&M "D.C. Cook Units 1 and 2 – Revised Draft RAI on the 3/12/10 submittal re: the 4th 10-year interval ISI program (TAC ME4495 and ME4496)," Accession Number ML102850748, dated October 12, 2010.

Dear Sir or Madam:

By letter dated March 12, 2010 (Reference 1), I&M, the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, submitted the Fourth Ten-Year Interval Inservice Inspection (ISI) Program Plan for CNP Unit 1 and Unit 2. By electronic communication dated October 12, 2010 (Reference 2), the NRC transmitted a Request for Additional Information (RAI) regarding the Fourth Ten-Year Interval ISI Program Plan.

The enclosure to this letter provides I&M's response to the RAI regarding the Fourth Ten-Year Interval ISI Program Plan for CNP Unit 1 and Unit 2.

This letter contains no new or modified regulatory commitments.

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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

MS/jmr

Enclosure: Response to Request for Additional Information Regarding Fourth Ten-Year Interval Inservice Inspection Program Plan

c: J. T. King, MPSC
S. M. Krawec, AEP Ft. Wayne, w/o enclosures
MDNRE – WHMD/RPS
NRC Resident Inspector
M. A. Satorius, NRC Region III
P. S. Tam, NRC Washington DC

Response to Request for Additional Information Regarding Fourth Ten-Year Interval Inservice Inspection Program Plan

By letter dated March 12, 2010 (Reference 1), Indiana Michigan Power Company (I&M), licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, submitted the Fourth Ten-Year Interval Inservice Inspection (ISI) Program Plan for CNP Unit 1 and Unit 2. By electronic communication dated October 12, 2010 (Reference 2), the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) regarding the Fourth Ten-Year Interval ISI Plan. I&M's response to the NRC's RAI is provided below.

NRC RAI 1

Supporting requirement (SR) IF-C3 in American Society of Mechanical Engineers (ASME) Probabilistic Risk Assessment (PRA) Standard RA-Sb-2005 identifies the failure mechanisms that shall be evaluated to determine the susceptibility of each safety-related structure, system, and component (SSC) in a flood area to flood-induced failures. Capability category II identifies failure by submergence and spray as requiring detailed analysis. Capability category III includes jet impingement, pipe whip, and humidity, condensation, and temperature concerns. Risk informed inservice inspection (RI-ISI) requires that all SSC failures induced by a pipe break be considered. Please demonstrate that all SSC failures that are induced by a pipe break are adequately addressed in your analysis.

I&M Response to RAI 1

One of the calculations, "Internal Flooding-Qualitative Screening Analysis" (QSA), performed as part of CNP's Internal Flooding Analysis (IFA) involved an evaluation of all flood areas to identify potential flooding sources, propagation paths, and equipment that may be adversely affected due to the accumulation of water, spray, dripping, and steam damage. The Internal Flooding QSA further states:

> "Walkdowns of the D.C. Cook Nuclear Plant were performed on a room-by-room basis ... during which time the location and presence of flood sources, equipment susceptible to damage, barrier openings (along with opening direction in the case of doors), general room layout, and flood propagation paths were noted. Only selected areas of Unit 2 were walked down such as those areas where asymmetries exist between the units, e.g., the area around the Unit 2 - AFW N Train Battery Room. Flooding was considered to occur as a result of leaks or ruptures of piping, gaskets, valves, pumps, expansion joints, tanks, or heat exchangers. Equipment failure was considered not only due to submergence but also as a result of spray and dripping. ...
>
> "Upon completion of the walkdowns, fire areas were subjected to a qualitative flooding screening procedure based upon, the presence of any PRA equipment in the room/area, the existence of a consequential initiating event given a flood,

the presence of a flood source in the room/area, and a determination of whether the flood can propagate to another area. ...

"The qualitative screening criteria for flood areas were taken from Draft Addenda B to ASME RA Sa 2003 (Ref 1) Flood Elements IF C5, IF C5a, and IF 6 Capability Category II ..."

An independent focused-scope PRA Peer Review was performed in 2009 by Westinghouse to determine compliance with Addendum B of the ASME PRA Standard (Reference 3) and Regulatory Guide 1.200 Revision 1. This Peer Review was conducted following the most recently available NEI Peer Review guidance (Reference 4). The scope of this review included the CNP IFA. This Peer Review stated that the IFA SR IF-C3 met Capability Category (CC) I/II. Although SSC failures by pipe whip and jet impingement were not explicitly addressed, I&M considers that all SSC failures that could be induced by a pipe break are adequately addressed in the analysis. This conclusion is based on the location of PRA-related equipment relative to the various spray sources, use of conservative estimates of possible spray coverage, and use of conservative SSC failure criteria if spray was judged to be able to reach the SSC.

<u>NRC RAI 2</u>

SR IF-C6 permits screening out of flood areas based on, in part, the success of human actions to isolate and terminate the flood. The endorsed RI-ISI methods require determination of the flood scenario with and without human intervention which corresponds to the capability category III (i.e., scenarios are not screened out based on human actions). Therefore a category III analysis is needed. To provide confidence that scenarios that might exceed the quantitative core damage frequency and larger early release frequency guideline are identified, please describe how credit is given to human actions.

<u>I&M Response to RAI 2</u>

One of the calculations, Internal Flooding QSA, performed as part of CNP IFA involved an evaluation of all flood areas to identify potential flooding sources, propagation paths, and equipment that may be adversely affected due to the accumulation of water, spray, dripping, and steam damage. The evaluation's scope included only the occurrence of flooding events in which a plant trip or requirement to shut down occurs. The CNP Internal Flooding QSA further states:

"Flood Area Qualitative Screening Analysis: A qualitative screening analysis based on flood sources, the presence of PRA sensitive equipment, potential for flood induced initiating events, and flood propagation was conducted. The criteria were taken from Draft Addenda B to ASME RA-Sa-2003."

However, the qualitative screening criteria provided in Draft Addenda B to ASME RA-Sa-2003 (Reference 3) includes the possibility that a flood area may be screened based on an operator action. Review of the detailed qualitative screening assessment confirms that the CNP Internal

Flooding QSA used only the screening criteria explicitly mentioned in the quote above and that no area was screened based on the assumption of any operator action.

Although no credit was taken for operator response to prevent the initiation of a flood scenario, credit was taken for operator actions in response to a flood-initiated event. Specifically, CNP's "Internal Flood Detailed Analysis" states:

> "Human reliability analyses (HRA) were performed and included the following scenario-specific performance shaping factors (PSF) for control room and ex-control room actions as appropriate:
>
> a) additional workload and stress (above that for similar sequences not caused by internal floods)
>
> b) uncertainties in event progression (e.g., cue availability and timing concerns caused by flood)
>
> c) effect of flood on mitigation, required response, and recovery activities (e.g., accessibility restrictions, possibility of physical harm)
>
> d) flooding-specific job aids and training (e.g., procedures, training exercises)"

According to CNP's "Internal Flood Human Reliability Analysis" (HRA), which provides the detailed HRA calculations, "The EPRI HRA Calculator® software [Ref. 2] was used for this analysis. The EPRI Calculator embodies the methodologies described in EPRI-TR-100259 [Ref. 3] and THERP [Ref.4]."

As stated above in response to RAI 1, an independent focused-scope PRA Peer Review was performed in 2009 that included the CNP IFA. This Peer Review stated that the IFA SR IF-C6 met CC III.

<u>NRC RAI 3</u>

> *SR IF-D5a addresses the development of flood initiating (pipe rupture) frequencies for use during the scenario development. The risk-informed inservice Inspection program is premised on inspecting locations with the highest risk, driven mostly by failure frequency. The plant-specific information collected and used should include experience related to degradation mechanisms that could indicate increased likelihood of pipe failure at particular locations. Please describe how plant-specific operating experience was used to identify experience related to degradation mechanisms and how this experience was incorporated into the development of pipe failure frequencies.*

<u>I&M Response to RAI 3</u>

One of the calculations, "Pipe Rupture Frequencies," performed as part of CNP IFA determined the rupture frequency of piping that could contribute to internal flooding. This calculation determined flood frequencies using the methodology in EPRI TR-102266 "Pipe Failure Study

Update", April 1993. This report provides what is termed the Jamali method. The Jamali method was used because it represented the most recent source of information on piping failures leading to floods at the time that the CNP calculation was performed. In this method, as stated in the calculation,

> "...piping is divided into pipe sections. A pipe section is defined as a segment of piping between major discontinuities such as valves, pumps, reducers, tees, etc. A pipe section is typically from 10 to 100 feet in length and contains 4 to 8 welds. Each section can also contain several elbows and flanges. Instrumentation connections are not considered major discontinuities."

To implement the Jamali method, plant-specific data was gathered, with the number of pipe sections and the components being determined from walkdowns and plant drawings. Then, the appropriate piping-section failure rates were selected. The Jamali method provides piping failure rates for different plant types (i.e., PWR, BWR), piping types (Reactor Coolant System, Safety Injection/Recirculation, Other Safety Related, Main and Auxiliary Feedwater and Condensate, Main and Auxiliary Steam, and other non-safety related systems), and piping sizes (i.e., 0.5 inch (in) – 2 in inside diameter (ID), 2 in – 6 in ID, and > 6 in ID). The Jamali method failure rates include contributions from all failure mechanisms. Finally, the Jamali method failure rates were combined with the plant-specific piping data.

The Jamali report identifies three types of plant-specific corrections that may be made to the generic failure rates: "plant age, system type, and failure cause susceptibility and/or resistance." Regarding plant age, the Jamali report states:

> "The above results suggest that the effect of plant age on the rupture failure rates is small compared with other effects over the expected plant life of forty years or so, and plant age can be ignored as a correction factor. Note that this conclusion may not hold in some cases, but the rupture data base is not extensive enough to allow for the quantification of the impact of plant age in specific applications."

Regarding system type corrections, the Jamali report states:

> "In certain applications, it may become desirable to apply additional correction factors for specific system types (i.e., Service Water) within a system group (OSR). ... Without a great deal of care, accounting for such variations may lead to more error than added precision."

Regarding failure cause susceptibility, the Jamali method includes a suggested method for adjusting piping failure rates based on a known susceptibility or lack of susceptibility to a particular failure mechanism. The degree of adjustment is based on the analyst's subjective view of available piping inspection information. No specific technical basis or direction is provided for determining the degree of adjustment based on information that should be available.

No adjustments were made in CNP's "Pipe Rupture Frequencies" calculation, nor have any such adjustments been applied since the calculation was completed. However, CNP's "Internal Flood Detailed Analysis" did adjust piping failure rates for the Component Cooling Water piping based on the availability of a later source of piping failure estimates in EPRI TR-1013141 "Pipe Rupture Frequencies for Internal Flooding PRAs – Revision 1", March 2006. This report provides what is termed the Fleming method for determining piping failure frequency. Since the Fleming method determines piping failure frequencies based on linear feet of piping instead of piping segments and CNP's data was assembled by piping segments, this report became available too late during the preparation of CNP's "Pipe Rupture Frequencies" calculation and therefore was not used in this calculation. Given the importance of CCW piping failures in the quantification using the Jamali method, additional system data was obtained and CCW piping failure frequencies were re-calculated using the Fleming method.

The Fleming method yields different high-level results than the Jamali method primarily due to:

> "Differences in the completeness of the underlying failure data. This is likely a major reason for the differences since the failure data in this study is derived from a more comprehensive set of sources, has been extensively validated, and captures data for more than a decade since the Jamali report was published."

Two key aspects of the Fleming method are:

> "A simple model is used based on the assumption that all pipe failures including wall thinning, cracks, leaks, and major structural failures are precursors to flooding. The model expresses pipe failure mode frequencies in terms of a failure rate that covers all failure modes of the pipe and a conditional probability of the flood mode of interest. In this study three flood modes are considered: failures that produce water sprays with leak flow rates up to 100 gpm [gallons per minute], failures that produce localized flooding with leak flow rates between 100 and 2,000 gpm, and pipe failures that produce major flooding with leak flow rates in excess of 2,000 gpm."

> "Variability in pipe failure mode frequencies due to differences in physical parameters such as system type, pipe material, pipe size, water or fluid characteristics, etc. are based on subdividing the database to provide different data sets for each combination of parameters."

The application of the Fleming method to the CCW system was compared to the Jamali method results in CNP's "Internal Flood Detailed Analysis." The Fleming method yielded smaller (in most cases, significantly smaller) CCW piping failure frequencies than predicted by the Jamali method for the most damaging CCW floods. The lower CCW piping failure frequencies obtained from the Fleming method are judged to be due to the data treatment by Fleming. The Fleming method specifically separated CCW piping failures from the broad Other Safety Related (OSR) piping category that was used in the Jamali approach.

The Fleming report includes a discussion of how its information can be used to meet the internal flooding requirements of the ASME standard; the pertinent statements are as follows:

"• Relevant generic industry data for estimation of flood initiating event frequencies is provided in this report. This includes events involving water sprays, significant floods, and major floods with spill rates up to and including the maximum flow rates of the system flood sources.

- The significant flooding events listed in Appendix C can be used to identify plant specific flooding events that can be considered for inclusion in the plant specific Bayes' updating ASME PRA Standard Requirements for Internal Flooding process. This step is the responsibility of the plant specific PRA team. The distributions provided in Appendix A can be used as prior distributions in these Bayes updates.

- The flood frequencies presented in this report are calculated on an events per reactor operating year basis where critical hours is used to approximate the actual operating hours. For use in full power PRAs or in low power or shutdown mode PRAs, the plant specific PRA teams must apply the predicted plant availability factor with an appropriate formula so that the resulting event frequencies are converted into events per reactor calendar year. Because of this treatment, the issue regarding the historical plant availability and its difference with the predicted plant availability is taken care of.

- Uncertainties in the generic data that influence internal flood initiating event frequency are quantified in this report and found in Appendix A."

The generic Fleming flooding failure rates per linear foot of piping were transformed into flooding initiating event frequencies using detailed, plant-specific piping lengths and plant-specific operating hours. No adjustment was made to the generic Fleming flooding failure rates since Appendix C of the Fleming report does not include any CNP-specific flood events.

As stated above in response to RAI 1, an independent focused-scope PRA Peer Review was performed in 2009 that included the CNP Internal Flooding Analysis. This Peer Review stated that the Internal Flooding Analysis SR IF-D5a met CC II/III on the strength of the plant-specific information obtained for use in calculating the various system piping failure frequencies.

<u>NRC RAI 4</u>

Code case N-716 has a minimum requirement of high safety significant inspections and relies on the PRA flooding analysis to identify additional inspection areas. The NRC safety evaluation (Accession No. ML072620553) dated Sept 28, 2007 approving RI-ISI N-716 for DC Cook states that the licensee reported two scenarios that exceeded the (risk) metrics. One scenario was reduced below the guideline values by reflecting a plant change in the analysis and the second scenario was reduced below the guideline value based on a more detailed analysis of the human error probabilities associated with the scenario. The current submittal states that PRA model revisions have occurred since the last RI-ISI program was approved. Please verify that a flooding analysis was performed using the latest PRA model for the fourth ten-year interval inspection and state whether any segments were identified that may have a CDF or LERF greater than 1E-6/year or 1E-7/year, respectively.

I&M Response to RAI 4

The CNP risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in applicable CNP PRA procedures and guidelines. Additionally, CNP Engineering procedures include directions to preparers of Engineering Changes and calculations to identify and communicate to the PRA group possible impacts on the PRA model. This helps assure that PRA model impacts due to plant configuration changes are identified and their magnitude estimated prior to implementation.

The calculations comprising the Internal Flooding Analysis were completed in mid-2006. Since that time, a small number of plant modifications with possible flooding implications have been identified and evaluated for PRA impact. Of these plant modifications, only the addition of a new closed loop containment cooling system called the Chilled Water System (CHW) with heat removal provided by Non-Essential Service Water (NESW) was identified as having potential flooding analysis implications. Qualitative evaluation of this plant change concluded that there would be minimal numerical impact to the PRA flooding model. The new CHW system was evaluated as having minimal flooding impact based on the limited water inventory in the system and the pipe routing through the Turbine Building. The new NESW system piping was evaluated as having minimal flooding impact based on the pipe routing through the Turbine Building.

The qualitative evaluation of the potential impact of the new NESW piping is readily shown to be reasonable on a quantitative basis. CNP's "Internal Flood Detailed Analysis" shows that floods caused by non-safety related piping ruptures in the Turbine Building flood areas result in Conditional Core Damage Probability (CCDP) values less than 1E-5. Based on the Jamali method piping failure rates and a gross over-estimate of the number of new piping sections, rupture of the new NESW piping routed through any individual flood area would be expected to add a flood initiator with a frequency no greater than 1E-3 per year. Accordingly, the new NESW piping would be quantitatively estimated as adding new flood initiating events of less than 1E-8/year to any of the flood areas through which it is routed.

Given the minimal numerical impact of the above plant changes from a PRA model perspective, no immediate update to the PRA model was performed. Further, as the quantitative estimate provided above shows, there have been no plant changes since the previous NRC approval of CNP's Risk-Informed Inservice Inspection Program (Reference 5) that have added piping segments leading to changes in CDF or LERF greater than 1E-6 per year or 1E-7 per year, respectively.

References

1. Letter from R. A. Hruby, Indiana Michigan Power Company (I&M), to Nuclear Regulatory Commission (NRC) Document Control Desk, "Fourth Ten-Year Interval Inservice Inspection Program Plan," AEP-NRC-2010-21, Accession Number ML100750680, dated March 12, 2010.

2. Electronic Communication from P. S. Tam, NRC, to H. L. Etheridge, I&M "D.C. Cook Units 1 and 2 – Revised Draft RAI on the 3/12/10 submittal re: the 4th 10-year interval ISI program (TAC ME4495 and ME4496)," Accession Number ML102850748, dated October 12, 2010.

3. Draft Addenda B to American Society of Mechanical Engineers, ASME RA Sa 2003, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated March 2005.

4. NEI 05-04, Revision 1 (Draft), "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard (Internal Events)," Nuclear Energy Institute, dated November 2007.

5. Letter from T. L. Tate, Nuclear Regulatory Commission, to M. K. Nazar, Indiana Michigan Power Company, "Donald C. Cook Nuclear Palnt, Units 1 and 2 – Risk-Informed Safety-Based Inservice Inspection Program for Class 1 and 2 Piping Welds," Accession Number ML072620553, dated September 28, 2007.