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

March 31, 2014

Mr. Lawrence J. Weber
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 – REQUEST FOR ADDITIONAL INFORMATION ON THE APPLICATION FOR AMENDMENT TO RESTORE NORMAL REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE CONSISTENT WITH PREVIOUSLY LICENSED CONDITIONS (TAC NO. MF2916)

Dear Mr. Weber:

By letter dated October 8, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13283A121), Indiana Michigan Power Company submitted an application for a license amendment to restore the normal reactor coolant system operating pressure and temperature consistent with previously licensed conditions for the Donald C. Cook Nuclear Plant, Unit 1.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject submittal and determined that additional information is needed to complete the review, as described in the enclosed request for additional information (RAI). The NRC staff had discussed the RAI in draft form with your staff on February 25, 2014. During that discussion your staff agreed to submit a response within 30 days of the date of this letter. As discussed during the conference call, this is the first of two RAI sets that will be issued by the NRC staff for this review. The second set is scheduled to be issued within the next few weeks.

Please feel free to contact me at (301) 415-4037 if you need any further clarification of the questions in the enclosure.

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosure:
Request for Additional Information

cc: Distribution via ListServ

REQUEST FOR ADDITIONAL INFORMATION

RESTORATION OF NORMAL OPERATING PRESSURE

AND NORMAL OPERATING TEMPERATURE

DONALD C. COOK NUCLEAR PLANT UNIT 1

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

By letter dated October 8, 2013 (Agencywide Documents Access and Management System (ADAMS), Accession No. ML13283A121), Indiana Michigan Power Company (the licensee) submitted for Nuclear Regulatory Commission (NRC) review and approval a license amendment request (LAR) to restore normal reactor coolant system (RCS) operating pressure and temperature, consistent with previous licensed conditions. To complete its review, the NRC staff requests the following additional information.

Reactor Systems Branch (SRXB)

SRXB RAI-1) Section 5.1.1, "Best-Estimate Large-Break LOCA [loss of coolant accident]," of WCAP-17762-NP indicates that the proposed normal operating pressure (NOP)/normal operating temperature (NOT) restoration was evaluated using the analysis of record (AOR), approved in 2008, as a baseline. The WCAP is clear that the AOR included the 571 °F value within its range of reactor coolant system (RCS) average temperature (T_{ave}); however, the hot full power RCS pressure is presented, from the AOR, at both 2100 and 2250 psia. The 2008 ASTRUM implementation LAR (ADAMS Accession No. ML080090268) also includes an allowance for both pressure bands (see Table 1 of Enclosure 2 to ASTRUM LAR), but it is not clear how the analysis accounts for these pressure bands.

1.a) Explain how the AOR accounts for the two pressure bands.

1.b) Explain whether the AOR peak clad temperature (PCT) case reflects the higher RCS pressure.

1.c) Explain why the AOR value provided in WCAP-17762-NP-A 3 (2128 °F) differs from that contained in the ASTRUM implementation LAR (2106 °F).

1.d) Explain whether the thermal conductivity degradation (TCD) estimate (ADAMS Accession No. ML12088A104) treated the RCS pressure consistently with the AOR and/or the WCAP.

SRXB RAI-2) Section 5.1.1, "Best-Estimate Large-Break LOCA," of WCAP-17762-NP, contains the following passage:

"Due to the non-linear effects of the design input changes (which were updated relative to the assessment reported in Reference 3 [ADAMS Accession No. ML12088A104 – estimated effects of thermal conductivity degradation {TCD}]), the return to NOP/NOT evaluation is being assessed against the Cook Unit 1 BE [best estimate] LB [large break] LOCA analysis of record (AOR), which was submitted in Reference 7 [ADAMS Accession No. ML080090268 – ASTRUM LAR] and approved by the USNRC ... Additionally, due to different cases becoming limiting at NOP/NOT conditions, the prior PCT assessment reported in Reference 10 [August 30, 2013, 30-day report of significant emergency core cooling system (ECCS) Evaluation Model error/change] is also re-considered ..."

The method of selecting limiting cases to determine the effect of a model change on the PCT prediction has been previously reviewed and accepted by the NRC staff; however, the method of identifying and analyzing the case sub-set is a topic of plant-specific review (see, for example, ADAMS Accession No. ML12173A025 – D.C. Cook Response to Request for Additional Information related to TCD estimate). Please provide information to enable NRC staff review of the case subset selection and validation process.

2.a) Provide a matrix of the significant sampled input parameters from the AOR and the various cases executed to estimate the effects of TCD, model changes and error corrections, and the restoration of NOP/NOT conditions.

2.b) Provide a summary of the case sub-set selection process: explain how the limiting cases were identified and what attributes were identified for the newly limiting cases.

2.c) Explain how the case sub-set selection method was validated, and how the results were verified to be limiting.

SRXB RAI-3) The licensee presents its evaluation of the NOP/NOT restoration with respect to the small break (SB) LOCA analysis in Section 5.1.2 of WCAP-17762-NP. The evaluation is based on an SBLOCA analysis that was provided to the commission by letter dated August 31, 2012 (ADAMS Accession No. ML12256A685). The succinct evaluation provided in WCAP-17762-NP concludes that the revised SBLOCA analysis explicitly accounts for the restored NOP/NOT conditions.

Noting that the August 2012, SBLOCA analysis was provided to the Commission, rather than submitted for review and approval, the NRC staff is reviewing the SBLOCA analysis as part of the NOP/NOT review effort to verify that it satisfies applicable regulatory requirements and confirm that it accounts for the proposed NOP/NOT operating conditions.

3.a) Figure 6 provides the core mixture level for the 3.25-inch (limiting) break. The figure shows that the core mixture level remains below 20 feet for a significant period of time (i.e., about 2500 seconds), despite that the PCT node is located at 11.75 feet (NRC staff infers that this elevation corresponds to approximately 21.8 feet on Figure 6). At the time of PCT, 1483 seconds, the hot node does not appear to be covered. The mixture level appears closer to 14.5 feet. Additionally, the rod film heat transfer coefficient depicted in Figure 15 shows that the coefficient is reasonably stable below approximately 50 BTU/hr/ft^2/°F, from 1000 through 3000 seconds of the transient. Furthermore, the accumulators begin to empty 200 seconds prior to time of PCT. Please explain how the PCT temperature excursion is being terminated and provide supporting tables and plots with additional data from the NOTRUMP and LOCTA runs.

3.b) Describe the loop seal clearing behavior depicted in Table 6 in greater detail. For all the breaks, provide the thermal-hydraulic conditions present in the intact loop seals. For the limiting break in particular, describe the reactor coolant conditions immediately prior to and following the loop seal clearing, especially with regard to the effect that the loop seal clearing has on the mixture level transient and system pressure.

3.c) The greatest fraction of pumped safety injection flows into the broken loop. Due to the large variation in liquid flow out the break throughout the duration of the transient, it is difficult to evaluate the broken loop flow behavior. Please provide detailed plots of liquid and vapor flow rates at the junctions or links connecting the broken loop to pumped safety injection sources, the break, the reactor coolant pump, and the vessel, for the first 2000 seconds of the limiting break. Include scaling appropriate prior to and following the loop seal clearing.

3.d) Describe the modeling of flow paths between the downcomer and the upper plenum and core barrel.

3.e) Provide plots of the hot assembly void fraction as a function of height for the limiting break at the time of minimum core level, and again at the time of PCT.

3.f) Provide the mass flow rate at the core exit as a function of time for the limiting break.

SRXB RAI-4) The D.C. Cook post-LOCA long term cooling (LTC) analyses demonstrate that boric acid concentration control measures are adequate, and that the ECCS recirculation flows "dilute the core and replace core boil-off, thus keeping the core quenched." WCAP 17762-NP refers to an analysis (ADAMS Accession No. ML11195A025), which is performed for D.C. Cook Unit 2.

4.a) Please explain how the calculation concludes that the ECCS recirculation flow is adequate.

4.b) Please address differences between the D.C. Cook Units.

4.c) The Unit 2 analysis states the following: "The current hot leg switchover time and plant operating procedures result in ECCS flows that temporarily drop below the injected flow necessary to replace core boil-off (plus entrainment) during the HLSO process."

4.c (i) Please explain what consequence, if any, the hot leg swapover evolution could have on maintaining a stable core quench.

4.c (ii) Please explain how this calculation accounts for entrainment.

4.d) Section 5.1.3 of WCAP-17762 does not appear to indicate, as other sections of the WCAP do, that the boric acid precipitation analysis reflects the Unit 1 NOP/NOT values. The WCAP states, "The inputs used to perform post-LOCA LTC analyses include core power levels, fuel dimensions, and RCS and ECCS volumes, temperatures, pressures, and boron concentrations." Explain whether, and how, these inputs are affected by the NOP/NOT restoration, and whether, and how, the analysis accounts for the NOP/NOT restoration.

SRXB RAI-5) Subsection 5.2.1, "Introduction and Background," to Section 5.2, "Non-LOCA Transients," discusses evaluations for events that take credit for the lower temperature/pressure, stating, "In particular were the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) setpoints, which utilized T' and T" values that were restricted below the full power T_{avg} primarily to provide overpower protection while maintaining the same ΔT setpoints." Subsection 5.2.3.2 discusses the Uncontrolled Rod Withdrawal at Power, and states, "Additionally, it was confirmed as part of the Return to RCS NOP/NOT Program that the OTΔT setpoints modeled in the current analysis remain valid at NOP/NOT conditions." Please explain how this confirmation was performed and provide additional detail regarding the results of the confirmation. In particular, explain whether the T' and T" values assumed in the Rod Withdrawal at Power analyses reflect the more restrictive values, and if so, how the setpoints remain valid for the proposed operating conditions.

SRXB RAI-6) The Steam Generator Tube Rupture Margin to Overfill (MTO) analysis discussed in Section 5.3.4 refers to NSAL 07-11. Please provide a copy of NSAL 07-11.

SRXB RAI-7) WCAP-17762-NP, Section 5.3.4.4, indicates that the present MTO analysis was re-assessed to address the proposed changes in RCS conditions, and to address the issues identified in NSAL 07-11. This analysis was incorporated into the Cook Unit 1 licensing basis via Amendment 256. The approving safety evaluation notes that the methods based on those described in WCAP-10698-P-A, along with the

LOFTTR2 code, were used to evaluate SGTR MTO. However, analytic assumptions were consistent with the Cook licensing basis and not necessarily the analysis approved in WCAP-10698-P-A.

7.a) The SE approving WCAP-10698-P-A states, "The design basis SGTR analysis assumes LOOP, the most reactive stuck rod, conservative initial conditions, safeguards capacities and setpoints, turbine runback, 120% of 1971 ANS decay heat rate, and the worst single failure." The NRC staff understands that these assumptions are not necessarily employed in the revised Cook MTO analysis. Explain which of these assumptions are applied to the Cook MTO analysis. Provide specific information regarding the "conservative initial conditions."

7.b) The Updated Final Safety Analysis Report indicates that the secondary volume of the replacement steam generators, which is smaller than that of the original steam generators, remains sufficient to accommodate the integrated leakage during the SGTR event. Explain whether the updated MTO analysis reflects the smaller volume of the replacement steam generators.

7.c) The MTO analysis approved in 2001 includes the assumption of time-critical operator actions. Explain what effect the NOP/NOT restoration will have on the time available to complete these actions.

7.d) The 2000-2001 NRC staff review of the MTO LAR included an assessment of the licensee's ability to execute the time-critical operator actions credited in the analysis. Provide an update to this assessment.

7.e) The 2000-2001 NRC staff review of the MTO LAR revealed that the analysis credited a substantial amount of non-safety grade equipment. For example, remote manual operation (i.e., using a switch in the control room) of SG PORVs is credited in the analysis. Nitrogen bottles are provided to ensure operability of the PORVs as added safety margin in the event of a coincident loss of offsite power. Explain whether the available bottled nitrogen supply would permit SG PORV operation for the duration of the analyzed MTO event. In addition, explain whether the SG PORV can be operated by local, manual action. If so, provide a quantitative estimate of the time required to execute a local PORV operation.

<u>Component Performance, Nondestructive Examination and Testing Branch (EPNB)</u>

EPNB RAI-1) Enclosure 7 to the October 8, 2013, submittal discusses the assessment of the impact of increased temperature and pressure on various systems. However, the assessment did not discuss any previously identified degradation that was evaluated and found acceptable under the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI flaw evaluation methods. Identify all evaluations that were completed using the lower temperature

and pressure and identify if these evaluations will be revised to take into account the higher temperature and pressure. If not, provide justification.

EPNB RAI-2) The examination frequency of control rod drive mechanism (CRDM) penetration nozzles is related to the temperature of the reactor vessel head and is based on the Effective Degradation Years (EDY) and Reinspection Years (RIY) calculations as specified in ASME Code Case N-729-1 as conditioned in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g)(6)(ii)(D). Discuss whether the lower temperature was used in calculating RIY and EDY parameters for the examination frequency of the CRDM penetration nozzles in the previous licensing actions. If yes, discuss whether the RIY and EDY calculations will be revised to take into account the higher temperature to obtain the examination frequency for the CRDM nozzles. If not, provide justification.

Please feel free to contact me at (301) 415-4037 if you need any further clarification of the questions in the enclosure.

Sincerely,

/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosure:
Request for Additional Information

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