October 2, 2002

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST, "POWER UPRATE MEASUREMENT UNCERTAINTY RECAPTURE," DATED JUNE 28, 2002 (TAC NO. MB5498)

Dear Mr. Bakken:

By application dated June 28, 2002, the Indiana Michigan Power Company (I&M) submitted a license amendment request that would revise the operating license and technical specifications for the Donald C. Cook Nuclear Plant, Unit 1, to allow the use of a more accurate flow measurement instrumentation and power calorimetric uncertainty value to allow the licensed core thermal power to be increased by 1.66 percent from 3250 megawatts thermal to 3304 megawatts thermal.

The Nuclear Regulatory Commission (NRC) staff has reviewed your June 28, 2002, application and concluded that it does not provide technical information in sufficient detail to enable the staff to make an independent assessment regarding the acceptability of the proposal in terms of regulatory requirements and the protection of public health and safety. Enclosed is the NRC staff's request for additional information (RAI).

The draft RAI was e-mailed to your staff to facilitate conference calls held September 13 and 19, 2002, with Mr. Gordon Arent, et. al., of your staff regarding the RAI. The enclosed RAI is similar in content to the draft RAI that was e-mailed to your staff. A mutually agreeable target date of October 11, 2002, for your response was established. The NRC staff will continue reviewing your application when your response to the enclosed RAI is received.

If circumstances result in the need to revise that target date, please contact me at (301) 415-1345.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosure: Request for Additional Information

cc w/encl: See next page

If circumstances result in the need to revise that target date, please contact me at (301) 415-1345.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
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Docket No. 50-315

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REQUEST FOR ADDITIONAL INFORMATION

2900 MEASUREMENT UNCERTAINTY RECAPTURE

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK, UNIT 1

DOCKET NO. 50-315

1. Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4 and NSAL 02-5, to document the problems with the Westinghouse designed steam generator water level setpoint uncertainties. NSAL 02-3 and its revision, issued on February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for steam generator water level measurements. These uncertainties affect the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, anticipated transient without scram (ATWS) and steamline break). NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation and affects the high-high level trip setpoint. NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the steam generator water level related safety analyses. The analyses may not be bounding because of velocity head effects or mid-deck plate differential pressures which have resulted in significant increases in the control system uncertainties. Discuss how D. C. Cook Unit 1 accounts for these uncertainties documented in these advisory letters in determining the steam generator water level setpoints. Also, discuss the effects of the water level uncertainties on the analyses of record for the loss-of-coolant accident (LOCA) and non-LOCA transients and the ATWS event, and verify that with consideration of all the water level uncertainties, that the current analyses are still adequate with regard to the power uprate.

2. Upon reviewing large-break LOCA models for power uprates, the Nuclear Regulatory Commission (NRC) recently found plants that require changes to their operating procedures because of inadequate hot leg switch-over times and boron precipitation modeling. Discuss how your analyses account for boric acid buildup during long-term core cooling and discuss how your predicted time to initiate hot leg injection corresponds to the times in your operating procedures.

3. Regulatory Issues Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," Section I.1.E, indicates that a calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty should be submitted with the uprate application. Please provide your plant-specific calculations for the total power measurement uncertainty at the plant.

4. RIS 2002-03, Section II.1.5, "Steam Generator Tube Rupture (SGTR) - Thermal-Hydraulic Analysis," indicates that a change in power has a negligible effect on the SGTR margin-to-overfill analysis. You further indicate that you performed a

sensitivity study to show that an SGTR occurring with a power level increase of 2 percent remains bounded by your "supplemental SGTR analysis." However, the supplemental SGTR analysis was performed at a nominal power level. Please describe the methodology of the sensitivity study to indicate why it remains bounded by the supplemental SGTR analysis performed at a lower power level.

5. In RIS 2002-03, Section IV.5.2, "Structural Integrity Evaluation," you state that "Mechanical repair hardware was not evaluated for the D. C. Cook Nuclear Plant Unit 1 steam generators because they are new replacements with no installed repair hardware and minimal tube plugging (less than 0.03 percent steam generator tube plugging (SGTP)." The NRC staff believes that the number of tube plugs currently installed in the steam generators is irrelevant and the licensee should evaluate the effect of the Measure Uncertainty Recapture (MUR) Power Uprate on the tube plugs.

6. In RIS 2002-03, Section IV.5.2, "Structural Integrity Evaluation," you state that "Results of the analyses performed on the BWI Series 51 steam generators show that all steam generators components continue to meet American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1989 Edition, limits for the 1.66 percent uprate conditions with the reactor coolant system (RCS) pressure at 2100 psia. The primary-to-secondary pressure differential remains below the design value of 1600 psid. For operation with the RCS at 2250 psia, the primary-to-secondary pressure differential remains below the design value of 1600 psid, provided the secondary side steam pressure is limited to 679 psia." Based on the last sentence, the NRC staff believes you are stating that ASME limits will not be met under the uprate conditions when the RCS pressure is 2250 psia and the secondary side steam pressure is not limited to 679 psia. The staff's understanding, based on their review of Table 3, Case 2, is that it is possible that the secondary side steam pressure may be as low as 618 psia. Based on this conclusion, explain why the power uprate conditions are acceptable. If the intent is to control the secondary side steam pressure such that it is limited to 679 psia, describe the vehicle under which this will be performed.

7. In RIS 2002-03, Section IV.5.3, "Tube Vibration and Wear," you state that "... the projected level of tube wear as a result of vibration would be expected to remain small, and will not result in unacceptable wear." Provide the NRC staff with additional details (e.g., actual possible increase in wear as a result of power uprate conditions). In addition, describe the basis used to conclude that "unacceptable wear" would not occur.

8. In RIS 2002-03, Section IV.5.4, "Regulatory Guide 1.121 Analysis," you state "The Regulatory Guide 1.121 analysis establishes minimum wall requirements for transient conditions corresponding to the 30 percent Steam Generator Tube Plugging (SGTP) case, which envelopes the primary-to-secondary pressure gradients for the 0 percent SGTP condition." In this analysis, is the assumed reactor coolant pressure 2250 psia (as seen in Table 3) or 2100 psia? State whether the assumed reactor coolant pressure bounds all possible pressures during operation (i.e., is the most bounding), and if not, analyze the bounding case and provide the results.

9. In RIS 2002-03, Section IV.5.3, "Tube Vibration and Wear," you describe the potential effects of the 1.66 percent MUR on steam generator tube wear. Discuss the potential

effects of the 1.66 percent MUR on other potential modes of steam generator tube degradation (e.g., axial and/or circumferential cracking, etc.).

10. Discuss the impact the power uprate will have on the required frequency of steam generator tube inspections.

11. RIS 2002-03, Section VII.6.4 discusses the Flow-Accelerated Corrosion (FAC) Program. The NRC staff has the following questions related to the FAC Program:

    - Briefly describe the purpose and elements of the FAC Program.

    - The submittal states that "Flowrates and temperatures for piping components within the scope of the FAC Program remain within the system design specifications." Explain how the system design specifications are used in the FAC Program (e.g., what decisions/assessments are made based on design flowrates and temperatures).

    - Identify the software utilized as part of the FAC Program to model the piping systems. Identify the source (e.g., design values, actual values, etc.) of the input parameters (e.g., operating pressures, flowrates and temperatures) to the software program.

    - The submittal indicates that no changes are required to the FAC Program, however, this appears to be related to the activities, elements and philosophy of the FAC Program. Please discuss whether any additional systems will need to be added to the FAC Program as a result of the power uprate. Also, discuss whether the power uprate will result in any changes to the software input parameters. If software input parameters will be affected, summarize the significance of the overall impact on the FAC Program activities.

12. Provide the details of how the power uprate will effect the steam dump system capabilities.

13. Provide the details of how the power uprate will effect air and hydraulic operated valves.

14. In the application, the pressure temperature curves on Figure 3.4-2 of the technical specifications (TS) have been changed to reflect the effects from the power uprate. By letter dated May 3, 2002, you stated the pressure temperature curves in the TS did not reflect the most limiting material. Please provide calculations and revised TS pages that reflect both the power uprate and the most limiting material.

15. In response to Item I.1.D of RIS 2002-03, Attachment 1, Submittal Attachment 3, cites WCAP-8567 for a description of "Improved Thermal Design Procedure" (ITDP) and states that the NRC has approved the use of ITDP at Cook Nuclear Plant Unit 1. The reference cited for this approval does not include NRC review and acceptance of this procedure for general use. Please clarify. Please describe the application of ITDP for the requested power uprate.

16. The "Sensitivity % Rated Thermal Power (RTP)" column of the submitted table appears to be mislabeled, and appears to represent the product of RTP sensitivity and the uncertainty in the various parameter measurements. The reference in that same table to "Root Mean Squared (RMS)" is interpreted to have been intended to mean "Square Root Sum of the Squares (SRSS)."

17. In your Submittal Attachment, 3 Section I.1.G/H, bullet 5 you state that failure of one plane of leading edge flow meter (LEFM) transducers would not affect power measurement, and cites Caldon Topical Report ER-157P and the associated Safety Evaluation Report (SER) as justification for this claim. The report and the associated SER do not support this claim. Loss of an entire detection plane in an LEFM CheckPlus system would render it functionally similar to an LEFM Check system. Such a reduced system would not be optimized for single-plane use, and so performance would likely fall short of a properly configured LEFM Check system. ER-157P clearly indicates a significant difference in accuracy between the Check and CheckPlus flowmeters. The topical report and SER indicate that continued operation without reduction in power with one LEFM CheckPlus <u>component</u> out of service <u>might</u> be justifiable, but leaves it up to the applicant to provide the justification. Please clarify.

18. The application refers to a serial link between the LEFM and the PPC (Plant Process Computer) and states that the venturi-based instrument will always be calibrated in accordance with the last "good" value from the LEFM. However, there is no discussion of the timing or operation of this link or of the calibration adjustment. Please provide a discussion of: (a) the nature and operation of the serial link, (b) the schedule by which the venturi-based flowmeter calibration is adjusted, (c) the method for adjusting the venturi flowmeter calibration, and (d) the means for distinguishing "good" from "bad" LEFM data for the purpose of calibration adjustment of the venturi meter.

19. If the plant computer or the plant computer serial link from the LEFM is not operational, automatic power calculations will not be performed. Please show that these conditions, and any other conditions that might interfere with automatic operation, are properly accounted for in the procedures and in appropriate limitations associated with the proposed modification. In particular, show that despite such conditions, the venturi-based flowmeter calibration will remain sufficiently correlated with the LEFM calibration to support continued operation above the pre-uprate power limit in the event of LEFM failure.

20. Please show that the time limit established for continued operation above the pre-uprate power limit with the LEFM out of service properly accounts for:

    a. decay of venturi-based flowmeter accuracy from the most recent LEFM-based calibration update to the time of LEFM failure,

    b. continued operation from the time of LEFM failure to the initiation of power runback, and

    c. continued operation during power runback, until the indicated power is at or below the pre-uprate power limit.

21. Please specify the time allowed from initiation of power runback until core power reaches the pre-uprate power limit, in the event of extended LEFM failure.

22. RIS 2002-03, Attachment 1, Section I.1.F requests certain information concerning all instrumentation involved in the power calorimetric. The licensee's response to this item addresses only the LEFM. Please provide the requested information for the remaining instrumentation.

23. Please confirm that the installation of the new flowmeter will not adversely affect the performance of the existing flow instrumentation.

24. Please clarify Item I.1.G/H of Attachment 3 of your application to Reference 1 to address operation at any power level in excess of the pre-uprate limit, not just exactly at the new limit.

25. Please provide a copy of the calculation that establishes the thermal power measurement uncertainty, as requested in Item I.1.E of Attachment 1, and reiterated and explained in Items I.2, I.6, and I.7 of Attachment 2, to RIS 2002-003. Item G.6 of RIS Attachment 2 also requests detailed information.

cc:

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