March 29, 2001

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION, “LICENSE AMENDMENT REQUEST FOR CONTROL ROOM HABITABILITY” (TAC NOS. MA9394 AND MA9395)

Dear Mr. Powers:

On June 12, 2000, as supplemented November 7, 2000, Indiana Michigan Power Company (I&M) submitted a proposed license amendment request that would revise the operating licenses for D. C. Cook Units 1 and 2 to use the methodology and the alternative source term in 10 CFR 50.67 as described in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants.”

The Nuclear Regulatory Commission (NRC) staff has reviewed your request 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.

The NRC staff finds that the additional information identified in the enclosure is needed.

Draft questions were provided to your staff on February 01, 2001, and were discussed with Mr. J. Waters, et al., of your staff on February 2, 2001, February 13, 2001, and February 27, 2001. The questions in the enclosure to this letter are the same as the draft questions. A mutually agreeable target date of June 30, 2001, for your response was established. The NRC staff will continue review of your amendment application when your response to the enclosed questions is received.

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

Sincerely,

/RA/

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

Docket Nos. 50-315 and 50-316

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION FOR

D. C. COOK UNITS 1 AND 2

SUBMITTALS C0600-13 AND C1100-10 (CONTROL ROOM HABITABILITY),

DATED JUNE 12, 2000, AND NOVEMBER 7, 2000

1) Requested Action 2 of generic letter (GL) 99-02 states, "If the system has a face velocity greater than 110 percent of 0.203 m/s [40 ft/min], then the revised technical specification (TS) should specify the face velocity."

   Please refer to or provide docketed information which indicates the <u>actual system face velocity and/or the actual residence time</u> for the control room emergency ventilation system (CREVS), engineered safety feature ventilation system (ESFVS), and storage pool ventilation system (SPVS) and describes how it is calculated for these systems.

   The actual system face velocities can be calculated by dividing the maximum accident condition system flow rates specified in the TS (nominal + typically 10 percent upper value) by the total exposed surface area of the charcoal filter media. (The guidance on calculation of the residence times in American Society of Mechanical Engineers (ASME) AG-1-1997, Division II, Sections FD and FE, Articles I-1000, or in American National Standards Institute (ANSI) N510-1975 can be used to calculate the actual system face velocities). It should be noted that the face velocity should be consistent with the bed depth and residence time. (Bed Depth = Face Velocity x Residence Time)

2) In order for the staff to verify that a safety factor as low as two is used, the staff needs to know the charcoal adsorber removal efficiencies which are credited in the <u>current and proposed</u> radiological accident analyses for organic iodide.

3) On page 19 of Attachment 1 to Letter C0600-13, it is stated that in case of CREVS the recent accident analyses assume 95 percent iodine removal efficiency for single-fan operation under normal system flow rate and 80 percent removal efficiency for two-fan operation at an increased face velocity during the first two hours of the accident. It is also stated that "...The 80 percent efficiency calculation includes a safety factor of two. To ensure the accident analysis assumptions remain valid for both single and two-fan operation, the surveillance requirement is revised to demonstrate a penetration of less than or equal to 1 percent when tested at normal system flow rate."

   (a) Clarify how at 80 percent filter efficiency the safety factor of two is calculated.

   (b) For two-fan operation, what is actual increased maximum face velocity across the charcoal bed.

   (c) Explain how 80 percent filter efficiency at increased face velocity compares with 95 percent filter efficiency at normal system flow rate.

(d) Demonstrate how the 1 percent penetration at normal system flow rate as the surveillance requirement bound both single and two-fan operation cases.

4) For accidents where the CREVS is not operated in the emergency mode, provide the bases for the assumption of only 1000 cfm of unfiltered makeup since there is no indication that other sources of unfiltered inleakage are considered.

5) For accidents where the CREVS is in the emergency lineup, your submittal assumes 98 cfm of unfiltered inleakage. Please clarify why the 98 cfm of unfiltered inleakage for Unit 2 is limiting following the damper repair in Unit 1. It is not clear how the 98 scfm due to damper repair in Unit 1 was obtained.

6) On page B 3/4 7-4a of your submittal, operability is defined by maintaining a positive pressure of greater than or equal to 1/16 inch water gauge relative to the outside atmosphere. However, industry test results have determined that pressurization (at any level i.e. 1/16, 1/8, etc.) does not demonstrate control room envelope/pressure boundary operability.

a) Provide justification for your proposed TS changes defining control room envelope/pressure boundary operability based on 1/16 inch water gauge pressure relative to the outside atmosphere.

b) The requested 24-hour allowed outage time (AOT) is tied to the definition of control room envelope/pressure boundary operability. In order for the Nuclear Regulatory Commission (NRC) staff to find the request for a 24-hour AOT acceptable, the request must be in accordance with the Technical Specification Task Force-287 (TSTF-287), which has been generically approved by the staff. Note, TSTF-287 does not include a definition of control room boundary integrity.

7) In numerous locations, your submittal references NUREG-1465 and Draft Guide-1081 as basis for your submittal. Please provide a commitment to the applicable provisions of Regulatory Guide (RG) 1.183, in lieu of the NUREG-1465 and DG-1081 referenced in your submittal, identifying proposed alternatives, if any, for staff consideration.

(The staff used some information from NUREG-1465 as part of the basis for the development of the regulatory guidance in DG-1081 and the final RG 1.183. However, the staff has not endorsed NUREG-1465 for use by currently licensed power reactors since NUREG-1465 is not specifically applicable to currently licensed power reactors, especially those with fuel burnups in excess of 40 GWD/MTU. It is the staff's intent that the guidance of RG 1.183 be used by licensees in preparing their initial application under 10 CFR 50.67 and that guidance, less any approved alternatives, would become the facility's alternate source term (AST) design-basis.)

8) DG-1081 was published for public comment in December 1999, and the final guide RG-1.183 was issued in July 2000. Your submittal was dated June 2000. In addressing the public comments and preparing the final guide, several analysis assumptions in DG-1081 were revised. As such, some assumptions identified in your submittal differ from those deemed acceptable in RG 1.183. For many of these differences, the staff believes that your submitted analyses could be shown to be bounding using the

outdated assumption, and as such, it may be possible to incorporate the updated assumption in your design-basis without resubmitting the analysis. Please compare your analysis assumptions against those provided in RG 1.183 and indicate your intent to either update the assumption or retain the assumption as a proposed alternative to RG 1.183. Provide a justification for each such proposed alternative.

9) Your analyses incorporated revised atmospheric dispersion (X/Q) values calculated using the ARCON96 computer code. The staff considers this to be a change in analysis methodology requiring staff approval. Please provide sufficient information for the staff to evaluate the acceptability of your X/Q values. The information should include:

(a) Confirmation that the meteorological data input to ARCON96 was collected by the site's meteorological instrumentation as described in the updated final safety analysis report (UFSAR) or T/S and subject to 10 CFR Part 50, Appendix B quality assurance requirements.

(b) Unit 1 and Unit 2 release point and receptor configuration information (e.g., height, velocity, distances, direction, etc.), release mode (e.g., ground, elevated, surface), and meteorological sensor configuration, as input to ARCON96.

(c) A floppy disk containing the meteorological data input to ARCON96, in the ARCON96 input data format.

10) Your analyses incorporated an iodine flashing fraction of 10^{-4} for emergency core cooling system (ECCS) leakage, contrary to the default 10^{-1} assumption provided in RG 1.183. On Pages 5 and 6 of Attachment 1 to your submittal, you attempted to justify these assumptions on an experiment reported in your existing UFSAR, and on theoretical iodine partitioning of 10^{-8}. The staff does not believe that the provided justification supports the use of 10^{-4} for the ECCS flash fraction. Based on the description of the experiment, the staff questions whether the experimental drying to evaporation can appropriately model leakage that could be sprayed from the leakage paths, or as droplets fall through air and impinge on nearby surfaces. The staff also questions how well Eggleton's mathematical treatment of steady state vapor partial pressures between the gas and liquid phases can adequately model the more dynamic situation associated with leakage from pressurized systems as is the case here. Your submittal quoted partitioning of 10^{-8} which appears to be at odds with the abstract for Eggleton work which reports partitioning values ranging from 0.012 at high iodine concentrations and low pH to less than 0.0001 at high pH and low iodine concentrations. Please provide additional justification, including consideration of sump pH and area ventilation rates and iodine entrainment in evaporated vapor, in support of your assumption.

11) Your analyses addresses a small break loss-of-coolant accident (LOCA) event in which containment sprays do not start or are terminated early. Page 11 of 30 of DIT-B-00069-06 contains a note that states:

Per DG-1081 Appendix A, gap fractions from Table 3 can be used for small-break loss-of-coolant accident (SBLOCA) if no fuel melt is projected.

While this provision may have been present in a pre-decisional version of the draft guide, this provision was not included in the draft guide published for public comment in December 1999, nor in the final regulatory guide published in July 2000. While the staff agrees with the conclusion that the fuel damage could be less than that assumed for a large-break LOCA, the staff expects the licensee to provide a technical justification for the amount of fuel damage being assumed. Please provide an acceptable basis for this conclusion. See §3.6 of RG 1.183.

12) On Page 7 of Attachment 1, you note your conclusion that the assumption of a constant break flow for 30 minutes is more limiting than using the actual operator response times. Although this assumption may be valid with regard to mass of reactor coolant system (RCS) transferred to the secondary, what is the sensitivity of other analysis parameters to delays in operator actions, such as break flow flashing fraction, steam release from the affected steam generator, and tube uncovery? The staff is concerned that these other parameters, and the time-dependent buildup of RCS activity due to iodine spiking, could negate the apparent conservatism in the RCS mass transferred. Please confirm your conclusion relative to the postulated dose to the control room operators. Please explain how your amendment request dated October 24, 2000, on steam generator tube rupture (SGTR) analysis methodology affects this control room amendment request.

13) Contrary to the guidance of RG 1.183, in some of your analyses you have assumed an iodine spike duration of 6 hours based on the depletion of the 12 percent iodine gap inventory. The iodine spiking phenomenon is generally understood to be the result of RCS liquid flushing out suspended iodine salts from the fuel rod via pin hole leakage. The transfer of iodine from the pellet to the plenum region is dependent, in part, on partial pressures of iodine in the gap and the pellet. In light of these considerations, please explain why basing your assumption on the gap inventory alone is appropriate.

14) §3.1.1 of Attachment 6, identifies the assumption that 3 percent of the gap activity is released from 30 seconds to 90 seconds and the remaining 2 percent of the gap is released over the next 28.5 minutes. RG 1.183 (and DG-1081) provided that the activity would be released from the core in a linear fashion over the duration of the release phase, or as an alternative, released instantaneously at the start of the particular release phase. Please provide a justification for this proposed alternative from RG 1.183.

15) §3.1.4 of Attachment 6, identifies that the sedimentation removal coefficient is conservatively assumed to be only 0.1 hr^{-1} and that sedimentation does not continue beyond a decontamination factor (DF) of 1000. Please justify the conservatism of these two assumptions against the DFs presented in Table 20 of NUREG/CR-6189, “A simplified Model of Aerosol Removal by Natural Processes in Reactor Containments,” and the effective decontamination coefficients presented in Table 24 of the same document.

16) For the analyses that have credited iodine partitioning in the steam generators, was the impact of steam generator tube uncovery during the transient considered? Was this considered in determining the flash fraction? If not, why not?

17) The 3rd and 4th paragraphs on page 27 of Attachment 6, appear to be addressing the same plant response but with different nomenclature. As we understand the system operation, the control room ventilation systems re-align on a safety injection signal, not a containment isolation signal as implied in the 3rd paragraph. Please confirm that the control room re-alignment occurs on an safety injection (SI) signal (e.g., low pressurizer pressure, low steamline pressure, high containment pressure, etc.).

18) Items L43 and L44 in DIT-B-00069-06 identifies spray coverage for the three regions in the containment. This parameter was not addressed in the Attachment 6 discussion and was not tabulated in Table 11 of Attachment 6. Please describe how the spray coverage was incorporated into the analysis.

19) The staff has reviewed the information in Attachment 7 to your submittal. Item 6 on page 3 of this attachment addressed an issue related to design controls on changes made in the control room flow rates between 1982 and 1986, and whether or not the consequences of these changes were adequately evaluated. While your current re-analyses using the AST demonstrate compliance with GDC-19 (as revised in late 1999) this conclusion may not be applicable to the issue cited in 1986 since the source term and acceptance criterion were different. The staff expects to approve the current amendment request without accepting this item. Please indicate if you are requesting the NRC review and approval of the changes made to the control room flow rates between 1982 and 1986.

20) Please provide a description of the SBLOCA T/H analysis that was performed for determining the source term. Please include a summary of and justification for the initial assumptions used, the sequence of events, the criteria used for determining fuel pin failures and/or fuel melting, the technical basis supporting the decision criteria, and the results of the analysis from the standpoint of justifying the analysis as limiting with respect to source term.

21) The current licensing bases for D. C. Cook Units 1 and 2, use departure from nucleate boiling ratio (DNBR) as the criterion for determining the degree of fuel damage resulting from a locked rotor event. The licensee has not submitted either a request to modify its licensing basis or sufficient justification to demonstrate that the use of the 2700 °F criterion is appropriate. We note that the staff has not accepted the use of the 2700 °F criterion at other plants and further that the staff continues to believe that the DNBR criterion is the appropriate criterion for determining the amount of fuel failure. If you choose to use a criterion other than DNBR, please provide the technical justification for that criterion. Also, the description provided for the locked rotor event indicates that no pins exceed the DNBR limit. However, the description of the analysis does not include sufficient information for the staff to conduct its review. Therefore, please provide a description of the analysis for the locked rotor event. Please include a summary of and justification for the initial assumptions used, the sequence of events, the criteria used for determining fuel pin failures and/or fuel melting, the technical basis supporting the decision criteria, and the results of the analysis from the standpoint of justifying the analysis as limiting with respect to source term.