June 5, 2008

Mr. Michael W. Rencheck
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: D. C. COOK NUCLEAR PLANT, UNIT 1 (DCCNP-1) - REQUEST FOR ADDITIONAL INFORMATION, REGARDING RE-ANALYSIS OF LARGE-BREAK LOSS-OF-COOLANT ACCIDENT (TAC NO. MD7556)

Dear Mr. Rencheck:

In a letter dated December 27, 2007, Indiana Michigan Power Company (I&M) requested an amendment to modify Technical Specification (TS) 3.4.1, "RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.5, "Core Operating Limits Report (COLR)," in support of a new analysis of a large-break loss-of-coolant accident (LBLOCA).

The Nuclear Regulatory Commission (NRC) staff reviewed the December 27, 2007, submittal and issued a draft Request for Additional Information (RAI) on April 10, 2008 (Accession No. ML081010615). On May 15, 2008, the NRC staff discussed the draft RAI with I&M staff, and participants agreed that the draft RAI can be formally issued. Accordingly, the finalized RAI is issued as enclosure to this letter.

In accordance with the timeliness provision of 10 CFR 2.108, please respond within 30 days of receipt of this letter. Feel free to contact me if you need clarification of this RAI.

Sincerely,

/RA/

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosure: As stated

cc: See next page

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REQUEST FOR ADDITIONAL INFORMATION

D.C. COOK NUCLEAR PLANT, UNIT 1 (DCCNP-1)

RE-ANALYSIS OF LARGE-BREAK LOSS-OF-COOLANT ACCIDENT (LBLOCA)

(1) To show that the referenced generically approved ASTRUM LOCA analysis methodology applies specifically to DCCNP-1, please provide a statement that Indiana Michigan Power and its vendor (Westinghouse) have ongoing processes which assure that the ranges and values of the input parameters for DCCNP-1 LOCA analyses bound the ranges and values of the as-operated plant parameters.

(2) The discussion in the December 27, 2007, submittal did not address the effects of the mixed core on peak cladding temperature (PCT) and oxidation for the pre-resident fuel, but it does seem to address the PCT and oxidation for the new fuel. Please clarify if this is a whole core reload. In its Rulemaking Hearing dated December 28, 1983, the Nuclear Regulatory Commission stated, regarding the performance criteria of 10 CFR 50.46 (b): "In view of the lack of experience in this hypothetical situation, we think it prudent to apply our criteria to all of the core and not to exempt any part."

Please address PCT and oxidation results for "pre-resident" fuel in the core, if any.

[Note: In a letter to NEI dated November 8, 1999, G. M. Holahan, reiterated the NRC position that "total oxidation" encompasses accident and pre-accident oxidation. This position continues to apply. Therefore, in response to this question, please provide total oxidation for the "other" (pre-resident) fuel (if any), including pre-accident oxidation, plus LOCA cladding outside oxidation, plus cladding inside oxidation.]

(3) Please verify that the treatment of the vessel wall (radial noding, etc.) during reflood remains as historically approved in addressing the issue of downcomer boiling.

Donald C. Cook Nuclear Plant, Units 1 and 2
cc:

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