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

February 16, 2012

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

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - INFORMATION REQUEST PURSUANT TO 10 CFR 50.54(f) RELATED TO THE ESTIMATED EFFECT ON PEAK CLADDING TEMPERATURE RESULTING FROM THERMAL CONDUCTIVITY DEGRADATION IN THE WESTINGHOUSE-FURNISHED REALISTIC EMERGENCY CORE COOLING SYSTEM EVALUATION (TAC NO. M99899)

Dear Mr. Weber:

This letter is being issued in accordance with the U.S. Nuclear Regulatory Commission's (NRC's) regulation in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.54(f). Pursuant to this regulation, Indiana Michigan Power Company is required to provide information regarding the effect of a potentially significant error, as defined in 10 CFR 50.46(a)(3)(i), associated with thermal conductivity degradation (TCD), on peak cladding temperature in the Westinghouse Electric Company (Westinghouse)-furnished realistic emergency core cooling system (ECCS) evaluation models, to enable the NRC staff to determine whether the Donald C. Cook Nuclear Plant, Units 1 and 2, licenses should be modified, suspended, or revoked.

The NRC staff issued Information Notice (IN) 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," on December 13, 2011. This IN identified an error in the Westinghouse-furnished realistic ECCS evaluation models, the estimated effect of which was potentially significant, as defined in 10 CFR 50.46(a)(3)(i), in plant-specific applications. According to 10 CFR 50.46(a)(3)(i), "a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50 °F [degrees Fahrenheit] from the temperature calculated for the limiting transient using the last acceptable model..."

Indiana Michigan Power Company has received approval to implement Westinghouse realistic ECCS evaluation models as follows:

- Donald C. Cook Nuclear Plant, Unit 1, received approval to implement ASTRUM by Amendment No. 306 dated October 18, 2008.

- Donald C. Cook Nuclear Plant, Unit 2, received approval to implement ASTRUM by Amendment No. 297 dated March 31, 2011.To date, the NRC does not have information from Indiana Michigan Power Company estimating the effect of the TCD error on the Donald C. Cook Nuclear Plant, Units 1 and 2, peak cladding temperature calculated for the limiting transient using the last acceptable ECCS evaluation model.

The regulation at 10 CFR 50.46(a)(1)(i) requires the identification and assessment of uncertainties in the analysis method and inputs so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for so that when the calculated ECCS cooling performance is compared to the criteria set forth in 10 CFR 50.46(b), there is a high level of probability that the criteria would not be exceeded.

The information obtained from Westinghouse, and discussed in IN 2011-21, indicates that the uncertainty assessments in the Westinghouse realistic ECCS evaluation models are in error, because they do not realistically model the effects of thermal conductivity degradation.

The current analyses of record indicate the following:

- At Donald C. Cook Nuclear Plant, Unit 1, the predicted peak cladding temperature is 2128 °F..

- At Donald C. Cook Nuclear Plant, Unit 2, the predicted peak cladding temperature is 2107 °F..

Based on the proximity of these peak cladding temperatures to the regulatory limit of 2200 °F., and in combination with the information obtained by the NRC to date, the NRC staff is currently unable to verify that there remains a high probability that the 2200 °F. acceptance criterion would not be exceeded, consistent with the regulation at 10 CFR 50.46(a)(1)(i).

Therefore, further information is needed so that the NRC can verify that the Donald C. Cook Nuclear Plant, Units 1 and 2, ECCS evaluations are consistent with the 10 CFR 50.46 analysis and reporting requirements.

In accordance with 10 CFR 50.54(f), this information is required to "verify licensee compliance with the current licensing basis," which includes the applicable requirements contained in 10 CFR 50.46(a)(1)(i). Specifically, the information sought will be used to ensure that, once corrected for TCD, the realistic ECCS evaluations demonstrate, with a high level of probability, that the 10 CFR 50.46(b)(1) acceptance criterion concerning peak fuel cladding temperature, would not be exceeded.

Additionally, 10 CFR 50.46(a)(2), states that the Director, NRR, may impose restrictions on operation if ECCS evaluations submitted are inconsistent with the requirements of 10 CFR 50.46(a)(1)(i). The information sought by the NRC will enable it to determine whether your license should be modified as permitted by 10 CFR 50.46(a)(2).

Accordingly, pursuant to sections 161c, 161o, 182a and 186 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 50.54(f), in order for the Commission to determine whether your license should be modified, suspended, or revoked, you are required to provide information within 30 days of the date of this information request.

In your response, you shall address the following specific issues:

(1) An estimation of the effect of the thermal conductivity degradation error on the peak fuel cladding temperature calculation for the emergency core cooling system evaluations at Donald C. Cook Nuclear Plant, Units 1 and 2.

(2) A description of the methodology and assumptions used to determine the estimates. This description shall include consideration of experimental data relevant to thermal conductivity degradation and specific information regarding any computer code model changes which were necessary to address these data.

Indiana Michigan Power Company's responses should provide sufficient detail to allow the NRC staff to determine whether, consistent with 10 CFR 50.46(a)(1)(i), there remains a high level of probability that the acceptance criterion at 10 CFR 50.46(b)(1), concerning the peak fuel cladding temperature, would not be exceeded when the model is corrected for TCD.

This request is covered by the Office of Management and Budget (OMB) clearance number 3150-0011, which expires October 21, 2014. The estimated reporting burden for this collection of information is 72 hours. This estimate includes the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, performing necessary analyses, and completing and review in the collection of information. Send comments on any aspect of this information collection, including suggestions for reducing the burden, to the Records and FOIA/Privacy Services Branch (T5-F52), the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet electronic mail to infocollects@nrc.gov, and to Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

In accordance with 10 CFR 2.390, "public inspections, exemptions, and requests for withholding," a copy of this letter and your response will be made available for inspection and copying at the NRC Website at www.nrc.gov, and /or at the NRC Public Document Room. If you believe that any of the information to be submitted meets the criteria in 10 CFR 2.390 for withholding from public disclosure, you must include sufficient information, as required by the subsection, to support such a determination.

Please address the required written response to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, MD 20852, under oath and affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, please submit a copy of the response to the Director of NRR.

After reviewing your response, the NRC will determine whether further action is necessary in accordance with 10 CFR 50.46(a)(2) to ensure compliance with regulatory requirements.

If you have any questions on this matter, please contact Peter Tam at 301-415-1451.

Sincerely,

Allen G. Howe for

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

cc: ListServ

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***/RA by A. Howe for/***

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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cc: ListServ

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