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

September 22, 2009

Mr. Joseph N. Jensen
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 2 – REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE LICENSE AMENDMENT REQUEST ASSOCIATED WITH THE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY (TAC NO. ME1017)

Dear Mr. Jensen:

By letter dated March 19, 2009, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090930453), Indiana Michigan (I&M) Power Company (the licensee) submitted a license amendment request (LAR) for the Donald C. Cook Nuclear Plant, Unit 2 (CNP-2). The proposed amendment would adopt a new large-break loss-of-coolant accident (LBLOCA) analysis for CNP-2. The new analysis is performed using the U.S. Nuclear Regulatory Commission (NRC)-approved methodology set forth in Westinghouse Topical Report WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)." I&M proposes to endorse this methodology by a revision to Technical Specification (TS) 5.6.5, "Core Operating Limits Report (COLR)."

The proposed amendment would also revise TS 3.4.1, "RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," to change the minimum RCS total flow rate from 366,400 to 354,000 gallons per minute.

The proposed amendment would also revise TS 3.5.2, "ECCS – Operating," Condition D, to allow the unit to be in Mode 1, 2, or 3 for an unlimited amount of time if a Safety Injection (SI) system cross-tie valve is closed, provided that thermal power is reduced to less than or equal to a specified value. The new LBLOCA analysis being proposed does not address a condition in which an SI cross-tie valve is closed. Therefore, the allowance provided by Condition D will be deleted, as well as reference to Condition D in TS 3.5.2, Conditions A and C.

The NRC staff in the Reactor Systems Branch has reviewed the LAR and determined that additional information is required in order to complete the review. The requested information is provided as an enclosure.

In a telephone conversation on September 17, 2009, Mr. Michael Scarpello committed to provide the requested information 60 days from the date of this letter. Please contact me immediately if it is subsequently determined that the submittal date cannot be met.

If you have any questions, please call me at (301) 415-3049.

Sincerely,

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-316

Enclosure:
As stated

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST RELATED TO THE

LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY

DONALD C. COOK NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-316

TAC NO. ME1017

The Reactor Systems Branch staff in the Office of Nuclear Reactor Regulation has reviewed the Donald C. Cook Nuclear Plant, Unit 2 (CNP-2) license amendment request (LAR) regarding large-break loss-of-coolant accident (LBLOCA) analysis methodology and identified areas to be clarified by the Indiana Michigan Power Company (I&M).

1. With respect to proposed change to Technical Specification (TS) 3.4.1, the current value for minimum reactor coolant system (RCS) total flow specified in the TS 3.4.1 Limiting Condition for Operations and Surveillance Requirements is 366,400 gallons per minute (gpm). The proposed LAR stated that the value of 366,400 gpm is a minimum measured flow value which includes allowances for flow measurement uncertainty. Therefore, based on U.S. Nuclear Regulatory Commission (NRC)-approved method WCAP-16009-P, the proposed change to TS 3.4.1 will use so-called current practice of the thermal design flow value of 354,000 gpm. It further states that the proposed change will not affect the 354,000 gpm value used in the current and the new LBLOCA analyses.

   Please provide the following:

   (a) A description to explain that the proposed term of the thermal design flow value is a common industry practice and identify applicable examples currently used in U.S. nuclear power plants;

   (b) Clarification that the proposed change will not affect the 354,000 gpm value used in the current and the new LBLOCA analyses;

   (c) A detailed assessment that a 3.4 percent reduction of the RCS total flow from current value of 366,400 gpm to proposed value of 354,000 gpm will not reduce plant operation safety margin during a LOCA, even considering an accurate flow measurement uncertainty, an uncertainty always exists; and

   (d) The real minimum RCS flow used in the LBLOCA analysis.

2. The current TS 3.5.2 Actions include a Condition D that allows the unit to be in Mode 1, 2, or 3 for an unlimited amount of time if a Safety Injection (SI) system cross-tie valve is closed, provided that thermal power is reduced to less than or equal to a specified value. It further states that this allowance is justified by the current LBLOCA and SBLOCA analyses. However, the proposed new LBLOCA analysis does not include a condition in which an SI subsystem cross-tie valve is closed. Therefore, I&M is proposing that Condition D be deleted from the TS 3.5.2 Actions, and reference to Condition D deleted from Condition A and Condition C.

Please provide the following:

(a) The rationale to delete Condition D which directly provides an action against a situation that an SI system cross-tie valve is closed;

(b) The action(s) to be taken if an SI system cross-tie valve is closed;

(c) A description of which allowance is justified by the current LBLOCA and SBLOCA analyses, and its relationship with the proposed deletion of Condition D.

3. Please provide a description and the results of the evaluation completed against the conditions and limitations stated in the staff's safety evaluation report on the ASTRUM methodology in WCAP-16009-P-A with respect to the CNP-2 plant-specific adaptation of the ASTRUM methodology. Identify any deviations and their safety impact on the plant operations.

4. Please describe the reason why higher peak centerline temperatures shown in Figure 1 fall in the range of CD * Abreak/ACL between 1 and 2.5.

5. Please describe the physical meaning and cause with respect to a negative vapor flow rate as shown in Figures 7 and 8 between 20 and 40 seconds after break.

6. (a) Please provide the revision number and/or approval date for the topical report (i.e., WCAP-16009-P-A) referenced in proposed TS 5.6.5.b.4.

   (b) Please provide reference to the NRC letter approving the use of the plant-specific adaption of the topical report listed in proposed TS 5.6.5.b.4.

   (c) Please ensure that the information being provided in 6(a) and 6(b) above, is incorporated in proposed TS 5.6.5.b.4.

If you have any questions, please call me at (301) 415-3049.

Sincerely,

/RA/

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-316

Enclosure:
As stated

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