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

March 27, 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 1 (CNP-1) – REVIEW OF THE SMALL BREAK LOSS-OF-COOLANT ACCIDENT (LOCA) EVALUATION MODEL REANALYSIS SUBMITTAL (TAC NO. MD5297)

Dear Mr. Jensen:

By letter dated March 29, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071000431), Indiana Michigan Power Company (I&M, the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC) the limiting case small break LOCA reanalysis for CNP-1 pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46(a)(3)(ii).

The NRC staff required additional information to complete its review. I&M subsequently submitted responses in letters dated February 29, 2008 and December 16, 2008 (ADAMS Accession Nos. ML080740053 and ML083660104, respectively). On February 18, 2009, the NRC staff conducted an audit at the Westinghouse office in Rockville, Maryland, to further review the small break LOCA spectrum analyses results.

The NRC staff found that the proposed small break LOCA reanalysis for CNP-1 was performed in accordance with NRC-approved methods and analyses. The reanalysis indicated acceptable emergency core cooling system performance at a power level of 3304 megawatts thermal power, including the 1.34 percent uncertainty and a total peaking factor at 2.32, and the results were in compliance with the requirements of 10 CFR 50.46. The results of the staff review are provided as an enclosure to this letter.

Should you have any questions, please contact 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. 50-315

Enclosure:
Safety Evaluation

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**NUCLEAR REGULATORY COMMISSION**
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SMALL BREAK LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL REANALYSIS

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

## 1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) staff reviewed the small break loss-of-coolant accident (LOCA) reanalysis for the Donald C. Cook Nuclear Plant, Unit 1 (CNP-1), submitted by Indiana Michigan Power Company (I&M, the licensee) in a letter dated March 29, 2007 (Reference 1). The analysis employed the NRC-approved NOTRUMP code for evaluations of emergency core cooling system (ECCS) performance following small break LOCAs. The licensee submitted a reanalysis of the small break spectrum using the NOTRUMP code to correct an error which impacted the limiting small break LOCA analysis for CNP-1. The NRC staff's review of the reanalysis is presented below.

## 2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46 (a)(3)(ii) requires licensees to report errors in the LOCA analyses and estimate its effect on the limiting break.

Pursuant to 10 CFR 50.46 (a)(3)(ii) a reanalysis of the break spectrum was performed, including the limiting break for the CNP-1.

The small break reanalysis showed that the worst break peak cladding temperature (PCT) was 1725°F with a peak local cladding oxidation of 3.61%, which are within 10 CFR 50.46 PCT and cladding oxidation limits of 2200°F and 17%, respectively.

## 3.0 TECHNICAL EVALUATION

The purpose of the CNP-1 reanalysis of the small break LOCA ECCS response was to evaluate the impact of the isolation of the high head safety injection cross-tie. To assess the impact of the isolated cross-tie, the entire break spectrum was re-analyzed with the NRC-approved NOTRUMP code. The break sizes evaluated included the following: 1.5, 2.0, 2.5, 2.75, 3.0, 3.25, 3.5, 3.75, 4.0, 6.0, and 8.75-inch diameter break sizes. The analysis was performed at 3304 MWt (megawatts thermal) power, including the 1.34% uncertainty. The hot rod peaking factor was 2.32. The analysis was performed in accordance with the NRC-approved

Enclosure

NOTRUMP code and analysis methods and practices. The worst break was found to be the 3.25-inch diameter break with a PCT of 1725°F, a peak local cladding oxidation of 3.61%, and a core wide oxidation of less than 1%.

During the review, the NRC staff was concerned with the behavior of the core mixture levels for the breaks analyzed with the NOTRUMP code. In particular, the limiting 3.25-inch diameter break displayed a constant mixture during uncovery from about 800 to 1000 seconds into the event. The constant mixture level during the boil-off period was not explained or properly understood during the licensee's response to two rounds of requests for additional information (RAI) (References 2 and 3). Behavior of the core two-phase mixture level during periods of uncovery for small breaks of this size and character should show a steadily decreasing core uncovery until the reactor coolant system has depressurized sufficiently to allow the increased high pressure safety injection flow to match and then exceed the core steaming or boil-off rate.

The reactor pressure vessel is a simple boiling pot; thus, both fluid levels in the vessel downcomer and inner vessel region should display a decreasing trend throughout the 800 to 1000 second time period.

The NRC staff requested an explanation for the mixture level behavior. Of particular interest to the NRC staff was why the 3.25-inch diameter break displayed a constant mixture during uncovery from about 800 to 1000 seconds. A second round of RAIs was issued in an attempt to arrive at a physical explanation for the level behavior. The NRC staff questioned the licensee's reason for the level behavior being due to drainage of liquid that was held up earlier in the event in the steam generator active tubes. The core exit mass flow rate showed only positive flow exiting the core during the 800 to 1000 second time period.

The NRC staff found the explanations for this behavior provided in the RAI responses (Reference 3) to be insufficient. To determine a reasonable explanation for the mixture level behavior, the NRC staff suggested an audit meeting be held with the Westinghouse staff to review the details of the analysis results and the NOTRUMP code output. The results of the audit review identified the physical behavior causing the mixture level behavior for the 3.25-inch diameter break size. A summary of the audit and results are given below.

<u>Audit Summary</u>

On February 18, 2009, the NRC staff met with the representatives from Westinghouse and the Donald C Cook Nuclear Plant at the Westinghouse office in Rockville, Maryland, to review the results of the NOTRUMP small break LOCA break spectrum analyses. The audit was motivated by the need to fully understand the mixture level behavior for the breaks that displayed core uncovery and, in particular, the 3.25-inch diameter break which was limiting for the spectrum. Specifically, an explanation was needed to resolve the constant mixture level during the boil-off period which was not properly understood during the RAI process.

Detailed investigations into the system response for the 3.25-inch diameter break revealed that the two-phase level recession in the core was slowed due to flashing (as a result of depressurization) and the large void generation in the lower plenum which caused fluid expansion and a resulting surge of two-phase fluid into the core. Plots of the downcomer fluid level, core mixture fluid level, and lower plenum voiding showed that the large increase in void in the lower plenum occurred during the 800 to 1000 second time frame for this break. This voiding terminated the decrease in the mixture level at this time, which increased in the core region. The increased core level produced an increase in steaming and a small increase in the upper plenum pressure, which also pushed liquid back into the downcomer momentarily (or at least slowed the loss of fluid from the downcomer which allowed the ECCS to accumulate in the downcomer at this time). This behavior provided a valid explanation for the mixture level behavior predicted by NOTRUMP for this limiting break.

The staff noted that there was also a large momentary spike/increase in core inlet flow during this period, which may have been exaggerated by lack of a completely stable numerical solution for a brief time during the 800 to 1000 second time frame. The staff expressed these concerns to Westinghouse. Westinghouse performed an additional calculation with a change to the drift-flux model (unofficial calculation) which causes hold-up of the fluid in the steam generator. The change allowed an immediate drainage of the liquid from the steam generators and produced a gradual uncovery/recovery period representative of breaks of this size. The core uncovered later in time because of additional liquid drainage into the vessel. The additional drainage produced less depth in the uncovery because the minimum level occurred later in time, when the decay heat generation is lower. The cladding temperature for this stable solution was lower than that for the official break analyses, demonstrating that it is conservative to hold water in the generators for small breaks, even with the atypical level behavior displayed by the NOTRUMP code.

The staff was provided with key plots (Reference 4) to understand the mixture level behavior for the 3.25-inch diameter limiting small break LOCA, as well as the modified case with changes to the drift-flux model to allow immediate drainage of the steam generators.

The NRC staff was able to request and view all of the necessary parameters for the limiting break to uncover the physical explanation for the mixture level behavior for the limiting 3.25-inch diameter break. The NRC staff appreciated the cooperation from the licensee staff and the Westinghouse LOCA analysis group in helping to resolve this issue.

## 4.0 CONCLUSION

The NRC staff finds that the proposed small break LOCA reanalysis for CNP-1 was performed in accordance with the methods and analysis practices approved by the NRC for the NOTRUMP code for evaluations of small break LOCA ECCS performance.

The NRC staff noted that there was a large momentary spike/increase in core inlet flow during the 800 to 1000 second time period, which may have been exaggerated by lack of a completely stable numerical solution for a brief time during this period. While the NRC staff believes this evaluation for the 3.25-inch diameter break to be conservative, the staff still remains concerned about the NOTRUMP flow spike for this break. It should be noted that the NRC staff requested Westinghouse to evaluate the flow spike and inform the staff of the results. The outcome of this evaluation should be considered in future licensee LOCA analyses.

The NRC staff further notes that an unofficial run of the 3.25-inch diameter break with a modification to the drift-flux model in the steam generators to preclude early hold-up of the liquid in this region showed that core uncovery was not as severe and also uncovered later in time producing a lower cladding temperature. For this reason, the results of the 3.25-inch diameter break are considered to produce increased core uncovery that represents a more limiting condition, which will tend to maximize the PCT for small breaks.

The limiting break produced a PCT of 1725°F, a peak local cladding oxidation of 3.61%, and a core wide oxidation of less than 1%, all of which are within 10 CFR 50.46 limits. Based on the above considerations, the NRC staff finds the results of the small break LOCA reanalysis to show acceptable ECCS performance at a power level of 3304 MWt, including the 1.34% uncertainty and a total peaking factor at 2.32.

## 5.0 REFERENCES

1. Letter from M. Peifer to U.S. NRC, "Donald C. Cook Nuclear Plant Unit 1, Small Break Loss-of-Coolant Accident Evaluation Model Reanalysis," dated March 29, 2007 (ADAMS Accession No. ML071000431).

2. Letter from M. Peifer to U.S. NRC, "Donald C. Cook Nuclear Plant Unit 1, Docket No. 50-315, Response to Request for Additional Information Regarding the Reanalysis of Unit 1 Small Break Loss-of-Coolant Accident," dated February 29, 2008 (ADAMS Accession No. ML080740053).

3. Letter from J. Jensen to U.S. NRC, "Donald C. Cook Nuclear Plant Unit 1, Docket No. 50-315, Response to Request for Additional Information, Second Round, Regarding Re-analysis of the Small Break Loss-of-Coolant Accident (TAC No. MD5297)," dated December 16, 2008 (ADAMS Accession No. ML083660104).

4. E-mail from K. Steinmetz to T. Beltz, transmitting plots associated with small break LOCA reanalysis audit at Westinghouse (TAC No. MD5297) (ADAMS Accession No. ML090700530).

Principal Contributor: L. Ward, NRR

Date: March 27, 2009

Should you have any questions, please contact 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. 50-315

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