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

March 7, 2013

Mr. Lawrence J. Weber
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, UNITS 1 AND 2 - EVALUATION OF REPORT CONCERNING SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR RELATED TO NUCLEAR FUEL THERMAL CONDUCTIVITY DEGRADATION (TAC NOS. ME8322 AND ME8323)

Dear Mr. Weber:

By letter dated March 19, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12088A104), as supplemented by letter dated June 11, 2012 (ADAMS Accession No. ML12173A025), Indiana Michigan Power Company submitted a report describing a significant error identified in the emergency core cooling system (ECCS) evaluation model, and an estimate of the effect of the error on the predicted peak cladding (PCT) temperature for the Donald C. Cook Nuclear Plant, Units 1 and 2. This report was submitted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.46, paragraph (a)(3), and referred to a letter from Westinghouse Electric Company dated March 7, 2012 (ADAMS Accession No. ML12072A035).

The U.S. Nuclear Regulatory Commission (NRC) staff has evaluated the March 19, 2012, report, along with its supplemental information, and concludes, as set forth in the enclosed safety evaluation, that it satisfies the reporting requirements of 10 CFR 50.46(a)(3). In addition, the NRC staff determined that the report satisfies the intent of the reporting requirements, as discussed in the statement of considerations published in the *Federal Register* (FR) on September 16, 1988; (53 FR 35996), for the realistic ECCS evaluations revision of 10 CFR 50.46. Further, the staff: (1) agrees with the licensee's assessment of the significance of the error; (2) has confirmed that the evaluation model remains adequate; (3) has verified that the licensee continues to meet the PCT acceptance criterion promulgated by 10 CFR 50.46(b); and (4) has determined that the licensee's proposed schedule for reanalysis is acceptable in light of the information provided. The staff's evaluation of the report is enclosed. This letter concludes the staff's efforts under TAC Nos. ME8322 and ME8323.

If you have any questions, please contact me at 301-415-4037 or via e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

Thomas J. Wengert

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure:
As stated

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

CLOSURE EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

REPORT DESCRIBING THE NATURE OF AND ESTIMATED EFFECT ON

PEAK CLADDING TEMPERATURE RESULTING FROM A SIGNIFICANT EMERGENCY

CORE COOLING SYSTEM EVALUATION MODEL ERROR

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated March 19, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12088A104), Indiana Michigan Power Company (I&M, the licensee), submitted a report describing a significant error identified in the emergency core cooling system (ECCS) evaluation model, and an estimate of the effect of the error on the predicted peak cladding temperature (PCT) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. This report was submitted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.46, paragraph (a)(3). The report was supplemented by letter dated June 11, 2012 (ADAMS Accession No. ML12173A025), and referred to a letter from Westinghouse Electric Company dated March 7, 2012 (ADAMS Accession No. ML12072A035).

The U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff has evaluated the report, along with its supplemental information, and has determined that it satisfies the reporting requirements of 10 CFR 50.46(a)(3), and also the intent of the reporting requirements, as discussed in the statement of considerations published in the *Federal Register* (FR) on September 16, 1988 (53 FR 35996), for the realistic ECCS evaluations revision of 10 CFR 50.46. The staff review is discussed in the following sections of this closure evaluation.

2.0 REGULATORY EVALUATION

2.1 Requirements Contained in 10 CFR 50.46

Acceptance criteria for ECCS for light-water nuclear power reactors are promulgated at 10 CFR 50.46. In particular, 10 CFR 50.46(a)(3)(i) requires licensees to estimate the effect of any change to, or error in, an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For the purpose of 10 CFR 50.46, a significant change or error is one which results in a calculated PCT different by more than 50 degrees Fahrenheit (°F) from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F.

Enclosure

For each change to or error discovered in an acceptable evaluation model, or in the application of such a model, 10 CFR 50.46(a)(3)(ii) requires the affected licensee to report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually. If the change or error is significant, the licensee is required to provide this report within 30 days and include with the report a proposed schedule for providing a re-analysis or taking other action as may be needed to show compliance with 10 CFR 50.46 requirements.

2.2 Additional Guidance

Additional clarification concerning the intent of the reporting requirements is discussed in the statement of considerations published in the *Federal Register* (FR) on September 16, 1988 (53 FR 35996), for the best estimate loss-of-coolant accident (LOCA) revision of 10 CFR 50.46:

> [Paragraph (a)(3) of section 50.46] requires that all changes or errors in approved evaluation models be reported at least annually and does not require any further action by the licensee until the error is reported. Thereafter, although reanalysis is not required solely because of such minor error, any subsequent calculated evaluation of ECCS performance requires use of a model with such error, and any prior errors, corrected. The NRC needs to be apprised of even minor errors or changes in order to ensure that they agree with the applicant's or licensee's assessment of the significance of the error or change and to maintain cognizance of modifications made subsequent to NRC review of the evaluation model...
>
> Significant errors require more timely attention since they may be important to the safe operation of the plant and raise questions as to the adequacy of the overall evaluation model... More timely reporting (30 days) is required for significant errors or changes... This final rule revision also allows the NRC to determine the schedule for reanalysis based on the importance to safety relative to other applicant or licensee requirements.

The NRC staff considered this discussion in the FR in its evaluation of the error report submitted by the licensee.

3.0 TECHNICAL EVALUATION

The report submitted by the licensee described the effects of an error in the ECCS evaluation model associated with the degradation of thermal conductivity in nuclear fuel. This issue is discussed in NRC Information Notice (IN) 2009-23, "Nuclear Fuel Thermal Conductivity Degradation" (ADAMS Accession No. ML091550527). Its potential effects in realistic ECCS evaluation models are described in IN 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation [TCD]," (ADAMS Accession No. ML113430785).

Based on the nature of the reported error, and on the magnitude of its effect on the PCT calculation, the NRC staff determined that a detailed technical review is necessary. Based on the regulatory evaluation discussed above, the staff's review was performed to ensure that the NRC agrees with the licensee's assessment of the significance of the error, and to enable the staff to verify that the evaluation model, as a whole, remains adequate. Finally, the staff's review also establishes that the licensee's proposed schedule for re-analysis is acceptable in light of the safety significance of the reported error.

3.1 Overview of Automated Statistical Treatment of Uncertainty Method

The licensee uses the NRC-approved Automated Statistical Treatment of Uncertainty Method (ASTRUM), documented in WCAP-16009-NP-A (ADAMS Accession Nos. ML050910157, ML050910159, and ML050910161, respectively), to evaluate ECCS performance for CNP, Units 1 and 2. ASTRUM relies on an approach based on order statistics in which a set number of cases with randomly varied initial conditions are analyzed using the WCOBRA/TRAC (WC/T) reactor system analysis code. The number of cases is chosen so that the highest predicted PCT within the case set becomes a predictor of the 95/95 upper tolerance limit for the PCT associated with a hypothetical population of LOCA scenarios. This result is used to show compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning PCT.

3.2 Summary of Technical Information in the Report

The licensee's March 19, 2012, report indicated that the effect of the TCD error was an increase in PCT of 384 °F for Unit 1 and 73 °F for Unit 2. The nature of the error, and the method used to estimate its effect on the calculated PCT, is discussed in greater detail in the March 7, 2012, Westinghouse letter and the June 11, 2012, request for additional information (RAI) response. In the report, the licensee also discussed additional changes made to the ECCS evaluation in order to offset the effects of TCD, and to recapture margin to the regulatory limit on PCT.

TCD Error Correction

The error in the ECCS evaluation model was caused by the inability of the fuel rod performance and design (PAD) fuel performance model to account for the effects of TCD with increasing fuel burnup. This error caused fuel temperature initial conditions to be non-conservatively low for higher burnup fuel rods that were analyzed in the ECCS evaluation. In order to correct the error, a burnup-dependent term was added to the nuclear fuel thermal conductivity equation, which caused the predicted initial fuel temperatures to compare better with experimental data obtained from the Halden Reactor Project[1]. The results from the modified PAD (PAD 4.0 + TCD) code were then used to re-initialize the WC/T cases that are performed in execution of ASTRUM.

The licensee used PAD 4.0 + TCD to generate inputs to the ASTRUM execution, which is different from the existing analysis. In the typical ASTRUM analysis, the WC/T initialization generates the fuel conditions using a MATPRO-based[2] analytic procedure. The initial conditions are also calculated using PAD 4.0, and then the WC/T fuel temperature is corrected to the PAD fuel temperature by adjusting fuel rod plenum heat transfer properties. With the

[1] Although comparisons of PAD 4.0 and PAD 4.0 + TCD predictions to Halden Reactor measurements and data are Westinghouse proprietary information, related information and similar comparisons are available from the NRC's FRAPCON computer code in NUREG/CR-7022, "FRAPCON-3.4: Integral Assessment." See in particular Chapter 3 of NUREG/CR-7022.

[2] Hagrman, D.L., G.A. Reymann, and G.E. Mason. 1981. *A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior*, MATPRO Version 11 (Revision 2), NUREG/CR-0479 (TREE-1280), prepared by EG&G Idaho, Inc., Idaho Falls, ID, for the U.S. Nuclear Regulatory Commission, Washington, D.C.

improved PAD correction, the STAV[3] model in WC/T is used for the initialization, instead. The STAV model more closely simulates the fuel performance predicted by PAD 4.0 + TCD than the MATPRO model.

In order to estimate the PCT effect of the TCD error correction, the licensee systematically identified the subset of cases within the ASTRUM analysis that had the potential to produce a limiting result, once corrected for TCD. These cases were re-analyzed within WC/T, using the adapted initial conditions described above.

Additional Model Changes

In order to ensure that facility operation remains compliant with 10 CFR 50.46 requirements and to restore margin, the licensee made additional model changes to offset the increase in predicted PCT due to the TCD model error.

The following design input parameter changes were applied to both Units 1 and 2:

- The licensee included peaking factor burndown and reduced the $F_{\Delta H}$ peaking factor for both units to take advantage of the available Nuclear Design margin. The licensee changed their core operating limits report (COLR) to implement this change.

- Steam generator tube-plugging was reduced from ≤10% to ≤ 2% for Unit 1 and ≤1% for Unit 2.

- Vessel average temperature was reduced from 575.4 °F to 556 °F for Unit 1 and from 578.1 °F to 574 °F for Unit 2. The uncertainty applied to the average temperature remains the same as in the analyses of record (AOR). The new values are consistent with the current operating values and the unit-specific operating procedures.

- Accumulator water temperature range was reduced from 60 °F≤ T_{ACC} ≤ 120 °F to 70 °F≤ T_{ACC} ≤ 100 °F for Unit 1 and to 60 °F≤ T_{ACC} ≤ 115 °F for Unit 2. CNP created a Technical Requirements for Operation (TRO) 8.6.6, Accumulator Temperature, that has been added to each of the unit-specific Technical Requirements Manuals to ensure the plant operates within this new range.

- Refueling water storage tank water temperature range was decreased from 70 °F≤ T_{SI} ≤ 105 °F to 70 °F≤ T_{SI} ≤ 100 °F. This range corresponds with the technical specification surveillance requirements (TS SR) 3.5.4.1 values.

- Safety injection initiation delay time was reduced from ≤ 27 sec (with offsite power) to ≤ 17 sec (with offsite power) and from ≤ 54 sec (without offsite power) to ≤ 28 sec (without offsite power). These reduced values are based on the current SRs at CNP.

[3] Harris, W.R., et al. 2006. *Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1, Volume 1*, WCAP-15836-NP-A, Revision 0, prepared by Westinghouse Electric Company, LLC, Pittsburgh, PA. ADAMS Accession No. ML061220455.

- The minimum pumped ECCS flow rate was reduced based on reduced refueling water storage tank level and containment spilling assumptions. The revised minimum ECCS flow rates are supported by the surveillance testing of the ECCS per TS SRs 3.5.2.1 through 3.5.2.7.

Additionally, for Unit 1, a more realistic but still conservative, containment backpressure input was used.

The net effect of incorporating all of the model changes was a reduction in predicted PCT of 381 °F for Unit 1 and 239 °F for Unit 2.

Reported Results

Following the correction for TCD and the model change, the current predicted PCT for CNP Unit 1 is 2131 °F and the current predicted PCT for CNP, Unit 2 is 1941 °F. By letter dated March 19, 2012, the licensee provided a proposed schedule for re-analysis using an NRC-approved evaluation model that accurately considers TCD, when such a model becomes available. This proposed schedule was submitted pursuant to 10 CFR 50.46(a)(3)(ii). The licensee proposed to submit the reanalysis by December 15, 2016.

3.3 Summary of NRC Staff Evaluation

The NRC staff evaluation of the error report submitted by the licensee included a review of the report itself, a detailed audit to review the analyses supporting the report, and a request for additional information, to which the licensee responded via the June 11, 2012, letter. The staff performed a detailed review of the input parameters and limiting results that were used to generate the estimate, and concluded that the estimate enables the current analysis to maintain a high level of probability that the 2200 °F PCT acceptance criterion is not exceeded.

By letter dated May 14, 2012 (ADAMS Accession No. ML12129A501), the NRC staff issued an RAI with questions for CNP, Units 1 and 2.

The RAI for Question 1 was a simple data request. RAI Questions 2, 3, 4, and 8, pertain to the Westinghouse analytic methods used to estimate the effects of TCD, RAI Questions 5 and 7 relate to methods used by the licensee to ensure that the ECCS evaluation, which includes the estimated effects of TCD, remains bounding of present and planned plant operation, and RAI Question 6 addresses peaking factor adjustments. RAI Question 9 applies to 10 CFR 50.59 processes for addressing changes to the analytic framework used in the licensing basis ECCS evaluation. Since the answers to these questions are important to the NRC staff's evaluation of this error notification, certain questions will be described in detail below.

The NRC staff compared the input data provided in response to Question 1 to the input data for the ASTRUM AOR to confirm the bounding and most limiting cases were included in the estimate analysis. The staff also reviewed the responses to Questions 2 and 8, which describe the analytic approach that the licensee employed to select cases for the estimate analysis. The staff concluded that the analytic approach was rigorous and provides a high level of confidence that the effects of the error have been appropriately estimated.

Question 3 addressed the change to the containment pressure design input. The containment pressure is affected by other design input changes. The licensee calculated a new containment pressure response with the design changes incorporated. The NRC staff has determined this is acceptable because the new containment pressure input is still conservatively low, as required by the ASTRUM methodology.

To estimate the effect of TCD using a more generic model that is more feasibly implemented, the licensee used a representative fuel rod design. In response to Question 4a, the licensee provided information comparing the representative fuel rod type to CNP-specific fuel rod design characteristics (Enclosure 2 to the June 11, 2012, letter). The information demonstrated that the representative rod was mostly the same as the CNP-specific design. Where deviations were identified, the licensee indicated that the representative rod was bounding of the CNP design, or offset by other margins in the CNP design. For the design characteristics, for which the representative rod was non-bounding, the licensee provided additional explanation showing that other, more bounding characteristics offset the effects of the unbounded characteristics. Based on its review of the information provided by the licensee, the NRC staff determined that the response to Question 4a was acceptable, because it demonstrated the acceptability of using a representative fuel rod design to estimate the effects of TCD.

In response to Question 4b, the licensee provided values for coefficients used in the PAD 4.0 + TCD uranium dioxide (UO_2) thermal conductivity equation (Enclosure 2 to June 11, 2012, letter). The information provided by the licensee allowed the NRC staff to verify that the thermal conductivity equation had been adjusted in a way that reflected the available experimental data. Figures included with the RAI response also showed reasonable agreement between the TCD-corrected versions of PAD and HOTSPOT in comparison to FRAPCON calculations. Based on these considerations, the NRC staff finds the response to Question 4b acceptable.

In response to Questions 4c and 4d, the licensee stated that the WCOBRA/TRAC and HOTSPOT code versions used in the evaluation of fuel pellet TCD do not include any error corrections, code improvements, or model changes from the AOR code versions. The NRC staff finds that the licensee's responses to Questions 4c and 4d are acceptable.

In response to Question 4e, the licensee provided additional detail about the validity of the corrected UO_2 thermal conductivity model in each code since the models are different and may be implemented differently within each code. In its response, the licensee explained that the technical basis for each model was the same, and clarified that the differences between the models were minor. The licensee's response included graphs that compared the predicted UO_2 thermal conductivity between FRAPCON 3.4, and the corrected versions of HOTSPOT and PAD. The NRC staff reviewed the information and concluded that the UO_2 thermal conductivity models produced similar results and were reasonably valid as compared to FRAPCON 3.4. Based on this information, the NRC staff determined that the RAI response was acceptable.

The NRC staff issued Questions 5 and 7 to determine how the licensee would ensure that any changed design limits or other input values would remain applicable to the operating cycles at CNP. In response to these RAI questions, the licensee provided a detailed explanation of fuel reload design, administrative control, and operating surveillance processes. The reload process ensures that specific cycle designs conform to the limits established by the large-break loss-of-coolant accident (LBLOCA) analyses, which now account for TCD effects. The administrative control and surveillance processes ensure that the core operates as designed throughout the cycle. In combination, the reload, the administrative control, and surveillance processes ensure that the core remains within revised limits associated with the TCD effect estimate. Because the licensee provided information indicating that the cycles will be designed, administratively controlled, and surveillanced in a manner that ensures adherence to revised limits associated with the effects of TCD, the NRC staff determined that the responses to Questions 5 and 7 were acceptable.

The licensee provided an explanation of the changes to the CNP peaking factors in response to Question 6. In its response, the licensee explained that a reduction in $F\Delta H$ peaking factor will utilize available nuclear design margin and the COLR will be updated appropriately to ensure the plant is operated within these new constraints. The NRC staff determined that the licensee's response was acceptable because it shows that the cycle-specific core designs meet the analyzed limits.

Since the licensee's evaluation is based on a rigorous analytic approach, the NRC staff finds that the licensee has demonstrated that the effects of the error are appropriately estimated, and that the licensee has provided assurance that it will not exceed 2200 °F following a LOCA. The staff may find that it is necessary to revisit this conclusion if other significant errors in the ASTRUM evaluation model are reported. By letter dated March 19, 2012, the licensee provided a proposed schedule for reanalysis, stated as follows:

> By December 15, 2016, I&M will submit to the NRC for review and approval a LBLOCA analysis that applies NRC-approved methods that include the effects of fuel TCD. The date for the analysis submittal is projected on the following milestones needed to perform a revised licensing basis LBLOCA analysis with an NRC-approved ECCS Evaluation Model that explicitly accounts for TCD:
>
> 1) Submittal by Westinghouse, to the NRC for review and approval, of a revised fuel performance and LBLOCA Evaluation Model methodologies that include the effects of TCD.
>
> 2) Prior NRC approval of a fuel performance analysis methodology that includes the effects of TCD.
>
> 3) Prior NRC approval of a LBLOCA Evaluation Model that includes the effects of TCD and accommodates the ongoing 10 CFR 50.46(c) rulemaking process.

The re-analysis requirement contained in 10 CFR 50.46(a)(3)(ii) states the following:

> ... and [the licensee] shall include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with [10 CFR] 50.46 requirements

The proposed schedule provided by the licensee satisfies this requirement by indicating a proposed date that a reanalysis will be provided.

Further, the NRC staff finds that the proposed re-analysis date is commensurate with the safety significance of the issue, based on the considerations described above. Therefore, the NRC staff finds that the licensee has adequately addressed the re-analysis requirement contained in 10 CFR 50.46(a)(3)(ii).

In its March 19, 2012, submittal, the licensee's proposed schedule for re-analysis was classified as a regulatory commitment. The NRC staff concludes that tracking this activity under the licensee's commitment management program provides the proper tracking and control mechanism.

4.0 CONCLUSION

Based on the considerations discussed above, the NRC staff finds that the report submitted pursuant to 10 CFR 50.46(a)(3)(ii), concerning an ECCS evaluation model error pertaining to TCD, satisfies the reporting requirements of 10 CFR 50.46(a)(3). Further, the submittals dated March 19, 2012, and June 11, 2012, enabled the staff to: (1) determine that it agrees with the licensee's assessment of the significance of the error; (2) confirm that the evaluation model remains adequate; (3) verify that the licensee continues to meet the PCT acceptance criterion promulgated by 10 CFR 50.46(b); and (4) determine that the licensee's proposed schedule for reanalysis is acceptable in light of the information provided.

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Date of issuance: March 7, 2013

If you have any questions, please contact me at 301-415-4037 or via e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

/RA/

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Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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ADAMS Accession Nos.: Proprietary LTR ML12331A333 Non-Proprietary LTR ML13077A137
***via memo dated October 16, 2012**

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