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

June 21, 2013

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 - CLOSURE EVALUATION FOR REPORT PURSUANT TO TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS*, PART 50, SECTION 50.46, PARAGRAPH (a)(3)(ii) CONCERNING SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR RELATED TO NUCLEAR FUEL THERMAL CONDUCTIVITY DEGRADATION (TAC NOS. ME8918 AND ME8919)

Dear Sir or Madam:

By letter dated June 14, 2012, Entergy Nuclear Operations, Inc., the licensee for Indian Point Nuclear Generating Unit Nos. 2 and 3, 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. The report was submitted pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46 (10 CFR 50.46), paragraph (a)(3)(ii).

The Nuclear Regulatory Commission staff evaluated the report and has determined that the report satisfies the intent of the reporting requirements promulgated at 10 CFR 50.46(a)(3)(ii). The staff evaluation is enclosed.

Please contact me at 301-415-1364 if you have any questions.

Sincerely,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosure:
Closure Evaluation

cc w/encl: Distribution via Listserv

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CLOSURE EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ENTERGY NUCLEAR OPERATIONS

INDIAN POINT ENERGY CENTER UNITS 2 AND 3

REPORT DESCRIBING THE NATURE OF

AND ESTIMATED EFFECT ON PEAK CLADDING TEMPERATURE OF A

SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR

DOCKET NOS 50-247 AND 50-286

1.0 INTRODUCTION

By letter dated June 14, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12184A038), Entergy Nuclear Operations, Inc., 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 Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and IP3). This report was submitted pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46 (10 CFR 50.46), paragraph (a)(3)(ii). The report referred to a letter from Westinghouse Electric Company dated March 7, 2012 (ADAMS Accession No. ML12072A035).

The U.S. Nuclear Regulatory Commission (NRC, or Commission) staff has evaluated the report, along with its supplemental information, and determined that it satisfies the reporting requirements of 10 CFR 50.46(a)(3)(ii), and also the intent of the reporting requirements, as discussed in the statement of considerations published on September 16, 1988, in the *Federal Register* (FR), for the realistic ECCS evaluations revision of 10 CFR 50.46 (53 FR 35996). 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 ECCSs 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 peak fuel cladding temperature different by more than 50 degrees Fahrenheit (°F) from the temperature calculated for the limiting transient using the last acceptable model, or is an accumulation 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 reanalysis 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 for the realistic ECCS evaluation revision of 10 CFR 50.46 (53 FR 35996):

> [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 *Federal Register* 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." 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."

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 it 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 Emergency Core Cooling System Evaluation Models

The licensee evaluates ECCS performance for large break loss-of-coolant accident (LOCA) scenarios differently at each unit. IP2 uses the Automated Statistical Treatment of Uncertainty Method (ASTRUM), and IP3 uses the Code Qualification Document (CQD) method.

For IP2, the licensee uses the NRC-approved ASTRUM evaluation model, documented in WCAP-16009-NP-A (ADAMS Accession Nos. ML050910157, ML050910159, and ML050910161), to evaluate ECCS performance. 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.

For IP3, the licensee evaluates ECCS performance using the NRC-approved CQD method, which is described in WCAP-12945-P-A.[1] The CQD method relies on a statistical approach using a response surface technique, in which a reference transient is analyzed using the WCOBRA/TRAC (WC/T) computer code, and a statistically significant number of perturbations are analyzed to determine how uncertainties affect the predicted PCT. Convoluted uncertainty responses for various categories of parameters, including power distribution, plant initial conditions, and thermal hydraulic parameters, are then added to the reference transient PCT. The 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 report indicated that the effect of the thermal conductivity degradation (TCD) error was 209 °F for IP2, and 185 °F for IP3. 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. 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 both ECCS evaluation models 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 evaluations. In order to correct for 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

[1] The non-proprietary version of WCAP-12945-P-A is WCAP-14747 (ADAMS Accession No. 9804070248).

with experimental data obtained from the Halden Reactor Project.[2] The results from the modified PAD (PAD 4.0 + TCD) code were then used in each evaluation model application to estimate the magnitude of the TCD error on the predicted PCT in each evaluation model application.

The approach used to perform the estimation varied between IP2 and IP3. For IP2, the licensee systematically identified the subset of ASTRUM cases likely to be most significantly affected by TCD. These cases were re-executed in WC/T using the corrected fuel parameters, and the difference in predicted PCT was determined. For IP3, the effect was estimated by correcting the fuel parameters in the reference transient analysis and re-executing the reference transient. This difference in approaches was required because of the differences between the two evaluation models.

Additional Model Changes to Offset TCD Effects

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. These included reductions in the steady-state and transient power peaking factors, a reduction in the hot assembly average power, and at IP2, a reduction in upper bound steam generator tube plugging. The net effect of incorporating all of the model changes was a reduction in predicted PCT of 63 °F for IP2 and 95 °F for IP3.

Reported Results

Following the correction for TCD and the model change, the current predicted PCT for IP2 is 2113 °F, and for IP3, it is 2058 °F. The licensee also stated that it would perform a reanalysis using an NRC-approved evaluation model that accurately considers TCD, when such a model becomes available. This submittal is currently expected in 2016.

3.3 Summary of Staff Evaluation

In its evaluation, the NRC staff reviewed (1) the approach used to estimate the effects of TCD, (2) the estimated effect of TCD at both units, and (3) the licensee's proposal for re-analysis in consideration of the approach used to estimate the effects of TCD. As discussed in the following paragraphs, the staff determined that the licensee's estimate and proposal for reanalysis are acceptable.

To estimate the effects of TCD, the licensee used a modified uranium thermal conductivity model to account for TCD, and re-executed the most appropriate[3] WC/T cases using inputs from the revised thermal conductivity model. The explicit model is described in the March 7, 2012, Westinghouse letter to the Commission. A proprietary enclosure to the Westinghouse letter also provides information to show that the modified uranium thermal conductivity model

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[2] The Halden Reactor Project is an international, collaborative research project intended to gather data and information pertaining to reactor technology. Although the specific 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.

[3] For IP2, "most appropriate" means the limiting cases in the ASTRUM run set. For IP3, "most appropriate" means the reference transient.

more accurately reflects available high-burnup data, as described in Section 3.1 in this evaluation. The NRC staff also notes, for IP2, that the licensee has examined the ASTRUM run set to identify the WC/T cases with potential to be most significantly affected by TCD, and explicitly re-analyzed these cases with initial conditions that accurately capture the TCD phenomenon. Since the IP3 estimation relies on re-execution of the reference transient, this identification was not necessary for Unit 3.

The NRC staff has reviewed estimating techniques for the same phenomena in the generically approved ASTRUM evaluation model for several other licensing actions. In a recent request for extended power uprate, the requesting licensee addressed a staff request for additional information by identifying a limiting subset of cases to re-execute, and then by completely re-executing the entire ASTRUM run set. In this investigation, the original, limited set of cases contained the new limiting PCT. Also, several reports submitted pursuant to 50.46 have provided TCD effect estimates using methods similar to those used for IP2 and IP3. In the case of the uprate, the staff concluded that the licensee had acceptably accounted for the effects of TCD in its ECCS evaluation; in the case of the 50.46 reports, the staff determined that the estimates provided in the reports satisfied the applicable reporting requirements.

Based on the following considerations: (1) The PAD 4.0 + TCD and revised HOTSPOT fuel performance models generate fuel stored energy initial conditions that result in reasonable agreement with available high burnup data, and (2) the licensee has identified the appropriate WC/T cases and re-executed them using the revised fuel performance models, the NRC staff concludes that the licensee's estimate of the effects of TCD is acceptable. The staff also notes, as discussed above, that this approach has been applied previously at other licensed facilities and accepted by the staff.

The estimated effect of TCD at IP2 is 209 °F. Recently received explicit estimates of the effects of TCD using the ASTRUM evaluation model have ranged from 73 °F to 384 °F; the estimate for IP2 falls within that range. The updated PCT for IP2 is 2113 °F, which falls within the regulatory acceptance criterion of 2200 °F. Because the effect of TCD is consistent with other, similar estimates, and because the updated PCTs meet the 10 CFR 50.46(b)(1) acceptance criteria, the staff did not identify any significant issues with the estimates for IP2.

The estimated effect of TCD at IP3 is 185 °F. Recently received estimates of the effects of TCD using the CQD evaluation model have ranged from 15 °F to 238 °F; the estimate for IP3 falls within that range. The updated PCT for IP3 is 2058 °F, which falls within the regulatory acceptance criterion of 2200 °F. Because the effect of TCD is consistent with other, similar estimates, and because the updated PCTs meet the 10 CFR 50.46(b)(1) acceptance criteria, the staff did not identify any significant issues with the estimates for IP3.

Attachment 3 to the licensee's letter dated June 14, 2012, indicated that the licensee would "submit to the NRC LBLOCA [large-break, loss-of-coolant-accident] analyses that apply NRC approved methods that include the effects of fuel pellet thermal conductivity degradation (TCD) for Indian Point Unit Nos. 2 and 3." The licensee stated that this action would be completed on or before December 15, 2016. Because the licensee included a proposed date that it would provide a re-analysis to the Commission, the NRC staff determined that the licensee satisfied the re-analysis requirement contained in 10 CFR 50.46(a)(3)(ii).

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 intent of the 10 CFR 50.46 reporting requirements. The submittal dated June 14, 2012, enabled the staff to (1) determine that it agrees with the licensee's assessment of the significance of the errors, (2) confirm that the evaluation models remain 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 re-analysis is acceptable in light of the information provided.

Principal Contributor: B. Parks, NRR

Date: June 21, 2013

June 21, 2013

Vice President, Operations
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Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
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Docket Nos. 50-247 and 50-286

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