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

September 4, 2013

Mr. Joseph E. Pacher
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – CLOSURE EVALUATION FOR REPORT PURSUANT TO 10 CFR 50.46(a)(3) CONCERNING SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR RELATED TO PREDICTED PEAK CLADDING TEMPERATURE (TAC NO. MF0582)

Dear Mr. Pacher:

Pursuant to 10 CFR 50.46(a)(3), R. E. Ginna Nuclear Power Plant, LLC, the licensee for R. E. Ginna Nuclear Power Plant, submitted a report, dated August 16, 2012, to U.S. Nuclear Regulatory Commission (NRC), describing a significant error identified in the emergency core cooling system (ECCS) evaluation model. In that report, the licensee provided an estimate of the effect of the error on the predicted peak cladding temperature.

The NRC staff evaluated the report, and has determined that the report satisfies the reporting requirements of 10 CFR 50.46(a)(3), and also the intent of the reporting requirements, as discussed in the rule's statement of considerations for the realistic ECCS evaluations revision of 10 CFR 50.46 (53 FR 35996) as published in the *Federal Register* dated September 16, 1988. The NRC staff evaluation is enclosed. This concludes the NRC staff's evaluation efforts under TAC No. MF0582.

Please contact me at (301) 415-1476 if you have any questions on this issue.

Sincerely,

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: Safety Evaluation

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CLOSURE EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

R.E. GINNA NUCLEAR POWER PLANT, LLC

R. E. GINNA NUCLEAR POWER PLANT

REPORT DESCRIBING THE NATURE AND ESTIMATED EFFECT OF AN ERROR IN

CALCULATING THE PEAK CLADDING TEMPERATURE

IN THE EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL

1.0 INTRODUCTION

By letter dated August 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12233A621), R.E. Ginna Nuclear Power Plant, LLC, 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 R.E. Ginna Nuclear Power Plant (Ginna). This report was submitted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 46 (10 CFR 50.46), paragraph (a)(3). The report referenced a letter from Westinghouse Electric Company, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation," dated March 7, 2012 (ADAMS Accession No. ML12072A035), and was supplemented by an additional letter dated June 19, 2013 (ADAMS Accession No. ML13175A357).

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), and also the intent of the reporting requirements, as discussed in the rule's 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 NRC staff's evaluation is described below.

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 peak fuel cladding temperature different by more than 50 degrees Fahrenheit (°F) from the temperature calculated for the limiting transient

Enclosure

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.

For each change to or error discovered in an acceptable evaluation model or in the application of such a model, paragraph (a)(3)(ii) to 10 CFR 50.46 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 published on September 16, 1988, in the FR for the best estimate loss-of-coolant-accident (LOCA) 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... the 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 the discussion in the *Federal Register* in its evaluation of the licensee's error report.

3.0 TECHNICAL EVALUATION

The report submitted by the licensee describes 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," and its potential effect on realistic emergency core cooling system evaluation models as described in IN 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation [TCD]."

Based on the nature of the reported error, and on the magnitude of the effects on the PCT calculation, the NRC staff concluded that a detailed technical review is necessary. The NRC staff's review was performed to ensure that the NRC staff 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 NRC staff's review also establishes that the licensee's proposed schedule for reanalysis is acceptable, because of the acceptable safety significance of the reported error.

Automated Statistical Treatment of Uncertainty Method (ASTRUM)

The licensee uses the NRC-approved ASTRUM, documented in WCAP-16009-NP-A (ADAMS Accession Nos. ML050910157, ML050910159, and ML050710161), 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. The result is used to show compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning PCT.

3.1 Summary Of Technical Information In The Report

The licensee's report indicated that the effect of the TCD error was 230 °F. The nature of the error, and the method used to estimate its effect on the calculated peak fuel cladding temperature, is discussed in greater detail in the March 7, 2012, Westinghouse letter.

TCD Error Correction

The error in the ECCS evaluation model was caused by the inability of the Westinghouse Improved Fuel Rod Performance and Design (PAD 4.0) 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 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 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.

Note that the TCD correction also includes a peaking factor burndown effect, which captures a reduction in the core peaking factors that naturally occurs throughout fuel life. This phenomenon partially offsets the net effect of TCD by lowering the initial stored energy in the fuel.

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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.

Estimation of the Effect of TCD in the PCT Calculation

The licensee stated that the effect of accounting for TCD, as described above, in the ASTRUM ECCS evaluation is described in the non-proprietary enclosure to the March 7, 2012, letter from Westinghouse Electric Company.

Additional Changes Reported

The licensee also reported an additional model change to compensate for the effects of TCD, while keeping the reported PCT within regulatory acceptance criteria. The licensee estimated that a reduction in steady-state heat flux hot channel factor (FQ) would reduce the predicted PCT by 96 °F.

Reported Results

Following the correction for TCD and the model change, the current predicted PCT for R. E. Ginna is 2041 °F.

3.2 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, 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 NRC 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 sensitive 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 more accurately reflects available high-burnup data, as described in Section 3.1 in this evaluation.

The NRC staff has reviewed this estimating technique 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 approximately 50 WC/T cases to re-execute, and then by completely re-executing the entire ASTRUM run set. In this investigation, the original, limited set of 50 cases contained the new limiting PCT. Also, several reports submitted pursuant to 10 CFR 50.46 have provided TCD effect estimates using a similar method. In the case of the uprate, the NRC staff concluded that the licensee had acceptably accounted for the effects of TCD in its ECCS evaluation; in the case of the 10 CFR 50.46 reports. The NRC staff concluded that the estimates provided in the reports satisfied the applicable reporting requirements.

Based on the following considerations: (1) The PAD 4.0 + TCD and related, revised elements of the ECCS evaluation model generate fuel stored energy initial conditions that result in reasonable agreement with available high burnup data, and (2) the licensee has identified the limiting 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 NRC 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 R. E. Ginna is 230 °F. Recently received explicit estimates of the effects of TCD using the ASTRUM evaluation model have ranged from 73 °F to 384 °F; this estimate falls within that range. The updated PCT is 2041 °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 NRC staff did not identify any significant issues with the estimates.

In its cover letter, the licensee stated the following:

> In accordance with 10 CFR 50.46, Ginna will conduct a re-analysis following approval by the NRC of a revised [Large Break Loss-of-Coolant-Accident] LBLOCA evaluation model, with explicit treatment of thermal conductivity degradation (TCD), if the model used for the impact of TCD in this 30-day report is determined to be non-conservative with respect to the new approved model.

By contrast, 10 CFR 50.46(a)(3)(ii) states, in part, that 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." As described in the Regulatory Evaluation, the statements of consideration explain further that "the 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."

After reviewing the May 30, 2012, submittal, the NRC staff was unable to determine that the above reanalysis statement provided by the licensee satisfies the requirements in 10 CFR 50.46. Due to this determination, the NRC staff issued a Request for Additional Information requesting that the licensee submit either a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with 10 CFR 50.46 requirements. In the RAI Response submitted on June 19, 2013, the licensee states:

> R.E. Ginna Nuclear Power Plant, LLC will perform a [Large Break Loss-of-Coolant-Accident] LBLOCA re-analysis that applies the NRC approved methodology, which includes the effects of fuel thermal conductivity degradation (TCD), within 24 months of the completion of the following three milestones...

The three milestones were (1) submittal by Westinghouse of a fuel performance methodology that accounts realistically for the effects of TCD, (2) NRC approval of the new Westinghouse method, and (3) NRC approval of a revised LBLOCA evaluation model. Since 10 CFR 50.46(a) requires the use of an acceptable evaluation model, the proposed schedule is acceptable in that it allows suitable implementation time following NRC approval of an evaluation model that accounts for the effects of TCD.

The NRC staff concludes, that the licensee's proposed schedule for reanalysis is acceptable and that the reanalysis requirement of 10 CFR 50.46 is presently satisfied.

The NRC staff reviewed the licensee's report estimating the effect of TCD on the large break LOCA analyses for Ginna. Based on the technical rigor employed by the licensee, which included correcting the TCD error using a model that agrees with available experimental data and explicitly re-evaluating a limiting subset of WC/T cases, the NRC staff concluded that the TCD estimate was acceptable. Also, the NRC staff reviewed the licensee's proposed schedule for reanalysis and determined that the licensee satisfied the reanalysis requirement set forth in 10 CFR 50.46(a)(3)(ii).

4.0 CONCLUSION

Based on the considerations discussed above, the NRC staff concludes that the report submitted pursuant to 10 CFR 50.46(a)(3), concerning an ECCS evaluation model error pertaining to TCD, satisfies the intent of the 10 CFR 50.46 reporting requirements. The report and supplemental information enabled the NRC staff to (1) conclude that it agrees with the licensee's assessment of the significance of the error, (2) confirm that the evaluation model remains adequate, and (3) confirms that the licensee continues to meet the PCT acceptance criterion promulgated by 10 CFR 50.46(b). The NRC staff concludes that the licensee's proposed schedule for reanalysis is acceptable and, therefore, the reanalysis requirement of 10 CFR 50.46 is presently satisfied.

September 4, 2013

Mr. Joseph E. Pacher
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – CLOSURE EVALUATION FOR REPORT PURSUANT TO 10 CFR 50.46(a)(3) CONCERNING SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR RELATED TO PREDICTED PEAK CLADDING TEMPERATURE (TAC NO. MF0582)

Dear Mr. Pacher:

Pursuant to 10 CFR 50.46(a)(3), R. E. Ginna Nuclear Power Plant, LLC, the licensee for R. E. Ginna Nuclear Power Plant, submitted a report, dated August 16, 2012, to U.S. Nuclear Regulatory Commission (NRC), describing a significant error identified in the emergency core cooling system (ECCS) evaluation model. In that report, the licensee provided an estimate of the effect of the error on the predicted peak cladding temperature.

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Please contact me at (301) 415-1476 if you have any questions on this issue.

Sincerely,
/ra/
Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: Safety Evaluation

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