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

November 16, 2012

Mr. Kelvin Henderson
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

Mr. Steven D. Capps
Vice President
McGuire Nuclear Station
Duke Energy Carolinas, LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (CATAWBA 1 AND 2), AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 (MCGUIRE 1 AND 2), 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. ME8447, ME8448, ME8449, AND ME8450)

Dear Messrs. Henderson and Capps:

Pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46 (10 CFR 50.46), paragraph (a)(3)(ii), Duke Energy Carolinas, LLC (Duke Energy, the licensee), the licensee for Catawba 1 and 2 and McGuire 1 and 2, 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 by letter dated March 16, 2012.

The U.S. Nuclear Regulatory Commission (NRC) 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), 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 NRC staff evaluation is enclosed.

If you have any questions, please call me at 301-415-1119.

Sincerely,

Jon Thompson

Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369, and 50-370

Enclosure:
As stated

cc w/encl: Distribution via Listserv

CLOSURE EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR REPORT DESCRIBING THE NATURE OF AND ESTIMATED EFFECT

ON PEAK CLADDING TEMPERATURE OF A SIGNIFICANT

EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2,

DOCKET NOS. 50-413 AND 50-414

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated March 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12079A180), Duke Energy Carolinas, LLC (Duke Energy, 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 Catawba Nuclear Station, Units 1 and 2 (Catawba 1 and 2) and McGuire Nuclear Station, Units 1 and 2 (McGuire 1 and 2). This report was submitted pursuant to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.46 Paragraph (a)(3)(ii) (10 CFR 50.46(a)(ii)). The report was supplemented by a letter dated May 29, 2012 (ADAMS Accession No. ML121510463), and referred to a letter from Westinghouse Electric Company (Westinghouse) 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). It also meets the intent of the reporting requirements as discussed in the statement of considerations (SOC) for the realistic ECCS evaluations revision of 10 CFR 50.46 as published in the in the *Federal Register* (FR) on September 16, 1988 (53 FR 35996). The NRC 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 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 degrees Fahrenheit.

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 SOC for the realistic ECCS evaluations 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 SOC (53 FR 35996) 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" (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 NRC 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 NRC 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 in light of the safety significance of the reported error.

3.1 Overview of Code Qualification Document Method

The licensee evaluates ECCS performance using the NRC-approved Code Qualification Document method (CQD), 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 15 degrees Fahrenheit. The nature of the error, and the method used to estimate its effect on the calculated PCT, is discussed in greater detail in the Westinghouse letter dated March 7, 2012. 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 for the error, a burnup-dependent term was added to the nuclear fuel thermal

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

conductivity equation, which caused the predicted initial fuel temperatures to compare better with experimental data obtained from the Halden Reactor Project.[2] The results from the modified PAD (PAD 4.0 + TCD) code were then used to re-analyze the reference transient from the CQD analysis.

The licensee used PAD 4.0 + TCD to generate inputs to the CQD reference transient analysis, which is different from the existing analysis. In the typical CQD analysis, the WC/T initialization generates the fuel conditions using a MATPRO-based[3] 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 improved PAD correction, the STAV[4] 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.

The estimated effect of the TCD correction for Catawba 1 and 2 is a 15 degrees Fahrenheit increase in the predicted PCT.

Additional Model Changes

Although the licensee's estimate of the effect of TCD was calculated using a modified version of PAD 4.0, the licensee currently calculates initial fuel stored energy for its ECCS evaluation using PAD 3.4. The enclosure to the Westinghouse letter dated March 7, 2012, indicated that the CQD evaluation model was originally licensed using PAD 3.4. Upon approval of PAD 4.0, its usage was extended to the CQD method on a forward-fit basis. The ECCS evaluation for Catawba 1 and 2 was performed prior to the approval of PAD 4.0.

Since the licensee used a corrected version of PAD 4.0 to estimate the effect of TCD, the licensee also provided an estimate of the effect of using PAD 4.0 instead of PAD 3.4 to determine the fuel initial stored energy. Notwithstanding the effects of TCD, PAD 4.0 provides a more realistic, and lower, prediction of fuel initial stored energy when compared to PAD 3.4, as analyzed. The estimated effect of a change from PAD 3.4 to PAD 4.0 is a reduction of 75 degrees Fahrenheit in the predicted PCT.

Reported Results

Following the correction for TCD and the model change, the current predicted PCT for Catawba 1 and 2 and McGuire 1 and 2 is 2085 degrees Fahrenheit. The licensee also stated

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

4 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).

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.

The report also included the estimated effect of a measurement uncertainty recapture uprate for McGuire 1 and 2, which is currently under NRC staff review. If approved and implemented, the effect of this proposed uprate would cause the predicted PCT for McGuire 1 and 2 to increase by an additional 16 degrees Fahrenheit, to 2101 degrees Fahrenheit. Because the change associated with this estimated increase is currently under a separate review, it is not addressed by this closure evaluation.

3.3 Summary of the 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 by letter dated May 29, 2012. The NRC 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 ensure that there is a high level of probability that the 2200 degrees Fahrenheit PCT acceptance criterion is not exceeded.

The NRC staff issued requests for additional information (RAIs) by letters dated April 23, 2012, (ADAMS Accession No. ML12110A194), and April 27, 2012 (ADAMS Accession No. ML12116A147), pertaining to several topics related to the TCD estimate for Catawba 1 and 2 and McGuire 1 and 2. These topics included the PAD 4.0 + TCD adjustment, the licensee's control of plant operating conditions relative to changed inputs in the ECCS evaluation, and overall compliance with 10 CFR 50.59 requirements in light of possible changes to the method of evaluation used to evaluate ECCS performance.

The NRC staff issued RAI question 1 to obtain additional detail concerning the development of PAD 4.0 + TCD and verify that its results were consistent with experimental data available for fuel at higher burnup values.

In response to RAI question 1.a., the licensee clarified that the final paragraph on Page 2 of 9 of the enclosure to the Westinghouse letter does not apply to the Catawba 1 and 2 and McGuire 1 and 2 TCD estimates, because fuel temperature and rod internal pressure analyses were performed using Duke-specific fuel rod dimensions (Enclosure 1 to May 29, 2012, letter). The NRC accepts this response, because it clarifies that the Duke TCD estimate is specifically applicable to the Catawba 1 and 2 and McGuire 1 and 2 Units.

In response to RAI 1.b., the licensee provided values for coefficients used in the PAD 4.0 + TCD uranium dioxide (UO_2) thermal conductivity equation (Enclosure 4 to May 29, 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 RAI 1.b. acceptable.

In response to RAI 1.c., the licensee provided reports documenting error corrections, code improvements, and miscellaneous code cleanup affecting WC/T and HOTSPOT between the time that the analyses of record were performed and the TCD estimates were generated. The NRC staff requested this information because the enclosure to the March 7, 2012, Westinghouse letter referred to these changes in a general sense, and the NRC staff determined that additional detail was necessary to verify that the code changes would not affect the TCD estimate in unintended ways. The NRC staff reviewed the provided documentation and did not identify any code changes referenced in the Westinghouse letter that would affect the validity of the TCD estimate. Based on this consideration, the NRC staff finds the licensee's response to NRC RAI 1.c. acceptable.

In response to RAI 1.d., the licensee stated that the code changes discussed in the previous paragraph did not affect the fuel thermal conductivity model, but that using a more recent HOTSPOT code version is appropriate because certain error corrections affect the TCD effect estimate. The licensee provided a fuel relocation error correction as an example. Correction of a fuel relocation error would cause the TCD estimate to be more realistic in that the energy stored in the fuel is affected by the amount of fuel present at a given location. Therefore, the NRC staff finds that the licensee's response to RAI 1.d. is acceptable.

In response to RAI 1.e., 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. The licensee's response 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.

In response to RAI 1.f., the licensee provided information describing the procedure and incremental adjustments that are performed within WC/T and HOTSPOT to obtain steady-state convergence. While most of the information described minor adjustments to boundary conditions to obtain convergence in the system state properties, a portion of the response described adjustments made to ensure that the fuel stored energy predicted in HOTSPOT agrees with the result predicted in WC/T. The information clarified that any adjustments made in the HOTSPOT calculations were typically minor, and that, because of the similarities in the HOTSPOT and WC/T fuel performance models, the adjustments usually resulted in similar state properties used by both codes. The NRC staff determined that this response was acceptable because it indicated that both HOTSPOT and WC/T were relying on similar properties when initiating the Large-Break-Loss-Of-Coolant accident (LBLOCA) transient calculations.

The NRC staff issued RAI 2 to determine how the licensee would ensure that any changed design limits or other input values would remain applicable to the operating cycles at Catawba 1 and 2 and McGuire 1 and 2. In response to the RAI, the licensee provided a detailed explanation of cycle design and operating surveillance processes. The cycle design processes ensure that specific cycle designs conform to the limits established by the LBLOCA analyses, which now account for TCD effects. The surveillance processes ensure that the core operates as designed throughout the cycle. In combination, the design 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 and surveilled in a manner that ensures adherence to revised limits associated with the effects of TCD, the NRC staff determined that the response to RAI 2 was acceptable.

The licensee provided a commitment to perform a reanalysis, stated as follows:

> Before December 15, 2016, Duke Energy will submit to the NRC for review and approval a LBLOCA analysis that applies an NRC-approved ECCS Evaluation Model that includes the effects of fuel thermal conductivity degradation.

Since the licensee's evaluation is based on a very 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 degrees Fahrenheit following a LOCA. The NRC staff may find that it is necessary to revisit this conclusion if other significant errors in the CQD evaluation model are reported.

The reanalysis 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 mat be needed to show compliance with [10 CFR] 50.46 requirements.

The commitment 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 reanalysis 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 reanalysis 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 submittals dated March 7, 2012, March 16, 2012, and May 29, 2012, enabled the NRC staff to (1) determine that the NRC staff 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.

If you have any questions, please call me at 301-415-1119.

Sincerely,

/RA/

Jon Thompson, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369, and 50-370

Enclosure:
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