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

May 17, 2013

Mr. David A. Heacock
President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2, CLOSURE EVALUATION FOR REPORT PURSUANT TO 10 CFR 50.46(a)(3) CONCERNING SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR RELATED TO NUCLEAR FUEL THERMAL CONDUCTIVITY DEGRADATION (TAC NOS. ME8727 AND ME8728)

Dear Mr. Heacock:

The Virginia Electric and Power Company (Dominion), the licensee for North Anna Power Station, Unit Nos. 1 and 2 (NAPS), submitted a report pursuant to 10 CFR 50.46(a)(3), describing the estimated effect of thermal conductivity degradation on the predicted peak cladding temperature associated with the emergency core cooling system (ECCS) evaluation model for large-break loss-of-coolant accidents in their letters dated May 16, 2012, as supplemented by letter dated December 19, 2012.

The U.S. Nuclear Regulatory Commission (NRC) has completed the review. Upon evaluating the report, the NRC staff has determined that the NAPS 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 on September 16,1988, in the *Federal Register* (FR), for the realistic ECCS evaluations revision of 10 CFR 50.46 (53 FR 35996).

A copy of the Closure Evaluation is enclosed.

Sincerely,

V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Closure Evaluation

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CLOSURE EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

REPORT DESCRIBING THE NATURE OF

AND ESTIMATED EFFECT ON PEAK CLADDING TEMPERATURE

OF A SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR

1.0 INTRODUCTION

By letter dated May 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12143A149), Virginia Electric and Power Company (Dominion), 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 North Anna Power Station (NAPS), Unit Nos. 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 December 10, 2012 (ADAMS Accession No. ML12356A300).

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 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 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 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 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," and 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]."

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 staff agrees with the licensee's assessment of the significance of the error. 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.

Background and Overview of ASTRUM

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), 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.

The licensee received approval to implement a plant-specific adaptation of the ASTRUM evaluation model in a license amendment issued on February 29, 2012 (ADAMS Accession No. ML12054A168). In the safety evaluation approving this license amendment, the NRC staff noted the significance of TCD, and acknowledged that the licensee would provide a report documenting the effect of accounting for the TCD error on the predicted peak PCT. The safety evaluation included the following passage:

> With respect to the fuel TCD issue, the licensee is not requesting approval of the PAD4TCD model to support future cycles of NAPS. Instead the licensee is using this unapproved code to assess the impact of TCD on ECCS performance, fuel mechanical design, and non-LOCA safety analyses. Currently approved Westinghouse methods will be maintained in the plants technical specifications... In accordance with 10 CFR 50.46(a)(3) reporting requirements, the licensee will submit a 30-day notification following startup of the reactor...

3.1 Summary Of Technical Information In The Report

The licensee's report indicated that the effect of the TCD error was 135 °F for NAPS Unit 1 and 101 °F for NAPS Unit 2. 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 Attachment 1 to the May 16, 2012 letter, and in the December 10, 2012, supplemental 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 Project1. The

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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.

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 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.

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.

In generating its estimate, the licensee executed large run sets to "stabilize the estimate of the PCT results. The effect of fuel TCD was then estimated by comparing the stabilized results between a rebaselined, pre-TCD run set and TCD run sets for fuel in its first and second cycle of irradiation" (ML12143A149).

Reported Results

Following the correction for TCD, the current predicted PCT for NAPS Unit 1 is 1987 °F, and for NAPS Unit 2, the current predicted PCT is 1972 °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 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 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. The licensee response to request for additional information item 2 at ADAMS Accession No. ML12356A300 clarifies that the explicit model is described in a March 7, 2012, Westinghouse letter to the Commission (ADAMS Accession No. ML12072A035). 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.

Once the thermal conductivity and related models were corrected, the NRC staff understood the above stabilization discussion to mean that the typical ASTRUM run set of 124 cases was expanded to a higher number by including additional cases to the original sample (rebaseline run set). Then, the run set was re-executed with adjustments to the models and parametric inputs that accounted for the TCD effects (TCD run set). The upper tolerance limit between the rebaseline run set and the TCD run set were then compared to generate the TCD estimate. Finally, the TCD estimate was then added to the licensing basis PCT (i.e., that obtained from analyzing the original, non-TCD corrected, set of 124 runs), to get the final result.

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 by re-executing a more limited subset of the original ASTRUM run set. 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 50.46 reports, the NRC 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 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 investigated the effects of TCD by generating a rebaseline run set and comparing the results to another, TCD-corrected, run set, the NRC staff did not identify significant issues with the estimation method.

The estimated effect of TCD at NAPS is 135 °F for Unit 1, and 101 °F for Unit 2. 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 1987 for Unit 1 and 1972 for Unit 2 respectively, which falls within the regulatory acceptance criterion of 2200 °F. Because the effect of TCD is consistent with other estimates of the same phenomenon, 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.

In its cover letter, the licensee stated the following:

> Before December 15, 2016, Dominion 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 based on the following milestones, which must be completed in order to perform a revised licensing basis LBLOCA analysis with an NRC-approved ECCS Evaluation Model (EM) that explicitly accounts for TCD:
>
> 1) NRC approval of a fuel performance analysis methodology that includes the effects of TCD. The new methodology for developing inputs to the LBLOCA EM would replace the current NAPS licensing basis methodology in WCAP-15063-P-A, Revision 1, which is referenced in Sections 4.2.1.3.1 and 4.4.3.4.2 of the NAPS Updated Final Safety Analysis Report (UFSAR).
>
> 2) NRC approval of a LBLOCA EM that includes the effects of TCD and accommodates the ongoing 10 CFR 50.46(c) rulemaking process. The new methodology would replace the current licensing basis analysis methodology, which is a plant-specific adaptation of WCAP-16009-P-A (Reference 2) as approved by the NRC in Reference 1.

In summary, the NRC staff reviewed the licensee's report estimating the effect of TCD on the LBLOCA analyses for NAPS Units 1 and 2. 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 comparing the effects of a TCD-corrected run set to a rebaseline run set, 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 finds 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 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, and (3) verify 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.

The staff notes that the licensee uses a plant-specific adaptation of ASTRUM, and that the "rebaseline" technique described above was used merely to estimate the effects of TCD. The statistical practices employed in generating the TCD estimate were not included in the scope of this review, since the licensee added the TCD estimate to its existing PCT. The "rebaseline" technique is not approved by this closure evaluation, and any evaluation models or other analytic methods based on this or similar techniques to accomplish an NRC-regulated function (i.e., UFSAR Chapter 15 Safety Analysis) would be subject to further NRC staff review and the conclusions associated therewith.

Principal Contributor: Benjamin Parks, NRR/DSS

Date: May 17, 2013

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President and Chief Nuclear Officer
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Sincerely,
/RA/
V. Sreenivas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Closure Evaluation

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