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

January 25, 2013

Mr. Paul A. Harden, Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 – EVALUATION OF REPORT DESCRIBING THE NATURE OF, AND ESTIMATED EFFECT ON, PEAK CLADDING TEMPERATURE RESULTING FROM A SIGNIFICANT EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL ERROR (TAC NOS. ME8409 AND ME8410)

Dear Mr. Harden:

By letter dated March 16, 2012 (Agencywide Document Access and Management System (ADAMS) Accession No. ML12079A111), supplemented by letter dated June 11, 2012 (ADAMS Accession No. ML12164A798), FirstEnergy Nuclear Operating Company (FENOC, the licensee), submitted a response to a U.S. Nuclear Regulatory Commission (NRC) information request made pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Paragraph 50.54(f), for Beaver Valley Power Station, Units 1 and 2 (ADAMS Accession No. ML120400672). The submittal referred to a letter from Westinghouse Electric Company dated March 7, 2012 (ADAMS Accession No. ML12072A035), that provided additional information. The 10 CFR 50.54(f) request was related to the estimated effect on peak cladding temperature (PCT) resulting from thermal conductivity degradation in the Westinghouse-furnished realistic emergency core cooling evaluation. The letter dated March 16, 2012, also stated that the response served as a 30-day report of a significant emergency core cooling system (ECCS) evaluation model change or error, in accordance with requirements of 10 CFR 50.46(a)(3)(ii).

The NRC staff has evaluated the March 16, 2012, 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). Further, the NRC 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 re-analysis is acceptable, in light of the information provided. The staff evaluation providing the basis for these conclusions is enclosed. This letter concludes the NRC staff efforts under TAC Nos. ME8409 and ME8410.

Please contact me at 301-415-2833, if you have any questions.

Sincerely,

Peter Bamford

Peter Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosure:
As stated

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

CLOSURE EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BEAVER VALLEY POWER STATION, 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. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated March 16, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12079A111), FirstEnergy Nuclear Operating Company (FENOC, 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 Beaver Valley Power Station (BVPS), Unit Nos. 1 and 2. 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 was supplemented by letter dated June 11, 2012 (ADAMS Accession No. ML12164A798), 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 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 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 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 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 the regulatory requirements and guidance discussed above 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 more 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 staff to verify that the evaluation model, as a whole, remains adequate. Finally, the NRC staff's review also established 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), to evaluate ECCS performance for BVPS, Unit 1. 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 loss-of-coolant accident (LOCA) scenarios. This result is used to show compliance with the 10 CFR 50.46(b)(1) acceptance criterion concerning PCT.

3.2 Overview of Code Qualification Document Method

The licensee evaluates ECCS performance for BVPS, Unit 2, using the NRC-approved Code Qualification Document (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 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.

3.3 Summary of Technical Information in the Report

The licensee's report dated March 16, 2012, indicated that the effect of the thermal conductivity degradation (TCD) error was +156 °F for Unit 1 and +10 °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. 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.

<u>TCD Error Correction</u>

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

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1. A non-proprietary version of WCAP-12945-P-A, WCAP-14747, was submitted to the NRC by letter dated March 19, 1998 (ADAMS Legacy Accession No. 9804070248). The NRC staff review of WCAP-12945-P-A and WCAP-14747 is documented in a letter and evaluation dated June 28, 1996 (ADAMS Legacy Accession Nos. 9607050086 and 9607050063, respectively). ADAMS Legacy documents may be obtained from the NRC's Public Document Room.

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.[2] 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 for Unit 1 and used to re-analyze the reference transient from the CQD analysis for Unit 2.

The licensee used PAD 4.0 + TCD to generate inputs to the ASTRUM and CQD evaluations, which is different from the existing analyses. In the existing 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 gap heat transfer properties. With the improved PAD correction, the FRAPCON 3.3[4] model with a revised degradation term[5] in WC/T is used for the initialization, instead. This 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 for Unit 1, 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. For Unit 2, the PCT effect estimate due to TCD was generated by re-executing the reference transient using corrected fuel initial conditions.

Additional Model Changes to Offset TCD Effects

BVPS Unit 1

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.

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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. 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.
4. Lanning, D. D., C. E. Beyer, and K. J. Geelhood. 2005. *FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties*, NUREG/CR-6534, Volume 4, prepared by Pacific Northwest National Laboratory, Richland, WA, for the U.S. Nuclear Regulatory Commission, Washington, D.C.
5. LTR-NRC-12-27, "Westinghouse Input Supporting Licensee Response to NRC 10 CFR 50.54(f) Letter Regarding Nuclear Fuel Thermal Conductivity Degradation (Proprietary/Non-Proprietary)," March 7, 2012 (ADAMS Accession No. ML12072A035).

In the submitted report, dated March 16, 2012, the licensee stated that the peaking factor margin changes accounted for a 305 °F reduction in the predicted PCT.

The licensee made additional design input changes in order to offset the increase in predicted PCT due to the TCD model error. The changes included the reduction of the upper bound steam generator tube plugging (SGTP) from 22 percent to 5 percent and an increase in the conservatively low containment pressure boundary condition. All of the changes made were to the model input and were not operating changes. The changes resulted in recapture of analytic safety margin. The result is a 180 °F reduction in the predicted PCT.

The net effect of incorporating all of the model changes, including an existing 2 °F rack-up item, was a reduction in predicted PCT of 327 °F.

BVPS Unit 2

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. As documented on Page 7 of Attachment 2 to the letter dated March 16, 2012, the licensee reduced steady-state and transient peaking factors, and reduced the hot assembly average power. These changes are attributable to a 190 °F decrease in predicted PCT.

Reported Results

Following the correction for TCD and the model change, the current predicted PCT for BVPS Unit 1 is 1834°F and the current predicted PCT for BVPS Unit 2 is 1837°F. By letter dated March 16, 2012, the licensee provided a proposed schedule for reanalysis 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 that the re-analysis will be submitted on, or before, December 15, 2016.

3.4 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 (RAI), to which the licensee responded via letter dated June 11, 2012. 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 ensure a high level of probability that the 2200 °F PCT acceptance criterion is not exceeded.

The NRC staff performed an audit of BVPS, Units 1 and 2, calculations and methodologies at the Westinghouse facility (ADAMS Accession No. ML12086A137). The staff reviewed the methods used to estimate the effect of TCD, as well as what the licensee changed to gain margin. By letter dated May 4, 2012 (ADAMS Accession No. ML121150501), the NRC staff issued an RAI, with eight questions for BVPS, Units 1 and 2, following the Westinghouse audit. Since the answers to these questions are important to the NRC staff's evaluation of this error notification, they are described in detail below.

Questions 1 and 2 asked the licensee to provide information on the Analysis of Record (AOR) ASTRUM inputs and their methods for choosing the number of cases run for the effects of TCD as well as the effects of TCD and the compensating model changes for BVPS Unit 1. The licensee responded with the AOR data requested, as well as justification for the number of cases run to find the effects of TCD. The staff reviewed the entirety of the data to verify that the bounding cases were identified and run. The staff concluded that the cases run were bounding and found the justification that the licensee provided for the number of cases and the choice of cases acceptable.

Question 3 asked the licensee to justify containment pressure changes made to obtain margin for Unit 1. The licensee responded, stating that a change was included in the assumed SGTP from 22 percent to 5 percent in the WC/T containment pressure boundary condition input. The release of mass and energy to containment increased, correspondingly. The NRC staff reviewed this response and found that the changes stay within operational limits, represent a more realistic estimate, and are therefore, acceptable.

Question 4 asked the licensee to justify the evaluation of reduced peaking factors to obtain analytic margin to offset the TCD effect for Unit 2. The licensee responded, specifying that the margin reduction came from a decrease in the maximum peaking factors considered in the large-break LOCA (LBLOCA) analysis, which are more conservative than the core operating limits report (COLR). The change, thus, represents a reduction in margin. The NRC staff agrees that the analysis peaking factors are more conservative than the allowable operating values contained in the COLR for Unit 2. The changes, thus, recapture margin from the analyzed peaking factors as compared to the allowable COLR limits. Since the COLR limits are enforced by the plant technical specifications (TSs), the staff finds that adequate control exists such that the plant will continue to be operated in accordance with the analysis limits.

Question 5 asked the licensee to address the Westinghouse analytic methods used to evaluate the effects of TCD. The licensee responded to this question, including its multiple parts (5a – 5f). For question 5a, the licensee stated key fuel parameters used for fuel temperature analyses were compared to a TCD analysis of a representative rod type. The licensee provided information on the fuel parameters and showed that the representative rod bounded the Unit 1 and Unit 2 rods, in most cases. In the cases where the representative rod was not bounding, the licensee provided information for why a more bounding characteristic bounded the situation. As a response to question 5b, the licensee provided values for coefficients used in the PAD 4.0 + TCD uranium dioxide thermal conductivity equation. The staff was able to review the equation and verify implementation. In its response to questions 5c and 5d, the licensee stated that there were no error corrections, code improvements, or miscellaneous code cleanup for Unit 1. For Unit 2, the licensee provided a table of various error corrections, code improvements, and miscellaneous code cleanup between the WC/T and HOTSPOT[6] code versions used in the AOR and the TCD effects evaluation. In question 5e, the licensee was asked to explain the differences between the HOTSPOT and PAD TCD models and the impact of the differences. The licensee provided information and graphs that supported the staff's evaluation of the differences and the impact of the differences. Finally, for question 5f, the licensee was asked to provide additional detail concerning the steady-state ASTRUM/CQD initialization process and

6. HOTSPOT is a one-dimensional conduction computer code used in the Westinghouse methodology to evaluate the effects of uncertainties associated with local parameters on the calculated PCT.

what fuel characteristics are adjusted within the models to obtain convergence among HOTSPOT, WC-T, and PAD 4.0 + TCD. The licensee provided parameters used to determine steady state convergence. The NRC staff reviewed the licensee's responses and concluded that the information provided supported the staff's conclusion that the evaluation model remains accurate.

Question 6 asked the licensee to explain how the changed design values will be verified during operation of the plant. For BVPS Unit 1, the steam generator tube plugging upper bound was changed. There are no associated TS limits or SRs. The number of tubes plugged and confirmation of adherence to safety analyses assumptions is confirmed during the reload process. The Unit 1 containment pressure boundary condition was also increased. The licensee stated the change came from margin included in the AOR from the actual containment evaluation code output. For Unit 2, the maximum peaking factors considered in the LBLOCA AOR were more conservative than the allowable COLR limits. The licensee also stated that no changes to the COLRs were required because the reduction came from margin in the AOR that was above the COLR limit. The licensee also explained the compensatory action that would be taken if the value of the peaking factors is found to be outside the limits. Peaking factor design values (the transient Heat Flux Hot Channel Factor (FQ) and Nuclear Enthalpy Rise Hot Channel Factor) have limits specified in the COLR and TS surveillance requirements (SRs). If they were found to not be met, the applicable action statements would be required to be followed. The steady state FQ and hot assembly average power parameters do not have COLR limits but are confirmed to be met during the reload process. The staff determined the response to this question provided adequate assurance of plant operation consistent with the analysis, based on the margin recovery, TS SRs, and the reload analyses controls of the plant.

Question 7 asked the licensee to explain the process for which the licensee and Westinghouse ensure that the LBLOCA analysis input values conservatively bound the as-operated plant values for those parameters. The licensee explained the reload process and the evaluation of plant changes resulting from loading different or new fuel into the core. The licensee also explained the generic fuel reload process and reliance on a bounding approach so that the safety analysis would be able to accommodate changes in fuel without having to perform new safety analyses. The plant generates a list of important LBLOCA analysis parameters during reload evaluation. They then confirm the parameters support the reload design and are evaluated with respect to LBLOCA analysis. The NRC reviewed the licensee's response and noted that this process is described in an NRC-approved topical report that is referenced in BVPS, Units 1 and 2, TS 5.6.3.b. Thus, the NRC staff concludes that the reload process represents a rigorous analytical approach, and the methodology reference in the TS provides sufficient control to ensure that the LBLOCA analysis input values conservatively bound the as-operated plant values.

Finally, Question 8 pertained to the process of using 10 CFR 50.59 to address changes to the analytic framework in licensing basis ECCS evaluation. The licensee stated that two types of changes were made to address the issue of TCD impact on PCT. The first involves changes that were made as inputs to the approved methodology. These changes were controlled under 10 CFR 50.59 and are being processed for inclusion in the Updated Final Safety Analysis Report. The second involves changes to the calculational framework which are outside of the as-approved evaluation model. The licensee stated that the provisions of 10 CFR 50.59 do not apply to these changes because the 10 CFR 50.46 regulation provides more specific criteria for

accomplishing such changes, pursuant to 10 CFR 50.59 (c)(4). Based on the information provided, the NRC staff does not object to the licensee's response to this question.

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 the PCT will not exceed 2200 °F following a LOCA. By letter dated March 16, 2012, the licensee provided a proposed schedule for re-analysis, stated as follows:

> On or before December 15, 2016, FENOC will submit to the Nuclear Regulatory Commission (NRC) for review and approval LBLOCA analyses that apply NRC approved methods that include the effects of fuel pellet TCD. The date for the analyses submittal is projected on the following milestones needed to perform a revised licensing basis LBLOCA analysis with an NRC-approved emergency core cooling system EM [Evaluation Model] that explicitly accounts for TCD:
>
> 1) Submittal by Westinghouse, to the NRC for review and approval, of revised fuel performance and LBLOCA EM methodologies that include the effects of TCD.
>
> 2) Prior NRC approval of a fuel performance analysis methodology that includes the effects of TCD. The new NRC-approved methodology would replace the current licensing basis methodology for the Beaver Valley Power Station, Unit No.1 that is described in WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005 and for the Beaver Valley Power Station, Unit No.2 that is described in WCAP-12945-P-A, Volumes 1 through 5, "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.
>
> 3) Prior 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 methodology, WCAP-16009-P-A for Beaver Valley Power Station, Unit No.1 and WCAP-12945-P-A for Beaver Valley Power Station, Unit No. 2.

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 re-analysis will be provided. 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).

In the licensee's submittal dated March 16, 2012, the proposed schedule for re-analysis was classified as a regulatory commitment. The NRC staff concludes that tracking of 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), and also the intent of the reporting requirements, as discussed in the statement of considerations for the realistic ECCS evaluations revision of 10 CFR 50.46. Further, the submittals dated March 7, 2012, March 16, 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 re-analysis is acceptable in light of the information provided.

Principal Contributors: B. Parks
J. Miller
P. Bamford

Date: January 25, 2013

Please contact me at 301-415-2833, if you have any questions.

Sincerely,

/RA/

Peter Bamford, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

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