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

December 31, 2013

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 – STAFF EVALUATION FOR REPORT CONCERNING SIGNIFICANT ECCS EVALUATION MODEL ERRORS/ CHANGES RELATED TO ECCS BYPASS AND UPPER PLENUM COLUMN WELDMENTS (TAC ME9719)

Dear Sir or Madam:

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46(a)(3), Entergy Operations, Inc., the licensee for Arkansas Nuclear One, Unit 1, submitted a report describing two significant errors/changes identified in the emergency core cooling system (ECCS) evaluation model, and an estimate of the effects of the errors/changes on the predicted peak cladding temperature. The report was submitted by letter dated March 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12080A120), as supplemented by letters dated December 17, 2012 (ADAMS Accession No. ML12353A489), and April 10 and October 22, 2013 (ADAMS Accession Nos. ML13101A267 and ML13296A744, respectively).

Based on its review, the U.S. Nuclear Regulatory Commission (NRC) staff concludes that the report submitted concerning ECCS evaluation model errors/changes pertaining to end of ECCS bypass and column weldments, satisfies 10 CFR 50.46 reporting requirements. The report and supplemental information provided by AREVA NP Inc. enabled the NRC 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 peak cladding temperature acceptance criterion promulgated by 10 CFR 50.46(b).

A copy of the NRC's staff evaluation is enclosed. If you have any questions, please contact me at 301-415-1480 or via e-mail at kaly.kalyanam@nrc.gov.

Sincerely,

Kaly Kalyanam

N. Kaly Kalyanam, Project Manager
Plant Licensing IV-2 and Decommissioning
Transition Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure:
Staff Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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STAFF EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REPORT DESCRIBING THE NATURE OF AND

ESTIMATED EFFECT ON PEAK CLADDING TEMPERATURE

OF A SIGNIFICANT EMERGENCY CORE COOLING SYSTEM

EVALUATION MODEL ERRORS/CHANGES

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46(a)(3), Entergy Operations, Inc., the licensee for Arkansas Nuclear One, Unit 1 (ANO-1), submitted a report describing two significant errors/changes identified in the emergency core cooling system (ECCS) evaluation model, and an estimate of the effects of the errors/changes on the predicted peak cladding temperature (PCT). The report was submitted by letter dated March 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12080A120), as supplemented by letters dated December 17, 2012 (ADAMS Accession No. ML12353A489), and April 10 and October 22, 2013 (ADAMS Accession Nos. ML13101A267 and ML13296A744, respectively). The licensee's response to a request for additional information (RAI) dated April 10, 2013, also referred to an AREVA NP Inc., letter submitted to the U.S. Nuclear Regulatory Commission (NRC) on March 28, 2013 (ADAMS Accession No. ML13091A075).

The NRC 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). The staff's review is discussed in the following sections of this evaluation.

Enclosure

2.0 REGULATORY EVALUATION

2.1 Requirements Contained in 10 CFR 50.46

The regulations in 10 CFR 50.46 provide the acceptance criteria for emergency core cooling systems for light-water nuclear power reactors. In particular, 10 CFR 50.46(a)(3)(i) requires, in part, that licensees

> ...shall 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 this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature difference by more than 50 °F [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 °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 Final Rule for revisions to the ECCS acceptance criteria published in the *Federal Register* on September 16, 1988 (53 FR 35996), which states, in part, that

> [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 discussion in the Final Rule 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 two errors/changes that affect the large-break loss-of-coolant accident (LBLOCA) analysis of record. The first item is an error in the determination of the end of ECCS bypass. The second item is a change in the evaluation model to include the effects of the upper plenum column weldments (CWs).

Based on the nature of the reported errors/changes, and on the magnitude of the effects on the PCT calculation, the NRC staff determined that a detailed technical review was 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 errors/changes, and to enable the staff to verify that the evaluation model, as a whole, remains adequate.

3.1 Summary of Technical Information in the Report

The licensee's report indicated two errors/changes that affect the PCT for the LBLOCA analysis. The first item is an error in the determination of the end of ECCS bypass resulting in a decrease in PCT of 80 °F. The second item is a change to include the effects of the upper plenum CWs resulting in a PCT increase of 80 °F. The nature of the errors, and the methods used to estimate their effects on the calculated PCT, is discussed in greater detail in the AREVA letter submitted to the NRC on March 28, 2013.

3.1.1 ECCS Bypass Error/Change

A mathematical error was identified with the control variables in the energy balance calculations used to determine the complete end of ECCS bypass in LBLOCA applications. As defined in the AREVA letter submitted on March 28, 2013, the complete end of bypass is achieved when all of the injected ECCS liquid reaches the lower plenum before core reflood analysis begins. When the ECCS flows can condense all of the steam flowing into the upper downcomer region, the end of blowdown has occurred. AREVA further defines the complete end of bypass time as the earliest end of blowdown time or the time at which the ECCS flows can condense all the steam flowing into the upper downcomer region. The error in the control variables incorrectly calculated the complete end of bypass time and liquid mass that should have remained in the vessel at the end of blowdown.

3.1.2 Estimation of the Effect of ECCS Bypass in the PCT Calculation

ANO-1 has 177 fuel assemblies and uses a lowered loop (LL) design Nuclear Steam Supply System. The effect of the ECCS bypass error was identified from analyses of a 205-fuel assembly plant. The control variables in the evaluation model are common to both 205-fuel assembly and 177-fuel assembly plants; therefore, the licensee determined that the error applies also to the ANO-1 ECCS evaluation. A blowdown reanalysis was performed using

RELAP5 (Reference 1) with corrected bypass variables for a 177-fuel assembly LL plant. The case analyzed was for a 2.506-foot peak power location at beginning of life.

With the error corrected, the analysis showed a reduction in the end of ECCS bypass time by roughly 2 seconds. When the end of bypass time occurred earlier, the amount of ECCS fluid that was not bypassed increased. The previously bypassed ECCS liquid added to the lower plenum and caused the bottom of core recovery to occur earlier. This caused the core refill period to become shorter, and reflood began sooner. Since core reflood occurred sooner, core cooling began earlier, which caused the predicted PCT to decrease.

The licensee reported that the limiting PCT decreased by 90.3 °F for the ruptured node and by 45.4 °F for the unruptured node in a 177-fuel assembly LL plant. The licensee conservatively reported a PCT decrease of 80 °F for ruptured segments and 40 °F for unruptured segments as a result of the end of bypass time occurring earlier in the LBLOCA analysis for 177-fuel assembly LL plant.

The results from small-break loss-of-coolant accident (SBLOCA) analyses were not impacted since the same ECCS bypass modeling is not used for SBLOCA.

3.1.3 Column Weldment Model Error/Change

The change in the evaluation model was caused by the inability of the RELAP5 model to account for the effects of CWs over the hot bundle. Sensitivity studies were performed with upper plenum CW added to the evaluation model. Column weldments are also known as control rod guide tube housings. They support the control rod and allow a portion of the flow exiting from the core channels underneath them to reach the upper head. This makes them important in determining the temperature of the fluid reaching the upper head.

The change estimate of the CW issue was developed using an iterative process that adapted a 205-fuel assembly plant's model to the 177-fuel assembly plant design. A LBLOCA RELAP5 model was initially developed for a 205-fuel assembly plant. Then, the CW design from the 205-fuel assembly plant was modeled on top of the hot bundle of a 177-fuel assembly plant with an LL design. This simplified approach was used because the CW design details for a 177-fuel assembly plant were not readily available when the CW modeling issue was first assessed. When the effect of CWs was being analyzed for a raised loop (RL) plant, additional details of the CWs for a 177-fuel assembly plant had been developed.

3.1.4 Estimation of the Effect of Column Weldments in the PCT Calculation

The licensee reported that when CWs modeled for a 205-fuel assembly plant were incorporated over the hot channel in a 177-fuel assembly LL plant, the RELAP5 blowdown analysis resulted in an increase in end-of-blowdown fuel temperature of 35.6 °F for the peak unruptured fuel cladding segment. The licensee estimated this temperature increase to be 40 °F. The fuel temperature increase observed at end-of-blowdown is generally maintained throughout the refill period because the fuel heats up adiabatically during this phase. The actual temperature increase at beginning-of-core recovery was analyzed to be 30.2 °F, indicating that estimation of 40 °F was conservative.

Typically, the PCT will increase in proportion to the end-of-blowdown fuel temperature. The licensee notes that due to the metal-to-water reaction inside the fuel, there is a two-to-one variation in PCT change between ruptured and unruptured segments. Therefore, the licensee estimated that the ruptured cladding segment would experience an 80 °F increase in PCT, estimated based on a 40 °F increase in end-of-blowdown temperature.

When the licensee analyzed this case using BEACH (Reference 2), there was a favorable shift of rupture time into the blowdown phase causing the ruptured segment PCT to increase by 26.2 °F from the BEACH calculation. BEACH calculated an increase in PCT of 11.5 °F for the unruptured fuel segment. (This is compared to an increase in fuel temperature of 35.6 °F from the end-of-blowdown calculation using RELAP5.)

Since analysis using an LL design with a 177-fuel assembly plant was not completed, the geometric properties between a 205-fuel assembly plant and 177-fuel assembly plant were compared. When CWs are explicitly modeled in the upper plenum model of the LBLOCA analyses, the structures impede coolant flow entering the top of the core after the core flow reverses direction. The cross flow into the lower CW holes and slots from the upper plenum are restricted in the reverse-flow direction due to the form losses. The hole and slot sizes in the lower CWs are identical between the 205-fuel assembly and 177-fuel assembly plant designs; therefore, the controlling resistances and flow areas for cross flow into or out of the lower CW are the same.

Even though the hole and slot sizes in the lower CWs are identical, the flow areas inside the CWs for the 177-fuel assembly plants are 7 percent smaller than those in a 205-fuel assembly plant. This difference is related to the 15x15 versus 17x17 fuel bundle arrangements and the 177-fuel assembly plant having a different number of control rods than the 205-fuel assembly plant.

Analyses for a 177-fuel assembly CW plant with a LL design were not explicitly completed. A sensitivity study was done using a 177-fuel assembly CW in an RL design. The results of this study show an increase in PCT of 3 °F when compared to a 205-fuel assembly CW with a LL design. The PCT change was due to the difference in flow area inside the CW between the 177- and 205-fuel assembly plants.

Compared to a case with only the corrected ECCS bypass calculation, it was observed that modeling the 177-fuel assembly CW in an RL plant increased PCT by 8.9 °F for an unruptured segment. For the LL plant, the PCT increased 11.5 °F for an unruptured segment, after modeling a simplified version of the 205-fuel assembly CW, compared to a case with only the corrected ECCS bypass calculation.

The licensee used a generic increase of 40 °F for unruptured fuel segments and 80 °F for ruptured fuel segments for the effect of CWs. The largest increase in PCT for the LL design with ruptured fuel segments was explicitly analyzed to be 26.2 °F when CW were included in the analyses. The generic increase in PCT of 80 °F that the licensee applied to rupture limited LL plants is considered bounding and conservative because there is significant margin over the calculated PCT increase of 26.2 °F. The 3 °F increase in PCT from analyzing CW in a 177-fuel assembly plant is also well within the margin.

An evaluation of the impact of CW on SBLOCA analyses was performed by the licensee. It was concluded that SBLOCA analyses are unaffected by the CW modeling because the net flow remains upward during these slower evolving transients.

3.1.5 Reported Results

Following the correction for ECCS bypass and the CW model change, the current predicted PCT for LBLOCA at ANO-1 is 2008.1 °F.

3.2 Summary of Staff Evaluation

In its evaluation, the NRC staff reviewed (1) the approach used to estimate the effects of the ECCS bypass error and the effects of upper plenum CWs, (2) the estimated effect of both errors/changes, and (3) the licensee's proposal to not perform a reanalysis in consideration of the approach used to estimate the effects of the errors/changes. As discussed in the following paragraphs, the NRC staff determined the licensee's estimates of the error and changes are acceptable.

To estimate the effects of the ECCS bypass error, the licensee analyzed the effect of correcting the control variables in the energy balance equation used to determine the complete end of bypass time in LBLOCA applications. The effect was identified from analyses of a 205-fuel assembly plant. The control variables are common to both 205-fuel assembly plants and 177-fuel assembly plants; therefore, the correction is applicable to ANO-1. The NRC staff determined that this estimate was acceptable because explicit analysis was completed to evaluate the effect of correcting the control variables.

To estimate the effects of upper plenum CWs, the licensee included CWs over the hot channel in a model of a 205-fuel assembly plant. This model was incorporated over the hot channel in a 177-fuel assembly LL plant and a RELAP5 blowdown analysis was completed. Sensitivity studies were performed using CW modeled for a 177-fuel assembly RL plant. The effects of the studies showed the generic estimate of the effect of CW was conservative for LL plants. The NRC staff determined that this estimate was acceptable because the effect of including upper plenum column weldments was explicitly analyzed.

The licensee estimated the effect of the ECCS bypass error to be a decrease in PCT of 80 °F for ruptured fuel segments and 40 °F for unruptured fuel segments. The licensee estimated the effect of upper plenum CW to be an increase in PCT of 80 °F for ruptured fuel segments and 40 °F for unruptured fuel segments. The current predicted PCT for LBLOCA at ANO-1 is 2008.1 °F.

The NRC reviewed the current estimated PCTs at ANO-1, and determined that the reported PCTs continue to remain below the 10 CFR 50.46(b)(1) acceptance criterion and, therefore, the reported PCTs remain acceptable.

As stated in 10 CFR 50.46(a)(3)(ii), if the change or error is significant, 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 §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."

In an RAI dated October 26, 2012, as revised by an RAI dated November 8, 2012 (ADAMS Accession Nos. ML12300A365 and ML12313A440, respectively), the NRC staff requested that the licensee "justify not providing a schedule for reanalysis or taking other action to show compliance with 10 CFR 50.46." In its response to the RAI dated December 17, 2012, the licensee indicated that both the error and change to the ECCS evaluation model that were presented in the 10 CFR 50.46 30-day report were analyzed in detail and that the impact of both items does not result in challenge to the 10 CFR 50.46(b) acceptance criteria. The licensee also concluded that the evaluation model is considered adequate since the error and change have been analyzed and there are no other known errors or changes in the model at this time. The NRC staff determined that the PCT error evaluations are supported by explicit analysis using the Babcock & Wilcox (B&W) plant ECCS evaluation model, and the error-adjusted LBLOCA PCTs for ANO-1 remain below the 10 CFR 50.46(b) regulatory acceptance criteria.

The NRC staff issued another RAI on August 20, 2013 (ADAMS Accession No. ML13232A354), asking the licensee to justify how the generic analysis for the B&W plant ECCS evaluation model satisfies the requirement, in 10 CFR 50.46(a)(1)(i), to calculate ECCS cooling performance "in accordance with an acceptable evaluation model". The RAI states, in part, that

> In light of the presently reported, significant, estimated effects of errors and changes, explain how the present ECCS cooling performance has been calculated in accordance with an acceptable evaluation model, such that any other action, as provided in 10 CFR 50.46(a)(3), has been taken to show compliance with 10 CFR 50.46 requirements, including those contained in 10 CFR 50.46(a)(1).

The licensee uses an evaluation model that conforms to Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 to evaluate ECCS performance. The use of an Appendix K-based evaluation model leads to a conservative estimate of PCT. Included in the licensee's RAI response dated October 22, 2013, is clarification that the current evaluation model (with the errors corrected) concluded the actual net PCT would decrease, but the licensee reported a net change of zero in PCT. Therefore, the existing results remain conservative compared to the use of an evaluation model that both conforms to Appendix K and explicitly corrects the reported changes and errors. Based on the above, the NRC staff concludes that the licensee's explicit analysis of both significant errors paired with the conservative estimate of PCT using an already conservative evaluation model is acceptable and satisfies all requirements of 10 CFR 50.46.

In summary, the NRC staff reviewed the licensee's report and supplemental information estimating the effect of the ECCS bypass error and CWs on the LBLOCA analyses for ANO-1. Since the evaluation included explicit analyses of the ECCS bypass error and CWs in the evaluation model, the NRC staff concludes that the error estimates are acceptable.

4.0 CONCLUSION

Based on the above, the NRC staff concludes that the report submitted pursuant to 10 CFR 50.46(a)(3), concerning ECCS evaluation model errors/changes pertaining to end of

ECCS bypass and CW satisfies the intent of the 10 CFR 50.46 reporting requirements. The report and supplemental information provided by AREVA NP Inc. 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 of 10 CFR 50.46(b).

5.0 REFERENCES

1. Rosenburg, S., memorandum to Ralph Landry, U.S. Nuclear Regulatory Commission, "Safety Evaluation Report - BAW-10164P-A, Revision 6, 'RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis,' (TAC MD2187)," dated February 16, 2007 (not publicly available).

2. Framatome ANP, Inc., BAW-10166P-A, Revision 5, "BEACH – A Computer Program for Reflood Heat Transfer during LOCA," November 2003 (not publicly available).

Principal Contributors: A. Guzzetta and B. Parks

Date: December 31, 2013

A copy of the NRC's staff evaluation is enclosed. If you have any questions, please contact me at 301-415-1480 or via e-mail at kaly.kalyanam@nrc.gov.

Sincerely,

/RA/

N. Kaly Kalyanam, Project Manager
Plant Licensing IV-2 and Decommissioning
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