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U.S. Nuclear Regulatory Commission
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Generic Implications of Issue Identified During an NRC Audit of Calculations Supporting an Extended Power Uprate Submittal

Ref. 1: Letter, D.T. Langley (Tennessee Valley Authority) to Document Control Desk (NRC), "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specifications (TS) Change TS-418 - Extended Power Uprate (EPU) - Supplemental Response to Request for Additional Information (RAI) Rounds 3 and 18 and Response to Round 20 Fuels Methods RAIs (TAC Nos. MD5263 and MD5264)," September 19, 2008.

In a January 14, 2009 conference call between AREVA NP Inc. (AREVA NP) and the NRC, the NRC requested that AREVA NP provide a description of the actions performed to address generic implications of an issue identified during an NRC audit of calculations supporting an Extended Power Uprate (EPU) submittal conducted in August 2008 at the Richland, Washington facility. The issue was addressed during the audit and in Response SRXB-91 to an NRC Request for Additional Information (Reference 1). The issue and the actions to address potential generic implications are described for the NRC's information in Attachment 1.

If you have any questions related to this informational submittal, please contact Mr. Alan B. Meginnis, Product Licensing Manager at 509-375-8266 or by e-mail at alan.meginnis@areva.com.

Sincerely,

Ronnie L. Gardner

Ronnie L. Gardner, Manager
Corporate Regulatory Affairs
AREVA NP Inc.

Enclosure

cc: H.D. Cruz
Project 728

Y001
T010
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Attachment 1

The Ohkawa-Lahey void-quality correlation is used in the COTRANSA2, XCOBRA, and XCOBRA-T computer codes. The correlation was developed based on data from several different test geometries. Relative to the data from tests performed for the ATRIUM-10 fuel geometry, the NRC noted that the Ohkawa-Lahey correlation has a slight bias and under-predicts the ATRIUM-10 data. The NRC requested an assessment of the impact that a bias in the void-quality correlation would have on the calculated peak reactor vessel pressure for the anticipated transient without scram (ATWS) and ASME overpressurization analyses. AREVA NP performed sensitivity analyses to assess the impact by using an adjusted void-quality correlation that removes the bias for the ATRIUM-10 data. Preliminary results from these sensitivity studies were presented to the NRC during the August audit in Richland. The analyses indicated that the bias in the Ohkawa-Lahey void-quality correlation could result in a 10 psi under-prediction of the peak reactor vessel pressure in the ATWS analysis. A similar study for the ASME overpressurization analysis indicated that the peak pressure could be under-predicted by up to 7 psi.

The impact of the correlation bias was also assessed for transient analyses performed to determine the operating limit MCPR (OLMCPR). These analyses demonstrated that the bias had an insignificant effect on the calculated results for the events analyzed to determine the OLMCPR.

In response to the NRC question, AREVA NP proposed that the calculated peak pressure be increased by 10 psi for ATWS analyses and by 7 psi for ASME analyses supporting the EPU submittal until additional calculations are performed to demonstrate adequate conservatisms to offset the impact of the void-quality correlation bias.

The issue identified above for the Ohkawa-Lahey correlation is not unique to the EPU submittal. Therefore, an internal AREVA NP Condition Report was initiated on August 29, 2008 to determine the potential impact on operating plants. A review was performed of all ATWS and ASME analyses performed by AREVA NP supporting currently operating or soon to be operating plants. The review determined that adequate margins exist for all plants to offset potential increases in calculated peak pressure due to the correlation bias. Because adequate margins exist, the Condition Report issue was determined to not be a deviation and thus does not result in a significant safety hazard or risk of violating a safety limit as defined in 10 CFR 21.

The Condition Report also requires the following interim action be performed for ATWS and ASME analyses until the issue is closed:

> For ATWS and ASME over-pressurization analyses performed with COTRANSA2, one of the following shall be performed and documented in the calculation notebook:

- Demonstrate that there is sufficient margin (10 psi for ATWS, 7 psi for ASME) to the calculated peak reactor vessel pressure to offset the potential penalty.
- Demonstrate that there are sufficient conservatisms (10 psi for ATWS, 7 psi for ASME) in the calculation to offset the required penalty. These conservatisms must be modeling assumptions or biased input parameters that exceed methodology or Technical Specification requirements.

The Condition Report addressing the issue will remain open until the NRC issues an audit report or Safety Evaluation for the EPU submittal. At that time, analysis procedures will be revised as needed to address any NRC requirements relative to using COTRANSA2 for ATWS or ASME overpressurization analyses.