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

August 23, 2011

Mr. R. M. Krich
Vice President, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING TECHNICAL SPECIFICATION TS-473, AREVA FUEL TRANSITION (TAC NO. ME3775)

Dear Mr. Krich:

By letter dated April 16, 2010, Tennessee Valley Authority submitted a request for a Technical Specification change to support the licensee's plans to transition to AREVA fuel.

As part of the U. S. Nuclear Regulatory Commission (NRC) staff's review of the application, the staff conducted an onsite audit of the AREVA EXEM BWR-2000 emergency core cooling system evaluation model insofar as it has been applied to support the transition to AREVA fuel and safety analysis methods at Browns Ferry Nuclear Plant, Unit 1. The audit was conducted the week of July 18, 2011, at AREVA's Richland, Washington, facilities.

Based on the results of the audit, the NRC staff finds that a response to the enclosed request for additional information is needed before we can complete the review.

This request was discussed with Mr. T. Hess of your staff on August 16 and 19, 2011, and it was agreed that a response would be provided within 45 days of the issuance of this letter.

If you have any questions, please contact me at (301) 415-1055.

Sincerely,

Siva P. Lingam for
Christopher Gratton, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosure: Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

TRANSITION TO AREVA FUEL

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1. Please provide a revised Emergency Core Cooling System (ECCS) Evaluation Summary that provides a detailed description of the most severe loss-of-coolant accident (LOCA) analysis, along with a description of other break sizes, locations and other properties that were evaluated to support the determination that the most severe postulated LOCA has been calculated.

2. Please provide a detailed description of the model changes made to address the staff's concern with the evaluation model's application.

3. Please provide the results of a sensitivity study demonstrating the effect of the EXEM-BWR 2000 pressure control assumptions on the break spectrum. Please include results for the limiting break size as determined using the modified EXEM-BWR 2000 analyses (0.21 ft^2) and for a smaller break size that would result in a delayed pressurization following a level-driven main streamline isolation.

4. Please provide a summary of the break spectrum results that include sufficient detail to compare the break spectra for each combination of power shape, core flow state point, single failure, and break geometry.

5. Please explain why the break spectrum results exhibit slightly discontinuous behavior in the intermediate range of break sizes. Identify the significant model aspects that are causing the behavior and provide an estimate or description of the impact on the evaluation.

6. Please determine the cause of the intermediate temperature transient observable in the plots of peak cladding temperature vs. time for the intermediate break size cases and provide a summary explanation. Justify the validity of the results, given the temperature trends depicted.

7. Title 10 of the *Code of Federal Regulations*, Section 50.46, requires ECCS cooling performance to be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. At the time EXEM-BWR 2000 was approved, ECCS research suggested that the large-break LOCA was generally limiting for boiling-water reactors, and there appears to be little consideration of post-power uprate plant operation.

Since the ECCS research was compiled and documented in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," operating experience has shown that the small break scenario can in fact result in a more limiting event. Because the small-break accident is limiting, AREVA Topical Report ANP-2908(P) includes a number of explicitly analyzed ancillary line breaks, since these breaks are smaller in size. The general trend is a liquid blow down at high pressure until the break uncovers, followed by a depressurization of the reactor coolant system caused by steam exiting the break. The analysis results indicate that, absent any emergency core cooling, the steam flow pressure reduction is a dominant mechanism in the event.

Please provide an analysis of the rupture of the bottom head drain line (which would not include the pressure reduction associated with break uncover) to demonstrate that the initial heatup would not contribute to the limiting peak cladding temperature. This analysis should consider the most limiting of the Battery Board B failure and the high-pressure coolant injection failure.

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