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

May 27, 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 NOS. 2 AND 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING A REQUEST TO ELIMINATE LOW PRESSURE COOLANT INJECTION PUMP MOTOR GENERATOR SETS (TAC NOS. ME5796 AND ME5799)

Dear Mr. Krich:

By letter dated February 25, 2011 (Agencywide Documents Access and Management System Accession No. ML110600408), Tennessee Valley Authority (TVA, the licensee) submitted a request for a Technical Specification (TS) change that would delete Browns Ferry Nuclear Plant (BFN), Unit Nos. 2 and 3, TS Surveillance Requirement (SR) 3.5.1.12. The SR 3.5.1.12 requires the verification of the capability to automatically transfer the power supply from the normal source to the alternate source for each Low Pressure Coolant Injection subsystem inboard injection valve and each recirculation pump discharge valve on a 24-month frequency.

Based on our review of your submittal, the Nuclear Regulatory Commission (NRC) staff finds that a response to the enclosed request for additional information (RAI) is needed before we can complete the review.

This request was discussed with Mr. Tom Mathews of your staff on May 12, 2011, and it was agreed that a response would be provided within 30 days from this date. During the call, the NRC staff provided the following clarifications about the questions contained in the RAI:

- For RAI IV-3: For the sensitivity studies, please provide figures analogous to Figures 6.1, 6.2, 6.4, 6.5, and 6.8-6.19 from ANP-2908(P). For both the lower- and upper-lower plenum components, provide plots of liquid mass and mass flow rates at the component boundaries as functions of time. For the renodalized lower plenum component (TVA has proposed to renodalize only the upper lower plenum), please also provide plots of two-phase level and fluid quality as functions of time. The quality plots could be generated for the inlet and exit nodes only.

- For RAI V: The NRC staff expects that the break nodalization will reflect the limiting postulated bottom head drain line break that is reflective of BFN configuration. If the reactor water cleanup line has check valves downstream of the recirculation loops, their presence should be reflected in the break nodalization.

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

Sincerely,

Christopher Gratton

Christopher Gratton, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

Enclosure: Request for Additional Information

cc w/ enclosure: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

REGARDING ANP-2908(P),

"BROWNS FERRY UNITS 1, 2, AND 3 105% OLTP

LOCA BREAK SPECTRUM ANALYSIS"

REQUEST TO ELIMINATE LOW PRESSURE COOLANT INJECTION

MOTOR GENERATOR SETS

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NO. 50-260 AND 50-296

By letter dated February, 25, 2011, Tennessee Valley Authority, the licensee for Browns Ferry Nuclear Plant (BFN), Unit Nos. 1, 2, and 3, requested approval of a license amendment supporting the deletion of low pressure coolant injection motor generator sets at BFN Units 2 and 3. The Nuclear Regulatory Commission (NRC) staff requires the following additional information regarding ANP-2908(P), "Browns Ferry Unit Nos. 1, 2, and 3 105% OLTP [Original Licensed Thermal Power] LOCA [Loss-of-Coolant Accident] Break Spectrum Analysis," which supports the license amendment request, to complete its review.

I. Results – 0.25 ft^2 SF-BATT|BB Top-Peaked Axial

1) Identify the reactor scram signal.

2) Explain whether main steam line isolation is modeled, the source of the isolation signal, the time of isolation, and the effect that steamline isolation has on the accident sequence.

3) The NRC staff's own analyses, as well as additional data gathered by the staff during the BFN power uprate reviews, have indicated that steady-state bundle exit void fractions are quite high. For core conditions matching the small break LOCA initial conditions, provide steady-state void fraction as a function of assembly elevation for a hot channel from XCOBRA and indicate the limiting plane analyzed in HUXY. Justify any significant discrepancies between XCOBRA and the RELAX/HUXY model.

4) Provide a plot of upper plenum liquid mass with higher resolution from 0-10000 lb from the period between 300 seconds and the end of the transient. Provide similar plots (i.e., increased resolution at the region of interest) of liquid mass for the core average, bypass, and hot channel regions and the lower plenum.

Enclosure

5) Identify the primary sources of increased lower plenum liquid mass from 350-450 seconds.

6) Provide a higher resolution plot of hot channel inlet flow rate from 350-500 seconds. Trace the zero line through the plot.

7) Explain the phenomena causing the [[ ]] drop in temperature of the hot channel coolant. What are the sources and directions of liquid flow in this time period?

8) Explain what causes [[ ]] in the 50-75 second timeframe.

9) Justify the selection of fuel channel nodalization.

10) Provide information concerning the thermal-hydraulic characteristics of the bypass region (mixture level, liquid mass, quality).

II. Results – Break Spectrum, SF-BATT|BA, Pump Discharge

1) Explain why the predicted peak centerline temperature (PCT) [[ ]] interval for the mid-peaked spectrum.

2) Explain similar trends on the top-peaked spectrum on the [[ ]] intervals, and on the [[ ]] intervals.

3) Provide plots analogous to Figures 6.1 – 6.19 of ANP-2908(P) for each of the break sizes identified in 1 and 2 above to reinforce the discussion. Also provide tabulated sequences of events.

III. Results – Break Spectrum, SF-BATT|BB

Explain why the mid- and top-peaked power shape PCTs are inverted for the 0.45 ft^2 break. Provide plots for the [[ ]] breaks and justify the coarseness of the break spectrum. Also provide tabulated sequences of events.

IV. Initial Conditions and Input Parameters

1) The emergency core cooling system fluid temperature has been reduced from 125 °F in previous analyses to 120 °F. Explain and provide a justification for this change.

2) Provide a copy of ANP-2912(P), Revision 0, "Browns Ferry Units 1, 2 and 3 105% OLTP LOCA Parameters Document," which is referenced in ANP-2908(P).

3) Provide a nodalization sensitivity study of the lower plenum for the PCT-limiting case.

V. Adequacy of Analyzed Break Spectrum

Title 10 of the *Code of Federal Regulations*, Section 50.46, requires emergency core cooling system (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, 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, 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, especially in the cases that delay the automatic depressurization system actuation.

Please provide an analysis of the rupture of the bottom head drain line, the accident sequence for which would not benefit from the pressure reduction associated with break uncovery, to demonstrate that the initial heatup would not contribute to the limiting PCT.

VI. Fuel Thermal Conductivity Degradation

Explain whether and how the RODEX2 fuel centerline temperature inputs to the RELAX/HUXY analyses are corrected for issues identified in NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation." If they are not corrected, explain why not.

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

Sincerely,

/RA/

Christopher Gratton, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
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Docket Nos. 50-260 and 50-296

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