September 15, 2006

Mr. Karl E. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 3 - REQUEST FOR ADDITIONAL INFORMATION REGARDING AMERICAN SOCIETY OF MECHANICAL ENGINEERS SECTION XI, INSERVICE INSPECTION PROGRAM SUBMITTAL OF THIRD 10-YEAR INSPECTION INTERVAL PROGRAM - RELIEF REQUEST NO. 3-ISI-21 (TAC NO. MC8795)

Dear Mr. Singer:

By letter to the Nuclear Regulatory Commission (NRC) dated October 19, 2005, the Tennessee Valley Authority (TVA) submitted Relief Request 3-ISI-21 for Browns Ferry Nuclear Plant, Unit 3 from the inservice inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. TVA proposes to adopt risk-informed selection of piping welds for examination.

Based on our review of TVA's submittal, 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 TVA staff on July 10, 2006, and it was agreed that a response would be provided within 30 days from the issuance of this letter.

If you have any questions, please contact me at (301) 415-2315.

Sincerely,

/RA/

Eva A. Brown, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-296

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

RISK-INFORMED INSERVICE INSPECTION RELIEF REQUEST

THIRD 10-YEAR INSERVICE INSPECTION PROGRAM RELIEF REQUEST 3-ISI-21

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 3

DOCKET NO. 50-296

1. On pages 31 and 32 of Relief Request (RR) 3-ISI-21 dated October 19, 2005, Section 7.0 Parts (1) and (2) contains the following:

    Note: Class 1 (Class 2) piping welds shall be in accordance with the RI-ISI [risk-informed inservice inspection] additional examination requirements of [American Society of Mechanical Engineers (ASME)] Code Case N-577, as outlined in Section 7.12 of this program.

    Section 7.12.5.4.J. states: “An evaluation shall be performed to establish when those examinations are to be conducted.” Use of an evaluation to determine when a second sample expansion is to be performed is inconsistent with regulations. If flaws or relevant conditions are identified, sample expansions are to occur during the current outage in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB-2430(b). The licensee needs to address the time frame in which sample expansions will be performed.

2. Of the items selected for RI-ISI listed in the table included in Attachment 1, state how many are socket welds. Provide a breakdown of the examination technique to be performed on the items selected for examination, include the frequency of examination.

3. Page 61, Section 7.12.4, references Part 6 of Table R-A and 3-SI-4.6.G-A. These references are not provided. The licensee also lists ASME Code Case N-577 as part of the guidance used to develop its RI-ISI program. The staff has not endorsed the use of Code Case N-577. The licensee needs to provide the references or explain the licensee’s Risk-Informed Process. The references or explanation needs to include how the program was developed, using what guidance, and explain any deviations from the referenced guidance.

4. Page 64 addresses the Corrective Action Program. The licensee states that “For Code Piping categorized as High Safety Significance (HSS) the corrective action shall be consistent with the provision of ASME Code Section XI.” Describe what corrective action measures will be used for Low Safety Significant (LSS) Code piping.

Enclosure

5. Define the smallest diameter pipe included in the scope of the RI-ISI program for Class 1 and 2 piping. Specify whether the scope included all Class 1 and 2 piping or if there was a defined minimum diameter. Provide justification for the defined scope.

6. Attachment 1 to RR 3-ISI-1 indicates that the number of inspections was reduced from 100 in the second interval to 71 proposed for the third interval. Notable are reductions in examinations of Category A intergranular stress-corrosion cracking (IGSCC) susceptible piping associated with the reactor recirculation, reactor water cleanup, and core spray systems, as well as of Category C IGSCC-susceptible piping associated with the core spray system.

   Tennessee Valley Authority (TVA) states on page 190 that "deletions from the previous program are entirely attributable to lower failure rates due to the implementation of the hydrogen water chemistry/noble metal injection program, with the corresponding impact on IGSCC."

   Provide a description of TVA's methodology and an explanation of how the estimated reduction in the failure rates propagated through the methodology result in the significant reduction in the number of inspections for the third 10-year interval. The explanation should address the following specific questions the staff has developed based on the information provided in the submittal.

   A. Explain precisely how TVA was able to justify the deletion of each of the discontinued inspections in its proposed RI-ISI program for the third interval. For those welds in segments recategorized from HSS to LSS, data for both the previous program and the proposed program, similar to that provided in Enclosure 2 of TVA's response to Requests for Additional Information dated January 18, 2000, is suggested. For deleted welds in segments still considered HSS, data in the format of Table 3.8-1 of TVA's original submittal is suggested. Also, add to this table the previous and proposed failure rate, core damage frequency (CDF), and risk reduction worth data for each of these welds. If certain welds with a previous "quantified failure rate" now have a "zero failure rate," provide documentation to demonstrate that the traditional ASME Section XI criteria are being met.

   B. It was observed that, despite the net reduction of proposed inspections, there were a few added inspections (e.g., two new intergranular stress-corrosion cracking Category C locations in the reactor recirculation system). Explain the reason for these additions.

7. A January 23-26, 2006 Nuclear Regulatory Commission (NRC) audit of the licensee's probabilistic risk assessments (PRAs) noted significant changes to the PRA model that can impact CDF, large early release frequency (LERF), and, ultimately, risk reduction worth. Many of these changes are described in a report from ERIN Engineering and Research, Inc. (ERIN), August 2003 (provided to the staff during this audit) which covers Levels A and B Facts and Observations from the industry peer certification of the Browns Ferry Nuclear Power Plant, Unit 3 (BFN3) PRA, as well as observations during a separate review by ERIN, identified as an evaluation of PER BFPER970822 R0.

Given the numerous changes to TVA's PRA, describe in more detail the process used to evaluate the impact of these changes on the current risk-ranking of BFN3's pipe segments. As part of the description, indicate whether or not the conditional core damage probability (CCDPs) of the segments with the current PRA model were recalculated.

If TVA did perform a recalculation, provide the date, revision number, base CDF and LERF of the PRA model used for redetermining CCDPs, ÎCDFs, or conditional CDF of the pipe segments.

If TVA did not perform a recalculation, explain why the changes made to the PRA in connection with the Facts and Observations and PER item resolutions of August 2003 do not impact the risk-ranking of BFN3's pipe segments.

8. The above audit (Agencywide Documents Access and Management System accession number ML060440588) assessed that the significant changes to the PRA model fit the definition of a PRA upgrade as defined in Section 2 of the ASME PRA standard. The audit also identified a lack of peer review of this upgrade, which is contrary to the guidance in the ASME PRA standard, Section 5.4, and reinforced by NRC Regulatory Guide 1.200 (page 1.200-49).

   Since a peer review has not been conducted, provide the following information:

   A. A description of the review processes TVA employed in conjunction with the upgrades to the PRA models. Along with this, describe the level of expertise of the reviewers.

   B. An evaluation of the impact of the non-EPU related modeling errors specifically noted in Section 3.8 of the audit report (also documented in the BFN3 Corrective Action Program as Problem Evaluation Report (PER) No. 96035) on the RI-ISI application. In other words, if these errors are corrected, what impact will this have on the relative importance of BFN3's pipe segments?

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BROWNS FERRY NUCLEAR PLANT

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