July 30, 2004

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3 LICENSE RENEWAL APPLICATION (TAC NOS. MC1704, MC1705 AND MC1706)

Dear Mr. Singer:

By letter dated December 31, 2003, Tennessee Valley Authority (TVA) submitted an application pursuant to 10 CFR Part 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2 and 3, for review by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified areas where additional information is needed to complete the review. Specifically, the enclosed requests for additional information (RAIs) are from the Scoping and Screening Audit performed the week of June 7-10, 2004 by the Division of Inspection Program Management, Quality and Maintenance Section.

Based on discussions with Gary Adkins of your staff, a mutually agreeable date for your response to the RAIs is within 30 days of the date of this letter. If you have any questions regarding this letter or if circumstances result in your need to revise the response date, please contact me at (301) 415-1594 or by e-mail at yks@nrc.gov.

Sincerely,

/RA/

Yoira K. Diaz Sanabria, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvements Programs
Office of Nuclear Reactor Regulation

Docket Nos.: 50-259, 50-260 and 50-296

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATED TO THE SCOPING AND SCREENING AUDIT OF BROWNS FERRY UNITS 1, 2, AND 3
JUNE 7-10, 2004
BROWNS FERRY NUCLEAR PLANT LICENSE RENEWAL APPLICATION

RAI 2.1-1 Safety-Related Definition 10 CFR 54.4(a)(1)(iii)

Section (a)(1)(iii) of 10 CFR 54.4 requires in part, that the applicant consider within the scope of license renewal those systems, structures, and components that ensure the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to those referred to in §50.34(a)(1), §50.67(b)(2), or §100.11.

The staff reviewed Section 2.1.2.1, "10 CFR 54.4(a)(1) - Safety-Related," of the license renewal application (LRA), and determined that a footnote on this page states, "The current licensing basis for BFN Units 1, 2, and 3 is 10 CFR 100.11. A BFN licensing action is being prepared to change the current licensing basis to 10 CFR 50.67." The staff found other definitions for safety related (SR) documented in NEDP-4, Revision 7, "Q-List and UNID Control," which do not refer to offsite exposures comparable to those referred to in §50.34(a)(1) and §50.67(b)(2).

During the audit, BFN personnel stated that they used the Plant Controlled Database Enterprise Maintenance Planning and Control (EMPAC) and Safe Shutdown Analysis (SSA) as source documents to develop the SR structures, systems and components necessary to address the requirements of 10 CFR 54.4(a)(1). The current licensing basis (CLB) for BFN Units 1, 2, and 3 is being changed to include 50.67, and NEDP-4 does not reference §50.67(b)(2). Therefore, the team requested the applicant to define the safety-related (SR) classification definitions that were used in developing the list of SSCs for the license renewal scoping and screening process, and describe how the offsite exposure limitations were factored into the LRA.

RAI 2.1-2 10 CFR 54.4(a)(2) Scoping Criteria for Nonsafety-Related SSCs

By letters dated December 3, 2001, and March 15, 2002, the Nuclear Regulatory Commission (NRC) issued a staff position to the Nuclear Energy Institute which described areas to be considered and options it expects licensees to use to determine what systems, structures, or components (SSCs) meet the 10 CFR 54.4(a)(2) criterion (i.e., all nonsafety-related (NSR) SSCs whose failure could prevent safety-related (SR) SSCs from performing their intended functions identified in paragraphs (a)(1)(i),(ii),(iii) of this section).

The December 3rd letter provided specific examples of operating experience which identified pipe failure events (summarized in NRC Information Notice 2001-09, "Main Feedwater System Degradation in Safety-Related ASME Code Class 2 Piping Inside the Containment of a Pressurized Water Reactor") and the approaches that the NRC considers acceptable to determine the piping systems which should be included in scope based on the §54.4(a)(2) criterion.

The March 15th letter further described the staff's expectations for the evaluation of liquid-filled piping SSCs to determine which additional NSR SSCs are within scope. The position states that applicants should not consider hypothetical failures, but rather should base their evaluation

on the plant's current licensing basis (CLB), engineering judgement and analyses, and relevant operating experience. The letter further describes operating experience as all documented plant-specific and industry-wide experience which can be used to determine the cause of a failure. Operating experience documentation sources would include NRC generic communications and event reports, plant-specific condition reports, industry reports, and engineering evaluations.

Based on the review of the license renewal application (LRA), the applicant's scoping and screening implementation procedures, and discussions with the applicant, the staff determined that additional information is required with respect to certain aspects of the applicant's evaluation of the 10 CFR 54.4(a)(2) criteria. Please address the following issues:

A. LRA Section 2.1.2.2, "10 CFR 54.4(a)(2) - Nonsafety-related SSCs Whose Failure Could Prevent Satisfactory accomplishment of Safety-Related Functions," states "Liquid-filled nonsafety-related SSCs directly connected to safety-related SSCs are in scope for 10 CFR 54.4(a)(1). Nonsafety-related supports in structures that contain safety-related SSCs are in the scope of license renewal per 54.4(a)(2) if they have the ability to prevent the satisfactory accomplishment of a safety-related function. Therefore the need to identify the first seismic anchor beyond any safety related/nonsafety-related interface was eliminated."

1. License renewal boundary Drawing 1-47E801 shows the four main steam lines in red color denoting that it is in scope of the LRA. This (red colored) piping exits the reactor building and becomes black (denoting that it is not in scope) in the turbine building. Describe the criteria used to determine that the integrity of the in-scope piping functions is preserved if a potential age-related degradation failure occurred on the attached NSR piping, given that this NSR piping is not included in the scope, and the piping is not anchored.

2. In the above example, explain how you determined that the SR piping in the reactor building is supported so that it would remain functional if a potential age-related degradation occurred on the NSR piping (in the turbine building) attached to it. This is based on our understanding that the NSR piping and their supports were not considered to be in the scope of the LRA.

3. Describe how the methodology ensured that the nonsafety-related piping up to first equivalent anchor point was included in the scope of the LRA.

B. As described in the March 15th, letter, if the applicant used a mitigative option when performing the scoping of nonsafety-related SSCs under 10 CFR 54.4(a)(2), the applicant should demonstrate that plant mitigative features are adequate to protect SR SSCs from NSR SSC failures, regardless of failure location. If an applicant cannot demonstrate that the mitigative features are adequate to protect SR SSC failures, then the entire NSR SSC is required to be brought into scope of license renewal.

In reviewing the LRA, the NRC staff was unable to determine if the applicant demonstrated that the twelve temperature switches installed in the steam tunnel portion of the turbine building were adequately protected from age-related degradation of NSR SSCs. Based on a review of the "10 CFR 54.4(a)(2) Scoping Methodology, Revision 0," document, the NRC staff was unable to determine how the applicant concluded that the occurrence of "Hot shorts" on the twelve temperature switches in the steam tunnel portion of the turbine building was not credible.

The staff requests the applicant to clarify its position and methodology relative to the consideration of spray and wetting of safety-related SSCs due to the age-related failure of nonsafety-related equipment by providing the following additional information:

1. Identify any moderate/low energy liquid filled piping systems located in the vicinity of the temperature switches.

2. Explain how the twelve temperature switches installed in the main steam tunnel are adequately protected from wetting and spraying resulting from a potential age-related degradation NSR SSC, regardless of the failure location.

3. Describe the methodology used to determine that the occurrence of "Hot shorts" on the temperature switches is not credible.

C. Based on the review of BFN's Procedure, "10 CFR 54.4(a)(2) Scoping Methodology, Revision 0," and discussions with the applicant, the staff determined that additional information is required with respect to certain aspects of the applicant's evaluation of the 10 CFR 54.4(a)(2) criteria. On page 71 of the above mentioned procedure, BFN uses the Intake Pumping Station and Residual Heat Removal Service Water (RHRSW) Tunnel as an example to discuss the approach to "Exposure Duration." The procedure considers a long-term exposure condition resulting from a failed NSR SSC (such as leakage spray) unlikely. The basis for this is that the leakage spray would be quickly identified by personnel walk-downs, sump level trends, by system parameter monitoring alarms, and once identified, appropriate corrective actions would be taken. It is also assumed that water spray from moderate/low energy liquid filled piping could not adversely affect passive components.

Specifically, the staff would like the applicant to provide the basis and justification for the philosophy that passive SR SSCs will not be adversely affected by failure of fluid-filled NSR SSCs in the proximity of those SR SSCs by addressing the following issues:

1. Clarify how you concluded it unlikely that a long-term exposure condition would occur in the RHR SW piping in the intake pumping station and the RHRSW Tunnel resulting from an age-related failure of NSR SSCs.

2. During the scoping and screening process, various effects of water spray must have been considered. Describe the various effects of water spray that you considered from an age-related moderate/low energy liquid-filled NSR piping failure on the SR SSCs installed in the Intake Pumping Structure and the RHRSW Tunnel.

3. Provide a list of NSR SSCs installed in the Intake Pumping Structure and the RHRSW Tunnel that could fail and cause a spray.

4. List the passive SR components such as pipes or manual valves that have been installed in the intake pumping station and the RHRSW Tunnel that could be potentially affected from a leakage spray caused by a failed NSR SSC.

RAI 2.1-3 Quality Assurance Program Attributes in Appendix A, "USAR Supplement," and Appendix B, "Aging Management Activities"

The NRC staff reviewed the applicant's aging management programs described in Appendix A, "Final Safety Analysis Report (USAR) Supplement," and Appendix B, "Aging Management Activities," of the Browns Ferry Nuclear Plant license renewal application. The purpose of this review was to assure that the aging management activities were consistent with the staff's guidance described in NUREG-1800, Section A.2, "Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)," regarding quality assurance attributes of aging management programs.

Based on the staff's evaluation, the quality attributes (corrective action, confirmation process, and administrative controls) described in Appendix B, Section B1.3, "Quality Assurance Program and Administrative Controls," of the LRA for all programs credited for managing aging effects were consistent with Branch Technical Position IQMB-1. However, the applicant has not sufficiently described the AMP quality attributes in Appendix A, "Final Safety Analysis Report Supplement." The staff requests that the applicant supplement the information provided in Appendix A to include a description of the quality assurance program attributes, including references to pertinent implementing guidance as necessary, which are credited for the programs to manage aging effects described in Appendix A and Appendix B of the LRA. The description in Appendix A should provide sufficient information for the staff to determine if the quality attributes for the programs credited with aging management effects are consistent with the review acceptance criteria contained in NUREG-1800, Section A.2, "Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)."

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