September 3, 2004

10 CFR 54

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 LICENSE RENEWAL APPLICATION – NRC SCOPING AND SCREENING AUDIT – REQUEST FOR ADDITIONAL INFORMATION (TAC NOS. MC1704, MC1705, AND MC1706)

By letter dated December 31, 2003, the Tennessee Valley Authority (TVA) submitted an application pursuant to 10 CFR 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, for review by NRC. As part of its review of TVA's license renewal application, the NRC staff conducted a Scoping and Screening Audit the week of June 7-10, 2004. As a result of that audit, the NRC staff identified several areas where additional information is needed to complete their review.

Enclosure 1 to this letter contains the specific NRC request for additional information and the corresponding TVA response. Enclosure 2 contains a list of the specific commitments made in this letter.

If you have any questions regarding this information, please contact Ken Brune, Browns Ferry License Renewal Project Manager, at (423) 751-8421

I declare under penalty of perjury that the forgoing is true and correct. Executed on this third day of September, 2004.

Sincerely,

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT (BFN)

UNITS 1, 2, AND 3

LICENSE RENEWAL APPLICATION (LRA),

NRC SCOPING AND SCREENING AUDIT, JUNE 7-10, 2004

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)

(SEE ATTACHED)

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT (BFN)

UNITS 1, 2, AND 3

LICENSE RENEWAL APPLICATION (LRA),

NRC SCOPING AND SCREENING AUDIT, JUNE 7-10, 2004

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)

By letter dated December 31, 2003, the Tennessee Valley Authority (TVA) submitted an application pursuant to 10 CFR 54, to renew the operating licenses for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 for review by NRC. As part of its review of TVA's license renewal application, the NRC staff conducted a Scoping and Screening Audit the week of June 7-10, 2004. As a result of that audit, the NRC staff identified several areas where additional information is needed to complete their review. Listed below are the specific NRC requests and the corresponding TVA response.

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.

TVA Response to RAI 2.1-1

Consistent with 10 CFR 54.4(a)(1)(iii), BFN utilized a definition of safety-related that incorporated potential offsite exposures as follows: "The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable." The applicable regulation for BFN is 10 CFR 100.11. 10 CFR 50.34 applies to applications for a construction permit and as such is not applicable to BFN. 10 CFR 50.67(b)(2) is applicable to plants revising their current accident source term to Alternative Source Term (AST). TVA has submitted a request for an amendment to the BFN Units 1, 2, and 3 facility operating licenses supporting a full scope application of the AST methodology. The application of AST is not approved by NRC hence, 10 CFR 50.67(b)(2) is not applicable to BFN. The BFN safety-related equipment classification and the SSCs included in the scope of license renewal continue to be based on potential offsite exposures contained in 10 CFR 100. Based on a review of TVA's AST submittal it is expected no new systems or component types will be added within the License Renewal scope that are not already identified in the application.

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 judgment 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 RHRSW 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.

<u>TVA Response to RAI 2.1-2, A.1, and A.2</u>

The seismic class I portions of the four main steam lines have anchors isolating them from the seismic class II piping. The seismic class I/II interface is at the anchor. The piping up to the anchor is designed to seismic class I requirements. The anchor locations are inside the Reactor Building, outboard of the isolation valves. The piping up to the anchor, and the anchor, is included in the scope of license renewal per 10 CFR 54.4(a)(1).

The nonsafety-related piping segments extending from the anchors to the Reactor Building/Turbine Building interface are qualified to seismic class II pressure retention requirements to support secondary containment. Since secondary containment is a safety-related function, these piping segments are in the scope of license renewal and are shown in red on the license renewal drawing. This is consistent with the BFN 10 CFR 54.4(a)(2) Scoping Methodology document which states: "Some non-safety related SSC have been determined to perform safety-related intended functions (e.g. secondary containment, or main steam alternate leak path). These SSCs will be designated 10 CFR 54.4(a)(1) rather than 10 CFR 54.4(a)(2). The SSCs that support a safety-related intended function will also be marked up in red on the license renewal boundary drawings…"

Significant Condition Report (SCR) BFNNEB8601 documented the BFN secondary containment piping penetration configurations did not meet seismic class I requirements. To resolve this SCR, an analysis of the existing secondary containment penetrations was performed to determine the potential for a pipe break on both sides of the secondary containment boundary. Such a break would result in an increase in the air leakage rate into secondary containment. The analysis is documented in "EQE Engineering Report No. 51001.04-R-001 Revision 0, TVA Browns Ferry Nuclear Plant Secondary Containment Piping Penetrations, dated October 12, 1988." The analysis documents that although seismic class I design requirements were not necessarily met, the secondary containment piping does meet seismic class II pressure retention requirements such that there is no credible likelihood of pipe breaks which would result in leakage area increases into the secondary containment volume for systems existing prior to this date. Additional details, related to secondary containment, can be found in UFSAR section 5.3.

The following is taken from NRC letter, dated October 24, 1989, "Supplement 1 to the Safety Evaluation Report on the Browns Ferry Nuclear Performance Plan – NUREG-1232, Volume 3:"

> "As the Browns Ferry FSAR states, the primary purpose of the secondary containment is to limit radioactivity releases after an accident. A DBE resulting in a design-basis accident (DBA) is not considered a credible event because of the seismically designed buildings and structures. Furthermore, the possibility that the two events (a DBE and a DBA) occurring simultaneously is extremely low and, therefore, is not considered credible. However, the licensee has taken a very conservative approach and is proceeding with a program to upgrade its penetration seals in order to conform with its FSAR."

Managing the aging of the main steam piping, components, and supports in the Reactor Building adequately maintains the function of secondary containment. Loss of pressure boundary of the piping in the Turbine Building due to age related failure is bounded by main steam line breaks discussed in Chapter 14 and Appendix M of the UFSAR. The main steam line break outside secondary containment does not prevent satisfactory accomplishment of any function identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

Also, it should be noted that portions of the main steam piping segments located in the turbine building for Unit 1 will be included in the scope of license renewal upon completion of the activities described in Appendix F.1 of the LRA. As documented in Appendix F.1 of the LRA, the Unit 1 current licensing basis for main steam isolation valve leakage does not incorporate an alternate leakage treatment pathway utilizing main steam system piping and the main condenser.

TVA Response to RAI 2.1-2, A.3

The methodology to ensure the liquid filled nonsafety-related piping up to the first equivalent anchor point was included in the scope of license renewal was based on BFN's 10 CFR 54.4(a)(2) Scoping Methodology document which states:

> "Using the building or structure approach, BFN typically decided to include all supports located in a building or structure that contains safety-related equipment into scope per 54.4(a)(2). By including all supports in a building or structure that contains safety-related equipment, the need to identify the first seismic anchor beyond any safety/non-safety interface was eliminated."

> "Apart from the few exceptions identified and discussed within the tables of this document, non safety-related liquid filled piping that is located in a building or structure that contains safety-related equipment is in scope per 54.4(a)(2)."

The underlying premise in this approach is that by including all supports and all liquid filled piping in a building the nonsafety-related piping segments up to the equivalent anchor were captured. Upon further review, it was determined that the possibility exists for supports required for the qualification of the seismic class I piping to be located outside the structure/building (i.e., located in adjacent building). Additionally, it is possible that rather than a support, an embedded piping segment or component was utilized as an equivalent anchor.

BFN did not include nonsafety-related air/gas system piping up to the first equivalent anchor in the scope of license renewal. Similar to air/gas systems, nonsafety-related heating ventilation and air conditioning ductwork segments were not included in the scope of license renewal. The nonsafety-related air/gas and heating, ventilation, and air conditioning (HVAC) SSCs are not in the scope of license renewal per 10 CFR 54.4(a)(2) because industry and site specific operating experience reviews confirmed that nonsafety-related air/gas and HVAC SSCs have not adversely impacted the satisfactory accomplishment of a safety-related intended function. Nonsafety-related structural support components for air/gas and HVAC systems that could prevent satisfactory accomplishment of a safety-related intended function are in the scope of license renewal per 10 CFR 54.4(a)(2).

Based on the above, BFN will review the seismic class I piping boundaries and identify any additional piping segments and supports/equivalent anchors that need to be placed in the scope of license renewal.

TVA Response to RAI 2.1-2, B

TVA will perform a design change notice (DCN) that will make these circuits qualified for wetting and spray from a moderate/low energy line break. Therefore, identification of moderate/low energy liquid filled piping systems located in the vicinity of the temperature switches is not necessary and the occurrence of "Hot Shorts" on the temperature switches is not credible. The DCN will mitigate the consequences of a moderate/low energy line break. This DCN will be implemented prior to the extended period of operation.

TVA Response to RAI 2.1-2, C, RHRSW Tunnel

Upon additional review of piping contained in the RHRSW Tunnel, TVA decided to include the 24-inch Raw Cooling Water discharge piping within the scope of license renewal. The following piping systems, located in the RHRSW Tunnel, were already included in scope: CAD piping, EECW piping, RHRSW piping, and Fire Protection Piping. By including the 24-inch RCW discharge piping in scope along with the other piping systems already in scope, all piping system components which could potentially spray the RHRSW piping are in scope for license renewal. Therefore, it is unlikely that a long-term exposure condition would occur. All supports within the RHRSW Tunnel are already in scope.

TVA Response to RAI 2.1-2, C, Intake Pumping Station

1) Spray due to nonsafety-related system leakage will be detected by plant personnel during activities such as operator rounds and system engineer walkdowns. However, the nonsafety-related components that could spray on RHRSW piping were not excluded from the scope of license renewal based on exposure duration. Spray of water from nonsafety-related components would not introduce a different environment/aging effect since the RHRSW piping is managed for loss of material due to being exposed to an outside air environment. (See item 2 below for additional discussion on this issue.)

2) The effect of water spray from NSR systems at the Intake Pumping Station Structure is bounded by the SR equipment's normal operating environment of outside air. Table 3.0.2 of the LRA includes precipitation as part of outside air. During evaluations, outside air environment considered the effects of periodic wetting and continuous wetting environments. In response to this request, TVA will revise the BFN 10 CFR 54.4(a) Scoping Methodology document to address components located in the lower compartments of the Intake Pumping Station. These areas are subject to submergence during the probable maximum flood. All safety-related passive electrical components installed at the Intake Pumping Station are designed to either be protected from the effects of a wetted environment or designed to perform their function in a wetted environment.

3) The following systems located at the Intake Pumping station contain components that could fail and cause a spray.

   A) The Raw Service Water System (System 25)

   B) The High Pressure Fire Protection System (System 26)

   C) The Condenser Circulating Water System (System 27)

   D) The Potable Water System (System 29)

   E) The Station Drainage System (System 40)

   F) The Raw Water Chemical Treatment System (System 50)

4) The following are the passive SR components in the Intake Pumping Station Structure that could be sprayed: piping, fittings, tubing, valves, pumps, strainers, cable, caulking and sealants, conduits and supports, doors, equipment supports and foundations, electrical panels, racks, cabinets, and other enclosures, supports (ASME equivalent, non-ASME equivalent, and instrument line), mechanical penetrations, electrical and I & C penetrations, reinforced concrete beams, columns, walls, and slabs, compressible joints and seals, cable trays and supports, and tube track.

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

The NRC staff reviewed the applicant's aging management programs described in Appendix A, "Updated Final Safety Analysis Report (UFSAR) 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)."

TVA Response to RAI 2.1-3

The following statement supplements Appendix A.1, "Aging Management Programs" of the BFN License Renewal Application:

> "The integrated plant assessment for license renewal identified new programs, enhancements to existing programs, and existing programs necessary to continue operation of BFN Units 1, 2, and 3 during the additional twenty years beyond the initial license term. This chapter describes those programs. The TVA Nuclear Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B. The TVA Nuclear Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls. These

elements are applicable to all aging management programs credited for license renewal. The corrective action program ensures corrective actions, including root cause determinations and prevention of recurrence are timely. The corrective action program also includes the confirmation process that ensures preventative actions are adequate and that appropriate corrective actions have been completed and are effective. Administrative controls provide for a formal review and approval process of program implementing documents.”

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT (BFN)

UNITS 1, 2, AND 3

LICENSE RENEWAL APPLICATION (LRA),

NRC SCOPING AND SCREENING AUDIT, JUNE 7-10, 2004

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

LIST OF COMMITMENTS

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1. BFN will review the seismic class I piping boundaries and identify any additional piping segments and supports/ equivalent anchors that need to be placed in the scope of license renewal.
2. TVA will perform a design change notice (DCN) for the twelve temperature switches, installed in the steam tunnel portion of the Turbine Building, which will mitigate the consequences of a moderate/low energy line break. This DCN will be implemented prior to the extended period of operation.
3. TVA will include the 24-inch Raw Cooling Water discharge piping located in the RHRSW Tunnel within the scope of license renewal.
4. TVA will revise the BFN 10 CFR 54.4(a)(2) Scoping Methodology document to address the lower compartments of the Intake Pumping Station.