October 26, 2005

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: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) FOR SEVERE ACCIDENT VULNERABILITIES - SUBMITTAL OF BROWNS FERRY NUCLEAR PLANT UNIT 1 SEISMIC AND INTERNAL FIRES IPEEE REPORTS (TAC NO. MC5729)

Dear Mr. Singer:

By letter dated January 14, 2005, Tennessee Valley Authority submitted the Browns Ferry Nuclear Plant (BFN) Unit 1 Seismic IPEEE Report and the BFN Unit 1 IPEEE Fire Induced Vulnerability Evaluation.

Based on our review of your submittal, the U. S. Nuclear Regulatory Commission staff finds that a response to the enclosed request for additional information is needed before we can complete the review. The NRC staff requests a response within 90 days from the date of issuance of this letter.

If you have any questions, please contact me at (301) 415-4041.

Sincerely,

/RA/

Margaret H. Chernoff, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

INDIVIDUAL PLANT EXAMINATION FOR EXTERNAL EVENTS

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

FIRE

1. The submittal provides very little discussion regarding the treatment of hot short cable failures and spurious operation circuit faults. The potential impact of fire-induced spurious actuations on the ability to achieve post-fire safe shutdown is recognized as an important fire risk issue. Describe how hot short cable failures and spurious actuation circuit faults were treated in the original analysis. Please provide such an analysis and discuss the impact of any resulting fire risk scenarios on the conclusions regarding fire vulnerabilities and potential plant improvements.

   It is expected that the Individual Plant Examination for External Events (IPEEE) fire analysis will, as a minimum, include treatment of those hot short and spurious actuation circuit configurations identified as the "Bin 1" items in Regulatory Issue Summary 2004-03.

2. Related to the preceding question, no discussion is provided on the possibility of experiencing a loss-of-coolant accident (LOCA) from a fire event. Such an event may occur from spurious actuation of the automatic depressurization system or other high/low pressure interfaces. Please discuss your analysis of LOCAs caused by the postulated fire scenarios, the frequency of such events and core damage frequency (CDF) associated with them. If such an analysis was not performed, assess the impact of those scenarios on the fire area and compartment CDFs.

3. No discussion is provided regarding the impact of fire on human error probabilities. Generally there are human error related basic events integrated into the probabilistic risk assessment (PRA) model for conditional core damage frequency (CCDP) quantification. These may derive either from the internal events analysis, but should also include fire-specific manual actions as specified in the plant's post-fire safe shutdown procedures. Describe how these manual actions were treated in the post-fire plant safe shutdown response model and in the Human Reliability Analysis portions of the IPEEE fire analysis. Note that all credited human actions should be supported by an assessment of the corresponding human error probability (HEP), including consideration of available staffing, scenario timing, and any conditions associated with the fire, which might contribute to an increase in the HEP value (e.g., smoke, blocked access routes, general confusion, etc.). If the human error probabilities were not adjusted to account for fire impact, either revisit and revise those probabilities according to the conditions posed by the fire scenario and recalculate the CDFs for the affected areas and

compartments, or provide a basis for assuming that fire will not impact each of the credited human actions.

4. Accepted practice for fire PRAs includes the use of a range of heat release rates (HRRs) representing an uncertainty distribution for this parameter. In the Browns Ferry Nuclear Plant (BFN) Unit 1 IPEEE, a single value has been used. For control cabinets, 480Vac Motor Operated Valve (MOV) Boards, 4kV Boards and other electrical panels, a 190 Btu/sec (200kW) peak HRR has been used. This value coincides with the 95th percentile peak HRRs recommended in the Fire Protection Significance Determination Process (SDP) (Inspection Manual Chapter (IMC) 0609, Appendix F) for motor control centers (MCCs) and Switchers. Selection of 95th percentile for peak HRRs is a conservative approach, minimizing the need to evaluate other HRRs from the distribution. However, the submittal has missed two important points of Appendix F:

   a) For control panels, the 95th percentile peak HRR is 650kW.
   b) For MCCs and Switchers, the analysis should include the possibility of high-energy faults.

   Please either provide the basis for not considering 650kW peak HRRs for control panels and not addressing high energy faults in the MOV and 4kV boards or reassess the fire propagation analysis results and re-quantify unscreened fire scenarios using an approach that explicitly treats peak HRRs consistently and includes the impact of high-energy faults. Assess the impact of any analysis changes on the study conclusions regarding fire vulnerabilities and potential plant improvements.

5. In Section 3.3.1, page 15, it is noted that penetrations exist in the slab separating compartments 16-1 and 16-2 that may not be sealed. It is also stated that "while these penetrations present a minimal potential for fire propagation to the Cable Spreading Room, the potential for this fire is, conservatively, being considered." No discussions could be found in the balance of the submittal addressing the noted consideration. Describe the basis for concluding that there are no vulnerabilities associated with the penetrations.

6. In Section 3.3.2, page 17, it is indicated that there is an opening between the Cable Tunnel and the Turbine Building. A discussion is provided about the possibility of fire spreading from the Turbine Building into the Cable Tunnel. The discussion is focused on cable fires and there is no mention of the possibility of turbine oil fire or generator hydrogen fire affecting the Cable Tunnel. Please provide an assessment of the risk contribution of fires involving turbine oil (including splattering of burning oil due to a breach in oil piping) and generator hydrogen fires that might spread into the Cable Tunnel.

7. In Section 6.2.1, page 72, a discussion is provided about the impact of nonqualified cables in the Reactor Building. The CDF associated with these cables is estimated as 1.5E-07, which is based on a CCDP of 3.37E-05. From the discussions provided it can be inferred that only loss of offsite power is postulated to occur as a result of fires involving the nonqualified cables and it appears that all other systems are assumed to remain unaffected by the fire. Provide a list of system components postulated to fail as a result of the nonqualified cable fire scenarios considered. Identify any other potentially

risk-important components that might be impacted by cable failure due to other cables in the same general area as the assumed fire ignition point. Requantify the CDF associated with the nonqualified cable fires if other PRA equipment/component related cables are present within the estimated zone of influence such that potentially risk-important components other than offsite power might be failed.

8. In Section 6.2.1, page 73, a bounding analysis is presented for CDF contribution of nonqualified cables in the Unit 1 Reactor Building. In that analysis, loss of offsite power is assumed as a bounding event for the postulated fire scenarios. However, no discussion is provided for the basis of the CCDP in terms of assumed failed system trains and components by the fire. Please (a) provide a discussion about what PRA components could potentially be affected by nonqualified cable damage in the Unit 1 Reactor Building; and (b) given that information, provide the basis for the CCDP used in the bounding analysis.

9. In Section 6.2.8.1, page 75, it is noted that there are no combustibles located in the corridor area of Fire Compartment 16-1. From this statement it is inferred that transient fires have been either screened out or are assumed to be very unlikely. Furthermore, the corridor is not explicitly addressed in any other parts of the analysis, and therefore no discussion is provided regarding the presence of any cables in this corridor. Identify the cables and components present in the corridor. Provide either the results of the risk quantification or the basis for dismissing the corridor from analysis. If PRA equipment/component related cables are present in the corridor, provide an estimate of the fire-induced CDF associated with corridor fire scenarios that includes transient fires. (The likelihood of transient fires may be quantified using, for example, the Fire Protection SDP procedures (IMC 06.09 Appendix F).)

10. In Section 6.2.8.1, page 77, and related analysis of the Auxiliary Instrument Room in Table 6-2.8.1 (pages 126 through 128), it is concluded that the fire risk contribution is insignificant. No description is provided for fire ignition frequencies (i.e., 8.71E-03, and 7.27E-04) and the CCDPs. Please (a) provide the basis for the ignition frequencies, (b) identify what other PRA components could potentially be affected by a fire in the Auxiliary Instrument Room, and discuss how failure of these components was modeled in the estimation of CDF. If the failure of any mitigating equipment not initially assumed to fail in the analysis is possible, please reassess the risk importance of the Auxiliary Instrument Room.

11. The contribution of transient fires has been dismissed for Fire Areas 4, 5 and 7, the Cable Spreading Room, and Pipe Tunnel based on the assumption that plant control procedures would eliminate the possibility. Please provide a revised CDF estimate for the identified fire areas that includes the impact of transient fuel fires and any other omitted ignition sources (i.e., nonqualified cables and nonqualified junction boxes). (The likelihood of transient fires may be quantified using, for example, the Fire Protection Significance Determination Process procedures (IMC 06.09 Appendix F).)

12. On pages 189 (Fire Area 7), 201 (Fire Area 17), and 203 (Fire Area 19), it is noted that there is no combustible loading associated with non-qualified cables. However, in the lower part of these pages, a frequency is estimated for "cable fire-welding" category ignition source. It seems that there is an inconsistency in the method used to estimate

the frequencies associated with these fire areas. Clarify the discrepancy between the two ignition source categories for these fire areas. If the overall fire frequency increases, or if a discrepancy in the analysis is noted, provide the CDF associated with the new frequencies.

SEISMIC

1. The Submittal does not provide sufficient detail to complete the review. Please provide copies of the following documents that were referenced in the Submittal:

   a) Calculation of Basic Parameters for A-46 and IPEEE Seismic Program, Rev. 0 (Reference 9 in the Submittal).

   b) Browns Ferry Nuclear Plant Unit 1 USI [Unresolved Safety Issue] A-46 Seismic Evaluation Report, Rev. 0, September 2004 (Reference 15 in the Submittal).

   c) Unresolved Safety Issue (USI) A-46/Seismic IPEEE Relay Evaluation Browns Ferry Nuclear Plant Unit 1, Rev. 0, January 2004 (Reference 16 in the Submittal).

   d) Seismic-Induced II/I Spray Evaluations at Browns Ferry Nuclear Plant Unit 1, Rev. 0, March 2004 (Reference 19 in the Submittal).

2. Please provide a graph of the Review Level Earthquake (RLE) spectra used for the seismic margin assessment (SMA). On the same graph also provide the site Design Basis Earthquake (DBE) ground spectra (Housner spectra with 0.2g peak ground acceleration), and the USI A-46 spectra.

   Discuss whether there is any exceedance of DBE or USI A-46 spectra over the RLE spectra in the frequency range of interest for the BFN Unit 1 systems, structures, and components (SSCs).

3. Please describe the scope and the kind of seismic spectra used for the seismic review of the BFN Unit 1 Restart Project. How is the Restart Project seismic review coordinated with the seismic IPEEE/USI A-46 review?

4. BFN Unit 1 has been out of service since March 1985. Considering that safety-related SSCs in BFN Unit1 have been idle for 20 years, how does Tennessee Valley Authority (TVA) ensure that these SSCs will be in working order and will perform their designed safety functions properly, especially under the seismic DBE conditions? Preoperational tests (if to be performed) and limited IPEEE seismic walkdown performed may not uncover all the potential seismic problems due to age-related degradation of SSCs. Are all these addressed in the BFN Unit 1 Restart Project?

5. Questions on Section 3 of the Submittal - System Description and Success Path Selection:

   a) Section 3 of the submitted report states that "The success path selection and identification of components for the BFN1 seismic IPEEE program were based on the previous BFN2 and BFN3 seismic IPEEE programs." It further stated that "Success path logic diagrams (SPLDs) were constructed for the BFN2/3 seismic IPEEE . . . ." and "They (SPLDs) were used as a basis for the identification of the equipment to be included on the BFN2/3 seismic safe shutdown equipment list (SSELs)." This description is rather confusing.

For the current BFN Unit 1 seismic IPEEE, did TVA prepare separate SPLDs and SSELs for BFN Unit 1 apart from those for BFN Unit 2 or BFN Unit 3? BFN Unit 1 should have its own set of SSELs that are basically different from those for BFN Units 2 and 3. If there is common equipment that appears on BFN Unit 1 and BFN Units 2 and 3 SSELs, please identify.

b) Section 3 of the Submittal provides two lists, one for the relevant plant functions and one for the front line systems to accomplish those functions, but there is no description regarding which function is accomplished by what front line system(s).

Please confirm whether the following function/system match-up for BFN Units 2 and 3 also applies for BFN Unit 1:

"The frontline systems selected to achieve the four shutdown functions are: a) control rod drive system (CRD) for reactivity control, b) safety/relief valves (SRVs) for reactor pressure control, c) core spray (CS) and low pressure coolant injection (LPCI) mode of residual heat removal system (RHR) (with reactor pressure vessel depressurization using SRVs) for reactor coolant inventory control, and d) suppression pool cooling mode of RHR for decay heat removal."

6. Section 3.2.5.8, "Nonseismic Failures and Human Actions," of NUREG-1407 stated that for Electric Power Research Institute SMA, "Success paths are chosen based on a screening criterion applied to nonseismic failures and needed human actions. It is important that the failure modes and human actions are clearly identified and have low enough probabilities to not affect the seismic margins evaluation."

Please provide information as to how this was considered in choosing the success paths and the associated equipment for BFN Unit 1 SMA.

7. Section 7 and Appendix C of NUREG-1407 states that a peer review should be conducted by individuals who are not associated with the initial evaluation, to ensure the accuracy of the documentation and to validate both the IPEEE process and its results. The Submittal has no mention of any peer review performed.

Please provide the following information for the seismic IPEEE: (1) composition of peer review team, (2) areas of peer review and major comments, and (3) resolution of comments.

8. Please provide copies of the following high confidence of low probability of failure (HCLPF) calculation for equipment not screened out:

a) MCC (ID No: 1-BDBB-281-0001A)

b) RHR Heat Exchanger (ID No: 1-HEX-74-900A)

Flat bottom tanks were identified in many previous seismic IPEEE reviews as components with potential low HCLPFs, but the Submittal has no discussion on this. Was the condensate storage tank included in the SSEL? If the condensate storage tank was not included, please explain why.

9. Chapter 6 and Table C.1, Item 3.2 of NUREG-1407 requires that coordination with ongoing programs and other seismic issues (such as USI A-17, 40, 45, eastern U.S. seismicity issue and other seismic safety issues such as Generic Safety Issue (GSI)-156, "Systematic Evaluation Program," GSI-172, "Multiple System Response Program (MSRP)") be described in the IPEEE submittal. Other than seismic induced fire/flood evaluation and USI A-46, the Submittal did not provide information on the coordination with ongoing programs and other seismic issues.

   Please provide the missing information.

10. The evaluation of seismic induced flooding in the Submittal does not discuss the failure potential of dams upstream of BFN Unit 1 and its consequences to BFN Unit 1. Generic Letter 88-20, Supplement 5, does not exclude review of dams, levees or dikes and consequences of their failures.

    Please provide this information.

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