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

Dear Mr. Singer:

By letter dated January 14, 2005, Tennessee Valley Authority (TVA, the licensee) submitted the Browns Ferry Nuclear Plant (BFN), Unit 1, Seismic Individual Plant Examination of External Events (IPEEE) Report and the BFN Unit 1 IPEEE Fire Induced Vulnerability Evaluation. By letter dated October 26, 2005, the U.S. Nuclear Regulatory Commission (NRC) staff requested additional information to support the review. By letter dated February 2, 2006, TVA responded to the staff's request.

Based on our review of your submittal, as supplemented, the NRC 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 60 days from the date of issuance of this letter.

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

Sincerely,

/RA by D. Duvigneaud for/

Margaret H. Chernoff, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
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 licensee has provided an explanation of how spurious operations as a result of fire damage are considered in the fire scenarios. The following examples are provided. Cable damage within 480-V reactor motor-operated valve boards results in closure of main steam isolation valves (MSIVs). Plant transients such as "turbine trips" have been conservatively assumed for a majority of the fire areas. These examples are not the type of spurious actuation typically considered in fire probabilistic risk analyses (PRA) because MSIV closure and turbine trip are the fail safe positions of the affected devices. Spurious operation is generally defined as a circuit fault mode (e.g., control circuit of a motor-operated valve (MOV) or a pump) wherein an operational mode of the circuit is initiated due to failures in one or more components of the circuit. For example, if the closed position of a MOV is considered as its safe position for a sequence of events, a circuit fault (caused by cable damage exposed to fire) leading to spurious opening of the MOV would be considered as a spurious operation of that MOV. Generally, in fire PRA, spurious operations that may cause a plant initiator or aggravate a chain of events are specifically identified and incorporated into the analysis.

   The effect of hot shorts and spurious actuation circuit faults has not been incorporated in the analysis. The previous request for additional information (RAI) stated that it is expected that the Individual Plant Examination of External Events fire analysis will, as a minimum, include the treatment of those hot short and spurious actuation circuit configurations identified as the "Bin 1" items in Regulatory Issue Summary No. 2004-03; however, it is not apparent whether the licensee has addressed this issue of hot shorts and spurious operation. Since the Nuclear Regulatory Commission staff recognizes the importance of addressing hot shorts and spurious operation as a result of fire damage, the staff has explicitly included hot shorts and spurious operation in Fire Protection Significance Deterministic Process, Inspection Manual Chapter No. 0609 Appendix F.

   Please provide an analysis of the effects of hot shorts and spurious actuation circuit faults and discuss the impact of any resulting fire risk scenarios on the conclusions regarding fire vulnerabilities and potential plant improvements.

2. From the licensee’s response, it is inferred that the human error probabilities (HEPs) have been set to 1.0 only for operator actions related to equipment directly affected by the fire. Otherwise, the HEPs from the internal events analysis are used without any

adjustment to account for the indirect effects of the fire event. HEPs associated with actions within the control room were assumed to be unaffected by the fire events. This assumption is valid for scenarios that do not affect instrumentation cables. Since instrumentation cables are not modeled in a typical fire PRA, it is recommended to use conservative HEP values for control room operator actions. The practice of using internal events HEPs for control room actions could be optimistic.

HEPs associated with operator actions outside the control room is treated in a similar manner. A HEP is assumed to be 1.0 for actions involving equipment affected by the fire. Otherwise, the internal events HEP is used. An upward adjustment to the internal events HEP may be needed based on the specifics of the postulated fire scenario. For example, the possibility of a fire affecting the plant operators' ability to reach areas of the plant where actions may be needed has not been included in the analysis. Similar to the control room operator actions, since the HEPs are not individually reviewed and adjusted for the effects of fire, the probability values of the affected scenarios could be optimistic.

As noted in the original RAI, please review all HEPs not already adjusted to account for fire impact, and, for each HEP, either: (a) revise the HEP to address the conditions posed by the fire scenario, or (b) provide a basis for assuming that fire will have a negligible impact on the human actions associated with the HEP.

Following revision of the HEPs, please assess the impact of any analysis changes on the study's conclusions regarding fire vulnerabilities and potential plant improvements.

3. The licensee states that the FIVE methodology was used for Unit 1, consistent with the approach used for Units 2 and 3 analysis. In the analysis provided in response to the RAI, the licensee does not explicitly address the 650-kW fire noted specifically in the RAI. Instead, the response increases the conditional core damage probability (CCDP) by two orders of magnitude to account for the effects of a 650-kW fire. Depending on cable routing characteristics of the plant, it is possible that loss of the cables present in the zone of influence could lead to a CCDP much greater than that postulated by the analysts.

   Also, in the licensee's response to the RAI, the probability of nonsuppression is assumed to be 0.1, but proper supporting information is not provided. It is important to note that the nonsuppression probability can only be used when it can be clearly concluded that the fire can be controlled before the postulated damage. This assumption may be valid for a cabinet fire leading to cable damage above the cabinet (thus providing sufficient time for fire brigade or other fire protection system response). However, the assumption is not valid for high-energy faults where damage occurs in a very short time.

   Please provide:

   1) The bases for increasing the CCDP by two orders of magnitude, considering the additional cables and equipment that may be affected by a 650-kW fire and high energy faults.

2) The bases for using the postulated nonsuppression probabilities given the timing of the events allowing timely actuation of the detection and automatic suppression systems and fire brigade response.

Please assess the impact of any analysis changes on the study's conclusions regarding fire vulnerabilities and potential plant improvements.

4. The licensee's response states that the list of affected components is not known and presents a bounding analysis. The CCDP used in the bounding analysis is much less than 1.0, which means that the response assumes that key equipment survives the fire (i.e., one or two systems are independent of the postulated fire scenario). This approach cannot be regarded as appropriate because it ignores the fact that there could be a case that the affected nonqualified cables may lead to much greater increase in the CCDP than what has been postulated. Given the lack of information on the location of key (nonqualified) cables in the Reactor Building, it is possible that key cables could be in the areas of interest and could be damaged by fire, and that the one order of magnitude increase in CCDP assumed in the licensee's response could be optimistic.

   Please, provide the basis for increasing the CCDP by one order of magnitude, considering the additional cables and equipment that may be affected in each of the zones. Please assess the impact of any analysis changes on the study's conclusions regarding fire vulnerabilities and potential plant improvements.

BROWNS FERRY NUCLEAR PLANT

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