March 7, 2006

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, UNITS 1, 2, AND 3 — REQUEST FOR ADDITIONAL INFORMATION REGARDING THE EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST (TS-418 AND TS-431) (TAC NOS. MC3812, MC3743, AND MC3744)

Dear Mr. Singer:

By letters dated June 25, 2004, and June 28, 2004, as supplemented by letters dated August 23, 2004, February 23, April 25, and June 6, 2005, the Tennessee Valley Authority (the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) amendment requests for Browns Ferry Nuclear Plant, Units 1, 2, and 3. The proposed amendments would change the operating license to increase the maximum authorized power level to 3952 megawatts thermal. These changes represent an increase of approximately 20-percent above the current maximum authorized power level for Unit 1, and approximately 15-percent above the current maximum authorized power level for Units 2 and 3. 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 your staff on March 1, 2006, and it was agreed that a response would be provided by March 31, 2006. If you have any questions, please contact me at 301-415-4041.

Sincerely,

/RA/

Margaret H. Chernoff, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosure:
Request for Additional Information

cc w/enclosure: See next page

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SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 — REQUEST FOR ADDITIONAL INFORMATION REGARDING THE EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST (TS-418 AND TS-431) (TAC NOS. MC3812, MC3743, AND MC3744)

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REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1. In its review of the Tennessee Valley Authority (TVA) responses to the U.S. Nuclear Regulatory Commission (NRC) requests for additional information (RAIs), the NRC staff noted some inconsistencies in the information provided. For example, the response to RAI SPSB-A.7 (References 1 and 2) for all three units is not complete, based on a comparison with the response to RAI SPSB-A.20 (Unit 1), and SPSB-A.22 (Units 2 and 3).

   Table SPSB-A.7-1 for Unit 1 omitted event BE_HOAL2, which is the operator action with the second-highest Fussell-Vesely importance in the response to RAI SPSB-A.20.

   Table SPSB-A.7-1 for Unit 2 has different Fussell-Vesely values for operator actions that correspond to those in the Unit 2 response to RAI SPSB-A.22. Table SPSB-A.7-1 includes event HRA_OBD_1, which is not in the SPSB-A.22 table, and is missing the following events that are important in the SPSB-A.22 table:

   - U1FALLHUMAN
   - OHL2
   - U1FHXHUMAN
   - OHC3
   - BEIVR10

   Table SPSB-A.7-1 for Unit 3 has different Fussell-Vesely values for operator actions that correspond to those in the Unit 3 response to RAI SPSB-A.22. Table SPSB-A.7-1 includes events BEHORVD2 and BEHORVD3, which are not in the SPSB-A.22 table, and is missing the following events that are important in the SPSB-A.22 table:

   - HER_HPRVD1
   - OHL2
   - BEIVR10

   Provide the following information for all three units:

   a. Corrected and complete responses to RAI SPSB-A.7.

   b. Corrected and complete responses to RAI SPSB-A.20 (Unit 1) and SPSB-A.22 (Units 2 and 3).

Enclosure

c. A description of the quality control measures that were used to ensure completeness and accuracy of the risk information provided in the license amendment request and in the responses to the RAIs that have been submitted to date, and that were used in answering these RAIs.

1. For all three units, answer the following questions for low power/shutdown operations, as they relate to the extended power uprate (EPU). These questions, taken from the Standard Review Plan, Chapter 19, Table III-1, provide an acceptable way to assess risks from low power/shutdown operations. The responses previously provided to RAIs SPSB-A.18 (Reference 1), and SPSB-A.17 (Reference 2), did not answer the staff's question.

   a. Does the proposed EPU introduce new initiating events or change the frequencies of existing events? Explain why or why not.

   b. Does the proposed EPU affect the scheduling of outage activities? Explain why or why not.

   c. Does the proposed EPU affect the ability of the operator to respond to shutdown events? Explain why or why not.

   d. Does the proposed EPU affect the reliability or availability of equipment used for shutdown conditions? Explain why or why not.

   e. Does the proposed EPU affect the availability of equipment or instrumentation used for contingency plans? Explain why or why not.

2. For all three units, provide the following information regarding the probabilistic risk analysis (PRA) success criteria for the safety/relief valves (SRVs):

   a. How many SRVs are required to open to mitigate an anticipated transient without scram (ATWS) event at EPU conditions? For Units 2 and 3, what was the pre-EPU number of SRVs required?

   b. How many SRVs are assumed to open on a transient event with pressure challenge under EPU conditions? For Units 2 and 3, what was this assumption pre-EPU?

   c. How many SRVs are assumed to need to close following a transient event with pressure challenge under EPU conditions? For Units 2 and 3, what was this number pre-EPU?

   d. What is the technical basis (e.g., Modular Accident Analysis Package calculations) for the success criteria given in answer to questions 2.a, 2.b and 2.c above?

   e. Does the model consider failures of different numbers of SRVs to close as different sequences (i.e., do they transfer to different loss-of-coolant accident event trees)? Describe in detail.

f. How is common cause failure of multiple SRVs to open handled in the PRA models?

g. How is common cause failure of multiple SRVs to close after opening handled in the PRA models?

3. During the PRA audit conducted the week of January 23, 2006, the audit team noted disagreement among the PRA model, the model documentation, and the explanations by the TVA PRA personnel regarding credit for control rod drive (CRD) injection in the post-EPU models. The staff also notes conflicting information in the RAI responses: The response to RAI SPSB-A.6.c (References 1 and 2) has a table that indicates “each unit models CRD injection.” For Unit 1, RAI SPSB-A.21 says there is no credit for CRD. For Units 2 and 3, RAI SPSB-A.20 states that CRD is no longer viable; but SPSB-A.23 states that CRD is credited for some sequences.

   For all three units, post-EPU: Provide details on whether CRD credited as an injection source. Describe the sequences for which CRD is credited. What is the basis for allowing credit? Provide the success criteria and timing for sequences where CRD is credited.

4. During the PRA audit, the audit team was told that core damage frequency (CDF) is quantified by first solving all the way through large early release frequency (LERF) and then “backing out” the CDF number. The team was also told that this practice results in some understating of the CDF number (i.e., a higher CDF would be calculated if the PRA model was solved for CDF directly). For all three units, provide a sensitivity that shows how much CDF is “lost” due to the practice of quantifying all the way through LERF and then determining the resulting CDF. What types of CDF sequences are truncated because of this practice (i.e., is the “missing” CDF spread evenly across the plant risk profile, or are there sequence types that are preferentially truncated)? Discuss the impact of this practice on the risk results provided for EPU.

5. For all three units, provide an analysis of the sensitivity of core damage risk to an assumed increase in the initiating event frequency for turbine trip and for loss of feedwater to aid in understanding the uncertainties of these frequencies given modifications to the turbine electro-hydraulic control software and to the feedwater pumps and controllers. The staff would suggest doubling the existing frequencies for these sensitivity analyses, unless justification is provided for a different approach.

6. For all three units, for important operator actions (i.e., Fussell-Vesely > .005 or Risk Achievement Worth > 2) that are time critical, justify that these actions can be completed within the time frame from receipt of the cue for the action to the point at which an irreversible plant state leading to core damage is reached under EPU conditions. (For this question, assume “time critical” means the action must be completed within 3 hours of the start of an initiating event or within 1 hour of receipt of the cue.) Provide the basis for the conclusion that the time available is sufficient to complete the action (e.g., information from simulator observations, job performance measures, walk-through, talk-through, etc.).

7. During the PRA audit, the audit team noted that the Unit 1 event tree for ATWS has both an “OAL” and an “OTAF” top event, representing lowering level and maintaining level at the top of active fuel, respectively. However, there is no calculation in the Human Reliability Analysis (HRA) notebook for the OAL event, but there is a calculation for OTAF. The definition of OTAF appears to include both lowering level and controlling level at the top of active fuel. The PRA staff indicated that the intent is to only have one event, representing OTAF, in the event tree model. However, in the response to RAI SPSB-A.20 (Reference 1), the event HOAL2 (representing OAL) is one of the events with the highest Fussell-Vesely importance measure, indicating that it exists in the PRA model and has an associated human error probability.

   Describe how events OAL and OTAF are modeled in the Unit 1 ATWS event tree. Provide the HRA for these events. Explain how dependency between these events is addressed.

8. For Units 2 and 3, provide an assessment of the increase in risk if only EPU is considered. For example, the effect of changing the high pressure coolant injection/reactor core isolation cooling common cause treatment was to lower risk, thereby offsetting part of the increase in risk that resulted from actual plant physical changes or reduced operator action timing. The staff notes that changing the common cause treatment does not represent a real change in risk. Changes in methodology, model enhancements, or correction of errors should be represented in both the base case model and the post-EPU model (or in neither) in order to obtain a representation of change in risk that is not masked by these nonphysical factors. Both the change in CDF and the change in LERF should be provided for both units.

9. RAI SPSB-A.12 asked about the impact of increasing the ultimate heat sink temperature from 91 to 95 degrees on Units 2 and 3. However, the response (Reference 2) only addressed Unit 1. Please address this for Units 2 and 3 as originally requested. Identify the PRA basic events affected by this change.

<u>References</u>

1. Letter from Brian O'Grady, TVA, to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 – Response to NRC Round 2 Requests for Additional Information Related to Technical Specifications (TS) Change No. TS-431 – Request for Extended Power Uprate Operation (TAC No. MC3812)," December 19, 2005, TVA-BFN-TS-431 (ADAMS Accession Number ML053560194)

2. Letter from Brian O'Grady, TVA, to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 – Response to NRC Round 2 Request for Additional Information Related to Technical Specifications (TS) Change No. TS-418 – Request for Extended Power Uprate Operation (TAC Nos. MC3743 and MC3744)," December 19, 2005, TVA-BFN-TS-418 (ADAMS Accession Number ML053560186)

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