September 27, 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 FOR EXTENDED POWER UPRATE - ROUND 10 (TS-431 AND TS-418) (TAC NO. MC3812, MC3743 AND MC3744)

Dear Mr. Singer:

By letters dated June 28 and 25, 2004, the Tennessee Valley Authority (TVA, the licensee) submitted amendment requests for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, as supplemented by letters dated August 23, 2004, February 23, April 25, June 6, and December 19, 2005, February 1 and 28, March 7, 9, 23, and 31, April 13, May 5 and 11, June 12, 15, 23 and 27, and July 6, 21, 24, 26, and 31, and August 4, 16, and 18, and September 1 and 15, 2006.

The proposed amendments would change the BFN operating licenses to increase the maximum authorized power level by approximately 20 percent above the current maximum authorized power level for Unit 1, and approximately 15 percent for Units 2 and 3. The proposed amendments would also change the Unit 1 licensing bases and associated Technical Specifications to credit 3 pounds per square inch gauge (psig) for containment accident pressure following a loss-of-coolant accident and increase the reactor steam dome pressure by 30 psig. The proposed amendments would also change the Units 2 and 3 licensing bases to revise the credit for containment accident pressure from 3 pounds for short-term and 1 pound for long-term, to 3 pounds for the duration of a loss-of-coolant accident, and revise the maximum ultimate heat sink temperature.

A response to the enclosed Request for Additional Information is needed before the Nuclear Regulatory Commission staff can complete the review. These requests were provided in draft form to your staff by e-mail September 22, 2006. A response is requested by October 4, 2006.

If you have any questions, please contact me at (301) 415-2315.

Sincerely,

/RA/

Eva A. Brown, 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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NAME	LRegner	EBrown	MChernoff	BClayton	KManoly by memo
DATE	9/27/06	9/27/06	9/27/06	9/27/07	8/17/06

APLA/BC	DSS/SCVB	DSS/SBWB	LPL2-2/BC
MRubin by memo	TMartin by memo	TMartin by memo	LRaghavan
8/31/06	8/22/06	9/14/06	9/27/06

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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, 50-296

EEMB

115/85. The Nuclear Regulatory Commission (NRC) staff requested a discussion of any weld reinforcement following fatigue cracking of drain channel in the BFN steam dryers. In its response to this request (identified as EEMB.C.1 on page E1-106 of the July 26 submittal), the Tennessee Valley Authority (TVA, the licensee) states it has periodically inspected the Units 2 and 3 repaired drain channel welds subsequent to 105 percent original licensed thermal power (OLTP) operation.

a. Identify the inspection technique used for Units 2 and 3 and explain whether that technique was qualified to detect fatigue cracks.

b. Specify whether these periodic inspections will be performed subsequent to 105 percent OLTP operation for Unit 1. If so, identify the inspection technique to be used and explain whether that technique is qualified to detect fatigue cracks.

116/86. The NRC staff requested a discussion of the post-modification inspection procedures for the BFN steam dryer modifications. In its response to this request (identified as EEMB.C.19 on pages E1-125 and 126 of the July 26 submittal), TVA stated that the post-modification inspection will be conducted employing visual inspection (VT-2). Discuss the adequacy of this inspection method, and the ability to conduct a more detailed inspection of the BFN Unit 1 steam dryer.

117/87. Section 9.9 of Rev. 2 of the steam dryer stress report states that TVA plans to use pressure transducers mounted in holes in the main steam lines (MSLs) to measure fluctuating pressures as input to the acoustic circuit model (ACM). Provide a schematic of the proposed installation, which shows clearly the location of the pressure transducer with respect to the inner surface of the MSL walls. Since pressure transducers exposed to steam flow will measure acoustic pressure and turbulence traveling through the MSLs and over the pressure transducer, quantify any bias error or uncertainty that might be introduced to the dryer leads computed with the Bounding Pressure ACM by the presence of turbulence-induced pressures in the ACM inputs.

118. In the July 26, 2006, response, the licensee indicated that Unit 1 is currently performing restart modifications and that the final stress analysis results, which reflect the as-built configuration, are not available for most of the reactor coolant pressure boundary and balance-of-plant systems. Provide the schedule for completion of the piping system evaluation for Unit 1. Upon completion, provide the evaluation summary for piping systems and their supports including main steam, feedwater, recirculation, residual heat removal, and torus-attached piping systems. The information should include the calculated maximum stresses and fatigue usage factors, as necessary, for piping systems and their supports similar to those provided for the Units 2 and 3 extended power uprate (EPU) evaluation.

<u>APLA</u>

27/29. For this request, an operator action is “important to risk” if any one of the following criteria is met: (1) Fussell Vesely (FV) importance to core damage frequency (CDF) greater than 0.005; (2) FV importance to large early release frequency (LERF) greater than 0.005; (3) risk achievement worth (RAW) importance to CDF greater than two; or (4) RAW importance to LERF greater than two. Provide the following information for operator actions modeled in the probabilistic risk assessment that are important to risk:

a. Basic event (operator action) name

b. Description

c. Where action is performed (e.g., control room, outside control room, both)

d. For the pre-EPU model;

   i. FV importance to CDF
   ii. RAW importance to CDF
   iii. FV importance to LERF
   iv. RAW importance to LERF
   v. time available to the operator from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release
   vi. Human error probability

e. For the post-EPU model:

   i. FV importance to CDF
   ii. RAW importance to CDF
   iii. FV importance to LERF
   iv. RAW importance to LERF
   v. Time available to the operator from receipt of the appropriate cue until the action must be complete for successful mitigation of core damage or large early release

vi. Human error probability

ACVB

68/66. The staff has determined that the information in the May 24, 1976, report may not be sufficient to justify credit for a value of required net positive suction head (NPSH) less than the 3 percent head loss value.

a. Provide any supporting information not included in the May 24, 1976, report that supports the use of a lower value such as:

i. accelerometer data,
ii. time that the Residual Heat Removal (RHR) pump was in cavitation, and
iii. the inspections performed on the pump before and after testing.

b. Describe the operational history of RHR pump 3A. Address whether pump RHR 3A experienced any abnormal operation since this testing.

69/67. Describe the peak short-term loss-of-coolant accident (LOCA) suppression pool temperature at 105 percent power. Provide the service water temperature assumed in this analysis.

70/68. Verify that at 105 percent power, for the short-term LOCA, the available NPSH is always greater than the required NPSH at the peak RHR pump flow (11,500 gpm) without reliance on the testing reported in the May 24, 1976, report.

SBWB

50/75. Provide the sequence of events tables for the limiting Appendix K Large Break LOCA and the limiting Appendix K Small Break LOCA (0.06 ft^2) discharge break with a battery failure and only five automatic depressurization system (ADS) valves actuated. The staff also requests the licensee to provide the low pressure core spray and low pressure coolant injection head versus flow curve, limiting axial power shape, and ADS relief valve set pressure and relief capacity used in the analysis.

Mr. Karl W. Singer
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

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