August 10, 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, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE - ROUND 8 (TS-431) (TAC NO. MC3812)

Dear Mr. Singer:

By letter dated June 28, 2004, 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, 11, 15, and 16, and June 2, 2006, the Tennessee Valley Authority submitted to the U.S. Nuclear Regulatory Commission (NRC) an amendment request for Browns Ferry Nuclear Plant, Unit 1. The proposed amendment would change the Unit 1 operating license to increase the maximum authorized power level from 3293 to 3952 megawatts thermal. This change represents an increase of approximately 20 percent above the current maximum authorized power level for Unit 1. The proposed amendment would also change the Unit 1 licensing bases and associated Technical Specifications to credit 3 pounds per square inch gauge (psig) for containment overpressure following a loss-of-coolant accident and increase the reactor steam dome pressure by 30 psig.

A response to the enclosed request for additional information is needed before the Nuclear Regulatory Commission (NRC) staff can complete the review. The steam dryer questions (EEMB) in this request were provided on July 12, 2006, while the remaining questions (SBWB) were provided July 13, 18 and 20, 2006. The July 20, 2006 questions are in support of an audit planned for August 8, 2006 at the site. These requests were discussed with your staff on August 2, 2006, and it was agreed that a response would be provided by August 18, 2006.

As stated in a letter dated August 8, 2006, some of the steam dryer questions contain information from Continuum Dynamics Incorporated (CDI) Report No. 05-28P, Bounding Methodology to Predict Full Scale Steam Dryer Loads from In-Plant Measurements, Revision 1 (05-28P) which was requested withheld from public disclosure pursuant to Title 10 of the Code of Federal Regulations (10 CFR), Section 2.390. However, information needed to complete the NRC staff's withholding review for this information has not been provided. Therefore, the NRC staff will release the information sought to be withheld after 30 days of this letter unless the information is withdrawn or amended consistent with the requirements of 10 CFR 2.390(b).

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

Sincerely,

/RA by EBrown 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

Enclosures: 1. Redacted Request for Additional Information
2. Proprietary Request for Additional Information

cc w/enclosure 1 only: See next page

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OFFICE	LPL2-2/PM	LPL2-2/PM	LPL2-2/LA	SBWB/BC
NAME	EBrown	EBrown for MChernoff	CGoldstein	GCranston by memo
DATE	8/9/06	8/ 9 /06	8/ 9 /06	7/26/06

OFFICE	EEMB/BC	SCVB/BC	LPL2-2/BC
NAME	KManoly by memo	RDennig by memo	LRaghavan
DATE	7/27/06	7/26/2006	08/10 /06

OFFICIAL RECORD COPY

REDACTED REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

EEMB

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59. Discuss which break flow model (Moody, HEM) is used for containment peak pressure at uprated conditions.

60. Address whether the only difference between the values of updated final safety analysis report peak drywell pressure of 50.6 pounds per square inch gage (psig) and the current method of 47.7 psig is due to the difference in LAMB models discussed in note (3) of Table 4-1 of the PUSAR. Discuss whether the method of inputting LAMB output for the current method at uprated conditions is the same as that used for the Vermont Yankee extended power uprate (EPU).

61. a. Provide a technical manual or a description of the MULTIFLOW program for calculating pressure losses in piping systems which is used for the EPU net positive suction head (NPSH) calculations.

b. Discuss what, if any, conservatism is included in the MULTIFLOW calculations.

c. Enclosure 6 of the March 23, 2006 submittal contains MD-Q0999-970046, NPSH Evaluation of Browns Ferry residual heat removal and core spray Pumps. Page 12, states that a piping roughness value of 1.5E-4 ft was selected, which corresponds to condensate quality water. Justify why this is acceptable for suppression pool water, and address whether this a significant assumption.

SBWB

32. In the SRLR (Supplemental Core Reload Report) dated May 15, 2006, different initial minimum critical power ratio (MCPR) values are given for different application conditions. However, for pressurization transients, the operating MCPR for normal operation with all the equipment operating is not given. Provide the operating limit MCPR with all equipment in operation. Address which transient is the limiting transient in determining the operating MCPR. Provide a table similar to the Table for Non-pressurization transients in Section 11 on page 38.

33. Pages 23 to 36 of the SRLR gives the uncorrected delta CPR (critical power ratio) for various events. Address why they are uncorrected. Discuss the purpose for no correction of the associated events.

34. In response to SBWB-25, which was transmitted in a letter dated March 7, 2006, TVA stated on page E1-136 that turbine trip with bypass failure will be analyzed for the first Unit 1 EPU core design (Cycle 7). The NRC staff has reviewed the SRLR and notes that it does not appear to include the turbine trip with bypass failure analysis. Address whether the analysis was reperformed as indicated and discuss why the analysis is not contained in the Cycle 7 Unit 1 SRLR.

35. Based on the Unit 1 EPU core design, demonstrate that the impact of the bypass voiding on the reliability and accuracy of the instability protection capability (e.g., detect and suppress capability (Option III), safety limit MCPR protection and the armed region calculation). Use limiting core conditions in the calculations such as the in-channel voids assumed at different elevations and the codes approved for Option III stability solution.

36. Figure 2-2 of NEDC-33173P shows a plot of the typical void-quality relation at high power/flow ratio. Evaluate the database supporting the void fraction correlation and plot the supporting validation measurement data on Figure 2-2. Identify the type of validation data on the plot. Provide a table summarizing the test conditions (e.g., pressure, mass flux), the type of the validation data (e.g., 4x4 bundle with part-length, multi-rod) and the applicability range. As is the norm, exclude the data used to develop the correlation.

37. Unit 1 is an initial core, with potential control rod blade replacement or the use of blades stored for long durations.

a. Address whether all or part of the control rods be replaced with new ones. Provide a control rod (CR) replacement plan. If CR blades in the spent fuel storage and being used, explain how the calculated CR worth is verified or assessed. Explain how it is confirmed that the CR worths used in the safety analyses (e.g., control rod drop analyses [CRDA], the scram worth) are consistent with the worth of the actual CR blades at the plant.

b. Assuming that the Unit 1 Technical Specification (TS) will be identical to Units 2 and 3, TS Section 3.1.1, SHUTDOWN MARGIN (SDM), states that SDM shall be within the limits provided in the core operating limit report (COLR). It appears that although the SDM requirement and the corresponding value do not change on cycle-specific bases, the SDM value has been relocated from the TS to the COLR. The Cycle 7 Ksro at the most reactive state (Ksro +R) is 0.984 ΔK/K and the all- rods-in Keff is 0.945 ΔK/K. This results in a one-rod-out control rod worth of 3.8 % ΔK/K. Address whether this value is consistent with the assumptions made in the generic CRDA analyses.

c. Unit 1 has been out of operation for over 20 years and therefore, no trend line exists to define the analytical methods [ ] calculations for Unit 1, and provide the basis for the bias applied.

d. Unit 1 does not have sufficient historical data to define a predictable and consistent [ ]. In addition, Unit 1 has some uncertainty defining the worth of individual control rods (e.g., aging if not new). Cycle 7 requires a whole-core reload, for which there is less industry experience. Cycle 8 will consist mostly of once-burned fuel, for which, again, there is little industry experience. Considering the above statements, justify why a local critical SDM demonstration is not warranted.

e. The response to SBWB-28, which was provided in a letter dated March 7, 2006, states that [ ] Address whether this shutdown margin value will also be applied to future Unit 1 cycles, including Cycle 8.

f. The June 30, 2006 response to RAI R2.1-1 provides the local critical eigenvalues for plant C (240 bundle core, yearly Cycle, 51.7 KW/l power density, 110% uprate, 1097 MWt, 17 % batch fraction, extended load line limit analyses [ELLLA] operating domain). Table R2.1-1 shows that for Cycle 30, Local 1, the difference between the design eigenvalue and the local measurement is -0.003 ΔK. In this case, criticality was reached [ ]. For this case (Local 1 Cycle 30), provide the actual calculated SDM. Justify why a SDM of 0.38 % should not be increased to the design value.

g. The response to SBWB-28 provides the cold critical calculation for Units 2 and 3. Unit 3 Cycle 10 shows a difference between projected and actual eigenvalue of [ ]. The plant reached criticality [ ]. For Unit 3 Cycle 10 provide the actual calculated SDM. Justify why a SDM of 0.38 % should not be increased to the design value. Given this level of uncertainty, justify why the TS required SDM should not be increased to the values seen in the analytical component of the SDM uncertainties. Based on this data, the potential of underprediction by a value greater 0.38 seems plausible.

38. Table 2-10 of NEDC-33173P provides sensitivity calculations of the impact of the 40% VF depletion assumption on the thermal overload protection (TOP) and maintenance outline procedure (MOP) for the LRNBP transient. The sensitivity analyses was based on a plant that differs from Unit 1 and may not be bounding. The following questions relate to the margins available for Unit 1 TOP/MOP.

    a. The sensitivity analyses show that the impact in terms of percent difference in TOP/MOP is approximately between [ ]. Address the corresponding TOP and MOP for GE13 and GE14 fuel types for Unit 1. Confirm that the LRNBP TOP/MOP for Unit 1 has sufficient margin available to account for the potential impact determined in the sensitivity analysis.

    b. Provide discussion on why the values established in the sensitivity analyses based on the reference plant are bounding for Unit 1 and the GE13 fuel loaded in the core.

    c. State the TOP/MOP limits for the GE14 fuel and GE13 fuel. Explain if these are the limits developed in the generic GE14 compliance to Amendment 22 or limits derived from specific GE14 and GE13 fuel design lattice loading types (e.g., plant-specific GE14 compliance to Amendment 22).

39. The NRC staff's assessment of GE's neutronic methods is based on improved version of TGBLA06. Specify the current NRC- approved production versions approved under Amendment 26 to GESTAR II (e.g. based on MFN-035-99 submittal). Also document the changes made to TGBLA06/PANAC11 since the approval of MFN-035-99 in Amendment 22 to GESTARII. State if the Unit 1 calculations are based on the Amendment 26 approved versions or the updated changes to TGBLA06/PANAC11. If the latter is true, provide the plant-specific information needed to support the necessary review and approval.

40. Describe the process followed by Tennessee Valley Authority to implement Long Term (L/T) Solutions including approved methodologies used, hardware modifications, and any interface between fuel vendors.

41. Address where Browns Ferry Nuclear Plant (BFN) is today in the implementation schedule and what is its implementation status. Address the affect, if any, of multiple fuel vendors.

42. Describe the BFN TSs affected by the L/T Solution implementation. Identify the related tech spec operability requirements.

43. Discuss the BFN experience with the period-based detection algorithm (PBDA) in response to noise and Solution III setpoint adjustment. Describe what actions are taken during cycle reload confirmations if the calculated setpoints are lower than expected. Describe the acceptance testing process used during the Solution III testing process. Include a description of PBDA results where false alarms were detected.

44. Describe any changes for the backup stability implementation (e.g. interim collective actions) associated with different fuel vendor's calculating method. Discuss whether BFN uses cycle-specific calculations for backup stability or generic regions. Describe

any Solution-III hardware implementation issues such as: location of the new hardware, periodic testing procedures, and signal response quality.

45. Describe the implications for operator training with respect to handling false alarms.

46. Describe what is the effect, if any, of the EPU upgrade on anticipated transient without scram and emergency operating instructions.

47. The NRC staff reviewed the response to SRXB A.11 submitted in the Unit 2 and 3 docket. In the response, the following statement was included:

> Because the maximum rod line does not change as a result of EPU, the power/flow history after RPT [recirculation pump trip] is similar for both EPU and pre-EPU.

This statement may be true for Units 2 and 3 since maximum extended load line limit analyses (MELLLA) was approved before the EPU. But for Unit 1, MELLLA is not approved, the maximum rod line is changed, and the reactor decay heat is increased due to EPU. Describe in detail the reasons for the suppression pool temperature decrease for Unit 1. Since the calculated pool temperature is 214 degrees F, sufficient justification is needed for the operability of emergency core cooling system pumps which may not meet the pump NPSH requirements due to cavitation.

48. Since Unit 1 is loaded with fresh GE14 and GE13, provide the peak data for the two fuel types.

a) For the peak power fuel assemblies, provide the limiting axial power distributions and radial peaking factors. For different exposures, select bundles with limiting axial power peaking operating with bottom peaked, double-hump or mid-peaked, and top peaked axial power distributions. Assure that the axial power distribution corresponding to the exposure with the highest hot bundle exit void fraction is also provided.

b) Include in the selected bundles, the power distribution and peaking corresponding to the maximum powered bundle selected for the cycle state point of 10 gigawatt days per standard ton. In the response to question 26, Figure 26-1 also shows that the bundle is operating at approximately 7.75 megawatts. Provide the corresponding predicted bundle operating conditions, including axial power distribution, void fraction distribution and bundle nodal exposure.

c) Also include the bundle inlet mass flow rate and inlet temperature.

BROWNS FERRY NUCLEAR PLANT

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