June 26, 2006

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
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Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE - ROUND 6 (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 (TVA, the licensee), 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 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.

With regards to the requests for additional information (RAIs) in the APLA section, the NRC staff reviewed the response to its original RAI (SPSB-A.11 - October 3, 2005, request) involving the use of containment accident pressure in the calculation of net positive suction head available to the core spray and low pressure coolant injection pumps. The response was provided in a letter dated March 23, 2006. The NRC staff notes that the licensee requested additional time to respond, provided the response later than committed, and failed to fully answer the question. As indicated in the March 1, 2006, letter to TVA, the timeliness and quality of the responses to the NRC staff's RAIs are essential to support the timely completion of this review. Further delays of this nature will significantly challenge the NRC staff's ability to support the requested completion date.

A response to the enclosed RAI is needed before the NRC staff can complete the review. This request was discussed with your staff on June 14, 2006, and it was agreed that a response would be provided by June 30, 2006. If you have any questions, please contact Ms. Eva Brown at (301) 415-2315.

Sincerely,

/RA/

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. RAI, Redacted Version
2. RAI, Proprietary Version

cc w/enclosure 1: See next page

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SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION REGARDING EXTENDED POWER UPRATE (TAC NO. MC3812)

Document Date: June 26, 2006

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REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

APLA (Previously SPSB-A)

22. It is recognized that the need to have containment accident pressure for emergency core cooling system (ECCS) net positive suction head (NPSH) should be based on a realistic analysis consistent with current probabilistic risk assessment (PRA) practices, as contrasted to a deterministic, design-basis calculation that employs excessive conservatism. Discuss which typical PRA accident sequences realistically require containment accident pressure in order to ensure that the ECCS pumps remain functional. This should include sequences currently modeled in the Browns Ferry PRA models or similar sequences, not currently modeled, that could be risk-significant if containment accident pressure is necessary and not available. This should also consider realistic fire scenarios, such as those considered in the Individual Plant Evaluation of External Events for Severe Accident Vulnerabilities study.

23. For each PRA accident sequence that realistically requires containment accident pressure, describe how much pressure is required and for what period of time.

24. For each accident sequence in #23 above, estimate the risk associated with the need for that accident pressure (i.e., the risk above the level that would exist if the ECCS pumps could function satisfactorily without the need for containment accident pressure). While a realistic core damage frequency and large early release frequency are the desired metrics for this risk estimate, the licensee may utilize sensitivity studies, bounding analyses or qualitative arguments, where appropriate, provided all conclusions are substantially supported by the discussion.

ACVB

37. The term design flow rate is used to describe the core spray pump flow rate and the residual heat removal (RHR) pump flow rate assumed in the NPSH analyses. Define precisely the “design flow rate” in terms of the pump and system curves.

38. The current Updated Final Safety Analyses Report Table 14.6-4 shows a higher drywell volume for Case 3, the limiting case for drywell pressure and temperature, than for Cases 1, 2 and 4. Discuss why there is a larger drywell volume assumed for this case, and whether the same assumption is made for the extended power uprate (EPU).

39. Provide the calculations used to determine the containment conditions (drywell, wetwell and suppression pool) for the loss-of-coolant accident (LOCA), Anticipated Transient Without Scram (ATWS), Station Blackout (SBO) and Appendix R Fire events.

40. Describe how the proposed crediting of containment accident pressure in determining available NPSH compares with the positions of Section 2.1.1 of Regulatory Guide 1.82, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, Revision 3, dated November 2003.

41. The units have drywell coolers which operate during normal plant operation. Address whether the drywell coolers are conservatively assumed to continue operation following accident initiation for the LOCA, ATWS, SBO and Appendix R Fire events.

42. Section 4.2.5 of the General Electric (GE) Analysis Report, PUSAR, states that the NPSH margins were calculated based on conservatively assuming RHR maximum flow rates and containment spray design flow rates in the short term analyses and RHR and containment spray design flow rates in the long term analyses. Describe the design provisions or operator actions that limit the pump flows to these values.

43. Describe how the make-up of nitrogen to the drywell and wetwell atmospheres could serve as a verification of containment integrity during normal operation.

44. Describe the measures taken to ensure that all containment penetrations are properly isolated prior to and during operation.

45. Describe any other actions/programs which contribute to assurance that the containment is isolated.

46. Address whether the RHR and core spray pumps can be throttled to increase available NPSH and decrease required NPSH. Discuss what, if any, guidance is provided in the emergency operating instructions (EOIs) or abnormal operating instructions regarding throttling these pumps to preserve NPSH margin during accident conditions.

47. Discuss whether any of the units have features to automatically terminate drywell or wetwell spray. Describe the conditions under which the operator would terminate drywell and/or wetwell spray under accident conditions in accordance with the EOIs. Address those measures put in place to prevent an operator from reducing wetwell pressure below that needed for adequate available NPSH.

48. In a letter dated September 4, 1998, Tennessee Valley Authority (TVA) requested the use of containment overpressure for Units 2 and 3. The letter stated that the short term NPSH analysis assumes a double-ended recirculation pump discharge line break while the long term analysis assumes a double-ended suction line break. Address whether this is the case for the EPU analyses. Any difference in assumptions should be explained.

49. Address the criteria in the EOIs for initiating drywell and wetwell sprays. Discuss how the timing of the actions resulting from these criteria compares with the 10-minute assumption in the accident analyses for initiating suppression pool cooling. Discuss

how the times for initiating drywell and wetwell sprays using the EOI criteria compare with times obtained in simulator training.

50. Using Figure ACVB 7-1 of the March 7, 2006, letter, explain the physical occurrences which result in (1) the reduction in the steep slope at approximately 2 seconds; (2) the small sudden increase at approximately 8 seconds; and (3) the following steep decrease. Discuss at what time the torus-to-drywell vacuum breakers to actuate.

51. Page E1-3 of the letter dated September 4, 1998, indicates that containment pressure is only needed in the short term for the RHR pump at the maximum flow conditions and that “other pathways are available and functional without containment overpressure being relied upon.” Discuss whether this is still true with the EPU NPSH analyses. If still true, elaborate on this statement.

52. In the safety evaluation dated September 3, 1999, on the credit for containment accident pressure in determining available NPSH, TVA discussed a 10-year frequency for suppression pool cleaning. Discuss whether suppression pool cleaning is still done on a 10-year frequency.

53. For Figures ACVB 7-3 and ACVB 7-4 from the March 7, 2006, letter, explain the physical occurrences that produce the significant changes in the shape of the curves as a function of time.

54. Table ACVB 22-1, in response to ACVB 22 from the March 7, 2006, letter, states that the licensing basis calculation of NPSH assumes no heat sinks while the realistic calculation does. Address whether the reverse should be true to ensure conservatism. Also, see TVA reply to ACVB 27 and Table SPSB-A.11-2, which states that not crediting heat sinks is conservative.

55. Table ACVB 22-1, in response to ACVB 22 from the March 7, 2006, letter, gives values of wetwell airspace and suppression pool volume that sum to different values for the realistic and the licensing basis values. Discuss whether the sums should be the same and equal to the total volume of the wetwell.

56. The response to RAI ACVB 18 provided curves of pressures and temperatures for the events crediting containment accident pressure for available NPSH. The curves for ATWS and Appendix R Fire should be extended to provide the total time that containment accident pressure is needed for available NPSH.

57. The response to RAI SPSB-A.11 provided Table SPSB-A.11-2, which contains calculations of suppression pool temperature with various assumptions. The cases are identified as either GE or TVA. Describe the analytical methods used for the TVA calculations and the steps taken to ensure a meaningful comparison with SHEX.

58. In Table 6 of Calculation MD-Q0999-970046, Rev. 8, provided in the March 23, 2006, response, the NPSH required (NPSHR) of the RHR pumps varies even when the pumps have the same flow rate. The Core Spray pumps, all with the same flow rate, also have the same value of NPSHR. Explain why the NPSHR varies even when the pumps have the same flow rate.

SBWB

26. Provide the following bundle operating conditions with exposure:

- maximum bundle power,
- maximum bundle power/flow ratio,
- exit void fraction of maximum power bundle,
- maximum channel exit void fraction,
- peak linear heat generation rate, and
- peak end-of-cycle (EOC) nodal exposure.

    Provide the maximum bundle operating conditions relative to EPU plants. Include the plant-specific data in the plots containing the high density and EPU plants maximum bundle operating conditions. Since there are no recent Unit 1 pre-EPU data and the units are similar, include the Units 2 and 3 pre-EPU data in the plots.

27. Provide quarter core map (assuming core symmetry) showing the bundle maximum linear heat generation rate and the minimum critical power ratio for beginning-of-cycle, middle-of-cycle and EOC. Similarly, show the associated bundle powers and exposures.

28. Figure 2-4 of licensing topical report, NEDC-33173P, Applicability of GE Methods to Expanded Operating Domain, shows the cold critical eigenvalues of reference plants. Figure 2-5 of NEDC -33173P shows the measured and predicted eigenvalues for Reference Plants. Provide the pre-EPU Units 2 and 3 cold critical eigenvalues measured and predicted differences for previous cycles based on the GE methods. Provide evaluation of any available local critical and startup shutdown calculations.

29. Provide a discussion addressing whether the traversing-in-core probes (TIPs) are gamma or thermal TIPs.

30. Based on the EPU Cycle core design, establish whether Unit 1 will experience bypass voiding [ ]. Specify the peak bypass calculated for any four bundle bypass zone at EPU conditions. Discuss why the bypass voiding is [ ]. Also calculate the bypass voiding for the second cycle where the large batches of fresh fuel loaded in Unit 1 will be at the most reactive state.

31. Based on the first/second EPU Cycle core design, determine the bypass voiding at the different local power range monitor elevations after a recirculation pump trip. Perform the calculations on limiting conditions (initial condition, axial power distribution and in-channel voids) and provide the results.

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BROWNS FERRY NUCLEAR PLANT

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