September 1, 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 9 (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 and 11, June 12, 15, 23 and 27, and July 6, 21, 24, 26, and 31, and August 4 16, and 18, 2006, Tennessee Valley Authority (TVA, the licensee) submitted 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 accident pressure 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 staff can complete the review. These requests were provided in draft from to your staff by e-mail and discussed on August 8-11, 2006. In discussions with your staff it was agreed that a response would be provided by September 15, 2006.

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

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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<u>DISTRIBUTION</u>: See next page

ADAMS Accession No. ML062350360 NRR-106

| OFFICE | LPL2-2/PM | LPL2-2/PM | LPL2-2/LA | DSS/SBWB |
|---|---|---|---|---|
| NAME | EBrown | EBrown for MChernoff | BClayton | JWermeil by memo |
| DATE | 9/ 01 /06 | 09/01/06 | 09/01/06 | 8/16/06 |

| OFFICE | APLA/BC | DSS/ACVB | LPL2-2/BC |
|---|---|---|---|
| NAME | MRubin by memo | TMartin by memo | LRaghavan |
| DATE | 8/31/06 | 8/22/06 | 09/01/06 |

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SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE - ROUND 9 (TS-431) (TAC NO. MC3812)

Dated: September 1, 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

25. In various correspondence the NRC staff has noted different values for the suppression pool (SP) bulk temperature limit:

    a. Section 4.8.6.2, Page 4.8-4 of the updated final safety analyses report (UFSAR) states a limit of 177 degrees Fahrenheit (EF), based on an analysis of the torus attached piping.

    b. The limit of 177 EF was used in the previous 5 percent power uprate for Units 2 and 3 (ADAMS Accession No. ML042670045).

    c. The draft Unit 1 Fire Protection Program Report (ADAMS Accession No. ML060620424) provides various limits as follows:

        i. Page 301- The design limit is 281 EF.
        ii. Page 309 - The residual heat removal (RHR) pump seals were rated for 160 EF, but have been re-evaluated for 215 EF.

    d. Table 4-1 of Enclosure 4 of the submittals dated June 28 and 25, 2004, uses the 281 EF limit. Provide the correct SP bulk temperature limit for evaluating the proposed containment accident pressure (CAP) credit.

26. Analysis (e.g., the August 4, 2006, submittal) indicates that containment accident pressure (CAP) credit is required to ensure adequate net positive suction head (NPSH) to the RHR pumps during an Appendix R scenario. The NRC staff understands that CAP credit is required for the pre-EPU [extended power uprate] plant as well as for the post-EPU plant. The Fire Protection Program Report defines the Appendix R scenario as a fire that results in a total loss of high-pressure makeup sources (feedwater (FW), high pressure coolant injection, and reactor core isolation cooling), followed by manual depressurization using three S/RVs and operation of one RHR pump and its associated heat exchanger in low pressure coolant injection (LPCI) mode (i.e., no suppression pool cooling(SPC)).

    For transient initiating events (e.g., loss of FW), the probabilistic risk assessment (PRA) credits manual depressurization using the S/RVs and use of either core spray (CS) or LPCI, along with SPC, upon the failure of all high-pressure makeup sources. The PRA also includes sequences initiated by transient events that lead to multiple stuck-open S/RVs (e.g., loss of FW and subsequent MSIV closure, which causes the S/RVs to open, followed by subsequent failure of the S/RVs to reclose). The previous risk evaluation of the proposed CAP credit does not address these types of accident sequences.

Provide a risk evaluation of the proposed CAP credit that includes the increase in core-damage frequency and, large early release frequency due to sequences that are initiated by transient events that lead to either (a) manual depressurization via the S/RVs and use of CS or LPCI upon the total loss of high-pressure makeup sources, and (b) sequences that are initiated by transient events that lead to multiple stuck-open S/RVs.

<u>ACVB</u>

62. The August 4, 2006, response to Request for Additional Information (RAI) Risk Assessment Containment & Ventilation Branch (ACVB) 37/35 states that, for the CS pump, the operator is instructed to maintain flow less than 4000 gallons per minute (gpm) and within the NPSH limit curves. However, for determining adequate NPSH, it is assumed that the operator would reduce flow in response to the NPSH limit curves, but not less than 3125 gpm.

    It appears that at a flow rate of 4000 gpm and the peak calculated suppression pool temperature, the pumps are in the acceptable region of the Emergency Operating Instruction NPSH limit curves. Therefore, explain what prompts the operator to reduce flow to 3125 gpm. If the operator can operate acceptably at 4000 gpm, address why shouldn't this more conservative flow rate be used in the NPSH analyses.

63. In the July 21, 2006, response to RAI Probabilistic Risk Assessment Licensing Branch A (APLA) 24/26, five fire areas are described. For those fire areas for which the safety analysis depends on RHR pumps (control room and turbine building), 2 RHR pumps are said to be available. Address why only one RHR pump is credited for the Appendix R analyses and NPSH analyses.

64. Enclosure 4 of the August 4, 2006, letter contains Calculation MDQ099920060011, Transient NPSH/ Containment Pressure Evaluation of RHR and CS Pumps. For the short term loss-of-coolant accident response, Figure 7.5 of Calculation MDQ099920060011 shows that the wetwell pressure required is less than the wetwell pressure available for the RHR pumps pumping into the broken recirculation loop. TVA indicated this was acceptable based on RHR pump tests reported in Enclosure 2 to a May 21, 1976 TVA letter to the NRC. A margin of 9 feet was shown to be available in these tests relative to the required NPSH based on a 3 percent head drop.

    (i) Provide the margin between the lowest NPSH value of the cavitation tests reported in the May 21, 1976 letter and the reduced required NPSH values used in Tennessee Valley Authority (TVA) Calculation MDQ099920060011.

    (ii) Discuss the difference between the required NPSH and the available NPSH at 600 seconds.

    (iii) Describe how the required NPSH value of 28.4 ft in Figure 7.5 of Calculation

MDQ099920060011 was obtained.

65. Table 10-2 of Enclosure 4 to the June 28, 2004, submittal, NEDC-33101P, DRF 0000-0010-9439, Browns Ferry Unit 1 Safety Analysis Report for Extended Power Uprate (PUSAR), shows that the peak drywell air temperature due to a steam line break (336 °F) exceeds the containment shell design temperature limit (281 °F). Verify that the shell temperature itself remains below the 281 °F design limit.

66. Provide the maximum RHR and core spray pump seal temperatures. If less than the calculated peak suppression pool temperatures, address why this is acceptable.

67. Provide the maximum acceptable temperature of the piping attached to the torus. If less than the maximum suppression pool water temperature, address why is this acceptable.

<u>SBWB</u>

49. Provide the head flow curves used in the limiting large break loss-of-coolant accident analyses (battery failure case). The curves should include the head flow curve for one low pressure core spray and one low pressure coolant injection pump discharging into each recirculation line. Also, provide the limiting axial power shape used in this limiting break.

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