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, UNITS 2 AND 3 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE - ROUND 9 (TS-431) (TAC NOS. MC3743 AND MC3744)

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, July 21, and August 4 and 18, 2006, the Tennessee Valley Authority (the licensee) submitted amendment requests for Browns Ferry Nuclear Plant, Units 2 and 3. The proposed amendments would change the Units 2 and 3, operating licenses to increase the maximum authorized power level from 3458 to 3952 megawatts thermal. This change represents an increase of approximately 15 percent above the current maximum authorized power level. 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 from to your staff by e-mail and discussed on August 8-10, 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) 415-2315.

Sincerely,

/RA/

Eva A. Brown, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

Enclosure: Request for Additional Information

cc w/enclosure: See next page

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Docket Nos. 50-260 and 50-296
Enclosure: Request for Additional Information

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

ADAMS Accession No. ML062420043 NRR-106

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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, UNITS 2 AND 3 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE - ROUND 9 (TS-418) (TAC NOS. MC3743 AND MC3744)

Dated: September 1, 2006

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REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260, AND 50-296

APLA

27. In various correspondence the Nuclear Regulatory Commission (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 supression pool (SP) bulk temperature limit for evaluating the proposed containment accident pressure (CAP) credit.

28. Analysis (e.g., the August 4, 2006 submittal) indicates that 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-extended power uprate (EPU) 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 main steam isolation valve (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 (CDF) and, large early release frequency (LERF) 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.

ACVB

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>

65. Provide the head flow curves used in the limiting large break loss-of-coolant accident LBLOCA 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.

66. In the Reload Analysis Report (RAR), submitted June 12, 2006, different minimum critical power ratio (MCPR) values are given for different operating conditions. However, the operating MCPR for normal operation (base case) with all the equipment available is not given.

    a. Provide the operating limit MCPR with all equipment in operation.

    b. Address which transient is the most limiting transient out of the five transients given on page 5-1 in determining the operating MCPR.

    c. Provide a table indicating the limiting transient for pressurization and non-pressurization transients.

67. In the RAR for Unit 2, the SLMCPR assumed for two loop operation is 1.08, but in the proposed Technical Specification (TS) 2.1.1.2, the safety limit MCPR (SLMCPR) is specified as 1.07. Address which is correct. For Unit 3, the proposed TS SLMCPR is 1.08. Address why the proposed SLMCPR values are different.

68. As stated in the Executive summary of Enclosure 5 of the June 25, 2004, submittal EMF-2982(P), or the Framatome Uprate Safety Analysis Report (FUSAR), the FUSAR provides results for the fuel-related analyses for a reference core of ATRIUM-10 fuel. Therefore, for fuel related issues concerning Units 2 and 3, the NRC staff has focused the review on the FUSAR rather than Enclosure 4 of the June 25, 2004, submittal Power Uprate Safety Analysis Report (PUSAR) which contained the fuel related analyses for a reference core of GE-14 fuel. For many of the RAI responses for fuel-related issues, TVA refers to the PUSAR rather than the FUSAR. For example, in response to

SRXB-A.2, TVA stated that:

> ...the conclusions of the PUSAR, NEDC-33047, are applicable and bounding for both Units 2 and 3.

Also, in response to SRXB-A.22, TVA stated that:

> The scenario and sequence of events remain valid for the fuel-related EPU analyses and are consistent with the event descriptions presented in the UFSAR.

Confirm that similar conclusions can be made for the FUSAR.

Additionally, in response to SRXB-A.22, TVA stated that:

> In most cases, the PUSAR analysis remains applicable for ATRIUM-10 fuel.

Identify the areas where the PUSAR analysis is not applicable for ATRIUM-10.

69. For Units 2 and 3, RAI SRXB-A.9 indicates that the peak calculated pressure for the reactor overpresssure analysis is 1204 pounds per square inch gage (psig). Address whether the response is applicable only for Unit 2 and 3, or does it also apply to Unit 1. Confirm that the proposed Units 2 and 3 TS Surveillance Requirement (SR) 3.1.7.6 standby liquid control system pump discharge test pressure of 1275 psig is satisfactory considering the operating margin for the pump discharge relief set pressure. Address why the change in pressure proposed for Unit 1 (1275 psig to 1325 psig in SR 3.1.7.6) is not applicable for Units 2 and 3.

70. In RAR Section 5.6, Fuel Loading Error, the acceptance criteria is given, provide the associated analyses for uprated conditions.

71. In RAR Section 5.6.1, identify the topical report and the evaluation model used for the Mislocated Fuel Assembly event.

72. In RAR Section 5.6.2, identify the topical report and the evaluation model used for the Misoriented Fuel Bundle.

73. Table 9.2 of the FUSAR does not include the following events: Loss of Auxiliary Power, Main Condenser Vacuum, Recirculation Flow Controller Failure, Trip of one pump, Trip of two pumps, Recirculation flow controller failure, or Start-up of idle pump. Confirm whether these events were analyzed and documented.

74. Address why the anticipated transient without scram analysis was not done in the RAR.

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BROWNS FERRY NUCLEAR PLANT

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