Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Facility Operating License Nos. DPR-33, DPR-52, DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Technical Specifications Changes TS-431 and TS-418 – Extended Power Uprate – Response to Round 25 – Request for Additional Information – SCVB-82 Through SCVB-86 and Replacement Documentation**

References:

1. Letter from TVA to NRC dated June 28, 2004, "Browns Ferry Nuclear Plant (BFN) Units 2 and 3, Proposed Technical Specifications (TS) Change - 431, Request For License Amendment Extended Power Uprate (EPU) Operation" [ML041840109]

2. Letter from TVA to NRC dated June 25, 2004, "Browns Ferry Nuclear Plant (BFN) Unit 1, Proposed Technical Specifications (TS) Change - 418, Request For License Amendment Extended Power Uprate (EPU) Operation" [ML041840301]

3. Letter from TVA to NRC dated April 10, 2009, "Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, Technical Specifications (TS) Changes TS- 431 and TS-418, Extended Power Uprate (EPU), Transmittal of Containment Parameters and Total Shutdown Power Fractions" [ML091060381]

4. Letter from NRC to TVA dated April 22, 2009, "Browns Ferry Nuclear Plant, Units 1 and 2 - Request for Additional Information for Extended Power Uprate - Round 23 (TS-431 and TS-418)" [ML091000283]

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5. Letter from TVA to NRC dated April 29, 2009, "Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, Technical Specifications (TS) Changes TS- 431 and TS-418, Extended Power Uprate (EPU), Response to Request for Information Regarding Containment Parameters for NRC Confirmatory Analysis" [ML091250177]

6. Letter from NRC to TVA dated July 15, 2009, "Unit 1, Request for Additional Information for Extended Power Uprate - Round 25" [ML091880959]

7. Letter from TVA to NRC dated August 31, 2006, "Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, Technical Specifications (TS) Changes TS- 431 and TS-418, Extended Power Uprate (EPU), Replacement Documentation" [ML062510371]

8. Letter from TVA to NRC dated September 19, 2008, "Browns Ferry Nuclear Plant (BFN) Units 2 and 3, Technical Specifications (TS) Change TS-418, Extended Power Uprate (EPU), Supplemental Response to Request for Additional Information (RAI) Rounds 3 and 18 and Response to Round 20 Fuels Methods RAIs [ML082690016]

In letters dated June 28, 2004, Reference 1, and June 25, 2004, Reference 2, TVA submitted license amendment requests to the NRC for the Extended Power Uprate (EPU) operation of BFN Unit 1 and BFN Units 2 and 3, respectively. The proposed amendments would change the operating licenses to increase the maximum authorized core thermal power level of each reactor by 14.2 percent to 3952 megawatts.

In Reference 3, on April 10, 2009, TVA submitted a modified version of Form OPL-4a, which is used to document containment analysis input parameters. TVA subsequently received questions on the April 10, 2009, submittal and several additional NRC requests for BFN physical data related to primary containment components. The NRC information requests were documented in the Round 23 Request for Additional Information (RAI) dated April 22, 2009, (Reference 4) which TVA responded to on April 29, 2009 (Reference 5).

This letter responds to RAI Round 25, dated July 15, 2009, (Reference 6), which contains five new NRC questions. SCVB-84 and SCVB-85 are follow-up questions on the April 10, 2009, submittal. SCVB-82 and SCVB-83 are related to TVA calculation MDQ0999970046, Revision 10, "NPSH Evaluation of Browns Ferry RHR and CS Pumps," which was submitted to NRC on August 31, 2006 (Reference 7). SCVB-86 is a request for additional physical data. The TVA RAI responses are provided in Enclosure 1.

Enclosure 2 is a compact disc (CD) copy of four net positive suction head (NPSH) calculation spreadsheets that were previously transmitted to NRC by e-mail on June 15, 2009. In an unrelated matter, on page E2-1 of Enclosure 2 of TVA's September 19, 2008, submittal (Reference 8), some AREVA proprietary information was inadvertently not removed from the non-proprietary copy. A replacement page E2-1 is provided in Enclosure 3.

TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed Technical Specification (TS) changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

No new regulatory commitments are made in this submittal. Should you have any questions concerning this letter, please contact J. D. Wolcott at (256) 729-2495.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on this 29 day of July, 2009.

Respectfully,

R. M. Krich
Vice President
Nuclear Licensing

Enclosures: Enclosure 1 – Response to Round 25 Request for Additional Information (RAI) SCVB-82 Through SCVB-86
Enclosure 2 – CD Copy of NPSH Spreadsheets Previously Sent to NRC
Enclosure 3 – Corrected Page E2-1 for September 19, 2008, TVA Submittal

cc: See Page 4

Enclosures
cc (Enclosures):

NRC Regional Administrator – Region II

NRC Senior Resident Inspector – Browns Ferry Nuclear Plant

State Health Officer – Alabama Department of Public Health

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418
EXTENDED POWER UPRATE (EPU)

RESPONSE TO ROUND 25 REQUEST FOR ADDITIONAL INFORMATION (RAI)
SCVB-82 THROUGH SCVB-86

SCVB (formerly ACVB)

NRC SCVB-82

In Tables 7.1 through 7.39 of Enclosure 1 to a letter dated August 31, 2006, a pressure is indicated in support of calculation of the NPSHa. Identify which pressure is represented in the table and provide information on where the pressure was measured as referenced from the pump [residual heat removal (RHR) and core spray (CS)] suction centerline. Also provide bounding values of the pump suction pressures for the DBA-LOCA, station blackout (SBO), anticipated transient with scram (ATWS), Appendix R, Steam Line Breaks that were used to calculate the actual NPSHa.

TVA Response to SCVB-82

RAI SCVB-82 pertains to TVA calculation MDQ0999970046, Revision 10, "NPSH Evaluation of Browns Ferry RHR and CS Pumps", which was submitted to NRC as Enclosure 1 of the TVA letter dated August 31, 2006 (Reference 7).

The pressure values given in Tables 7.1 through 7.40 of calculation MDQ0999970046, Revision 10 are the calculated gauge pressures at the designated pump suction center line elevations and are obtained from the Multiflow model output files. Table 8 on page 50 of the calculation provides the mathematical formula used to calculate the individual table cells and for the determination of available net positive suction head (NPSHa). The bounding pump suction pressures from Tables 7.1 through 7.40 for the Design Basis Accident (DBA)-Loss-of-Coolant Accident (LOCA), SBO, ATWS, and Appendix R analyses have been compiled into a table, which is included in the response to the next RAI, SCVB-73. Values for steam line breaks are not calculated since they are bounded by DBA-LOCA.

NRC SCVB-83

Provide the friction pressure loss (includes strainer loss, header and inlet piping loss) up to the suction flange of the RHR and CS pumps for flows under all events analyzed in Enclosure 1 to a letter dated August 31, 2006.

TVA Response to SCVB-83

The friction pressure loss up to the pump suctions, $P_{friction}$, is determined using the following equation:

$$P_{friction} = P_{static} - P_{suction}$$

where P_{static} = (suppression pool water level elevation - pump elevation) / conversion factor (pounds per square foot absolute (psia) to feet)) and $P_{suction}$ is the calculated suction at the pump centerline. The pump centerline elevations used in the calculation are:

CS pumps = 521.3 feet
RHR pumps = 521.6 feet

The suppression pool water level elevation used for LOCA and SBO is 535.711 feet and 536.062 feet for ATWS and Appendix R. Refer to pages 27 through 30 in calculation MDQ0999970046 for the derivation of these elevation values. P_{static}, $P_{suction}$, and $P_{friction}$ are presented in the below table for the bounding cases.

Pump Suction Pressure and Friction Pressure Loss											
Event	Pump	NPSH Calculation Spreadsheet	Bounding Cases						Friction Pressure Loss		
			Case Designation (see Table 6*)	Table	Limiting Pump for Case	Temperature (° F)	Suction Pressure (psig)	NPSHa (feet)	P_{static} (psi)	$P_{suction}$ (psig)	$P_{friction}$ (psi)
LOCA ST	CS	EPU_RAI_6_LOCA_R1	LOCA 2B	Table 7.4	CS Pump A	155.4	-1.332	20.79	6.114	-1.332	7.446
	RHR-IL	EPU_RAI_6_LOCA_R1	LOCA 2B	Table 7.4	RHR Pump B	155.4	-0.097	23.70	5.986	-0.097	6.083
	RHR-BL	EPU_RAI_6_LOCA_R1	LOCA 2B	Table 7.4	RHR Pump C	155.4	-1.364	20.71	5.986	-1.364	7.350
LOCA LT	CS	EPU_RAI_6_LOCA_R1	LOCA 3C	Table 7.7	CS Pump A	187.3	3.17	20.85	6.046	3.170	2.876
	RHR	EPU_RAI_6_LOCA_R1	LOCA 3C	Table 7.7	RHR Pump C	187.3	4.638	24.35	5.920	4.638	1.282
ATWS	RHR	EPU_RAI_6_ATWS	ATWS	Table 7.20	RHR Pump C	211	4.408	10.56	6.011	4.408	1.603
APP R	RHR	EPU_RAI_6_APPR_R3	APP R - C	Table 7.35	RHR Pump C	223	4.836	3.33	5.981	4.836	1.145
SBO	RHR	EPU_RAI_6_SBO	SBO A/C	Table 7.39	RHR Pump C	200	4.726	18.18	5.891	4.726	1.165

* *Table 6 is contained in calculation MDQ0999970046*

Abbreviations
°F = degrees Fahrenheit
psig = pounds per square inch gauge
psi = pounds per square inch absolute

NRC SCVB-84

Item q on page E1-12 in Enclosure 1 to a letter dated April 10, 2009, describes the mode of RHR suppression pool cooling as modeled in the SHEX containment analysis after the drywell pressure drops below 2.6 psig. Provide the time that this mode is initiated during the transient. Describe how a negative pressure in the containment is prevented, in the SHEX containment analyses, while the suppression pool cooling continues, given that the reactor building-to-wetwell vacuum breakers are not modeled.

TVA Response to SCVB-84

Item q, *Drywell pressure above which drywell sprays can operate,* represents the RHR system interlock that prevents drywell spray from being initiated if drywell pressure is not greater than the interlock pressure. The drywell spray mode of RHR requires that the operator manually open the spray valves. In the LOCA analysis, it is assumed that the operator initiates drywell spray at 10 minutes (per page E1-10, item d.1 in the April 10, 2009 submittal (Reference 3)). At 10 minutes, the drywell pressure is much greater than the interlock, so spray would be permitted and SHEX models drywell spray as being initiated. In the limiting LOCA NPSH analysis, drywell pressure remains above the interlock value for the analysis duration and drywell spray operates during the entire duration.

NRC SCVB-85

Enclosure 1 to the letter dated April 10, 2009, page E1-12, item y addresses shutoff of LPCI above a certain water level. Provide the water level at which LPCI will be automatically shutoff, and discuss how the LPCI shutoff water level is addressed in the SHEX containment analysis. Also address the resultant water level should the RHR system be aligned to continue to support LPCI, with the RHR heat exchangers in use, and aligned in the alternate cooling mode.

TVA Response to SCVB-85

Item y in the April 10, 2009, OPL-4 submittal is *Vessel water level above which LPCI will be shut off* and has a value of 588 inches vessel elevation, which is approximately the highest water level that the reactor water level normal and emergency range instrumentation in the control room can indicate (equivalent to 60 inches vessel level instrument indication). Item h on page E1-13 is the same entry for CS. The RHR Low Pressure Coolant Injection (LPCI) mode of operation has no automatic high reactor water level shutoff features and it is assumed that during accidents and events other than Appendix R, the operator maintains reactor water level below this value. This is consistent with the directions provided in the plant Emergency Operating Instructions.

In the Appendix R event, LPCI is used to inject and flood the reactor to above the elevation of the main steam lines (approximately 670 inches vessel elevation/634 feet reactor building elevation) to establish a cooling loop from the suppression pool, through the RHR heat exchangers, to the vessel, into the main steam lines, and returning to the suppression pool via the main steam Safety Relief Valve (SRV) tail pipes. This mode of operation is commonly referred to as Appendix R alternate shutdown cooling.

NRC SCVB-86

Provide the flow areas for each of the main steam relief valves.

TVA Response to SCVB-86

Each of the 13 SRVs has a valve orifice diameter of 5.125 inches.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418
EXTENDED POWER UPRATE (EPU)

CD COPY OF NPSH SPREADSHEETS PREVIOUSLY SENT TO NRC

The enclosed compact disc contains a zip file with the following four net positive suction head spreadsheets that were previously sent to NRC by e-mail on June 15, 2008.

1) EPU_RAI_6_LOCA.xls
2) EPU_RAI_6_LOCA_R1.xls
3) EPU_RAI_6_APPR_R2.xls
4) EPU_RAI_6_APPR_R3.xls

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGES TS-431 AND TS-418
EXTENDED POWER UPRATE (EPU)

CORRECTED PAGE E2-1 FOR SEPTEMBER 19, 2008, TVA SUBMITTAL

On page E2-1 of Enclosure 2 of TVA's September 19, 2008, submittal (Reference 8), some AREVA proprietary information was inadvertently not removed from the non-proprietary copy. A replacement page is attached.

NRC staff conducted an audit of AREVA fuel methods from August 18 through August 28, 2008, at the AREVA engineering facilities in Richland, Washington. As a result of the audit, TVA agreed to provide supplemental responses Round 3 RAIs SRXB-A.34 and SRXB-A.42. The previous Round 3 responses were originally submitted on March 7, 2006 (ML060680853). A revised response to SRXB-A.34 was also submitted on May 11, 2006 (ML061360148).

NRC RAI SRXB-A.34 (From Round 3)

Describe qualitatively the cross-section reconstruction process incorporated in CASMO-4 and MICROBURN-B2. The response should reflect the information provided in the slides (1-35) of the August 4 presentations, including high void fraction effects and accuracy. Provide flow chart(s), road map(s) and any other means to demonstrate the process, starting from the gathered raw void fraction data, how that data is used by CASMO-4 to generate the required cross-sections. In addition, briefly describe the development of the void fraction correlation and associated uncertainties.

Supplemental Response to SRXB-A.34

MICROBURN-B2 versions prior to 2003 treated cross section dependency on spectral history differently between the fuel nuclide depletion module and the neutron flux calculation module. The fuel nuclide depletion module used **[ ]** while the neutron flux iteration calculation module used a **[ ]**. This inconsistency was remedied starting in 2003 by changing the depletion module to the **[ ]**. Starting from 2006, both modules were converted to the **[ ]**.

These changes over the years were mainly due to code maintenance concerns and did not impact any result due to the **[ ]**. Unlike the cross section dependency on the instantaneous void, the **[ ]** is rather weak. This is shown in Figure SRXB-A.34.1 for Pu-239 and in Figure SRXB-A.34.2 for Pu-240. The **[ ]**. At the high end of **[ ]**, the difference between the **[ ]**. This kind of difference is entirely within the uncertainty of nuclear cross section measurement and its evaluation process including the CASMO-4 lattice code. It has no observable effect on the reactor nodal power distribution and the reactor criticality evaluation as has been verified in the code maintenance record of MICROBURN-B2.