Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

February 26, 2007

TVA-BFN-TS-431
TVA-BFN-TS-418

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
	)	50-296

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 12 REQUEST FOR ADDITIONAL INFORMATION (RAI) - (TAC NOS. MC3812, MC3743, AND MC3744)

By letters dated June 28, 2004 (ADAMS Accession No. ML041840109) and June 25, 2004 (ML041840301), TVA submitted license amendment applications for the EPU of BFN Unit 1 and BFN Units 2 and 3, respectively. On January 26, 2007 (ML070190473), the NRC staff issued the Round 12 RAI regarding the EPU license amendment requests.

Enclosure 1 to this letter provides TVA's responses to the fourteen Round 12 RAI questions. In addition, included in Enclosure 1 is supplemental information regarding Round 10 RAI question EEMB-118, which TVA originally addressed by letter dated October 5, 2006 (ML062860267).

Enclosure 1 contains information that General Electric Company (GE) and Areva NP, Inc. (Areva) consider to be proprietary in nature and subsequently, pursuant to 10 CFR 9.17(a)(4), 2.390(a)(4) and 2.390(d)(1), such

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information should be withheld from public disclosure. Enclosure 2 is a redacted version of Enclosure 1 with the proprietary material removed and is suitable for public disclosure. Enclosure 3 contains affidavits from GE and Areva supporting this request for withholding from public disclosure.

TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed 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).

One new regulatory commitment has been made in this submittal as noted in Enclosure 4. If you have any questions regarding this letter, please contact me at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 26th day of February, 2007.

Sincerely,

William D. Crouch
Manager of Licensing
and Industry Affairs

Enclosures:
1. Response to Round 12 Request for Additional Information - (Proprietary Information Version)
2. Response to Round 12 Request for Additional Information - (Non-Proprietary Information Version)
3. GE and Areva Affidavits
4. List of Regulatory Commitments

Enclosures
cc (Enclosures):

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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)

RESPONSE TO ROUND 12 REQUEST FOR ADDITIONAL INFORMATION
(NON-PROPRIETARY INFORMATION VERSION)

This enclosure provides a redacted version of TVA's response to NRC's January 26, 2007, Round 12 Request for Additional Information regarding TVA's request for EPU of BFN Units 1, 2, and 3.

NRC RAI SBWB-57 (Unit 1)

State what reload analyses will be performed to support the actual Cycle 7 extended power uprate (EPU) operation plan, which could consist of initial 105-percent power operation followed by subsequent operation at EPU conditions.

TVA Response to RAI SBWB-57 (Unit 1)

Two complete and separate reload licensing analyses have been performed for BFN Unit 1 Cycle 7, one for operation at 105% core power and another for EPU operation at 120% core power, i.e., 120% of original licensed thermal power (OLTP). The Supplemental Reload Licensing Report (SRLR) for EPU (120% OLTP) operations was transmitted to NRC on May 15, 2006 (ML061450390). The SRLR for 105% OLTP operations was submitted on January 29, 2007 (ML070320392). Together the two sets of analyses envelope initial 105% OLTP operation followed by subsequent operation at EPU conditions.

NRC RAI SBWB-58 (Unit 1)

For Cycle 7, address whether the Supplemental Reload License Report (SRLR) and Core Operating Limits Report (COLR) will be revised based on the actual operating plan. Discuss any deviation from the standard reload analysis set, and address whether any analyses will be dispositioned based on a "bounding" analyses approach.

TVA Response to RAI SBWB-58 (Unit 1)

As discussed in the response to RAI SBWB-57 above, two complete sets of standard reload licensing analyses have been performed for 105% and 120% core power, and submitted to NRC. Each of these reload licensing analyses addressed the standard reload analysis set. Together the two sets of analyses envelope initial 105% power operation followed by subsequent operation at EPU conditions.

A COLR for the initial 105% power operation will be issued prior to Unit 1 restart. Prior to operation at EPU conditions, the COLR will be revised to incorporate EPU based core operating limits in accordance with Technical Specification 5.6.5, "Core Operating Limits Report."

NRC RAI SBWB-59 (Unit 1)

Since there is no specific operational plan for Cycle 7, explain how the maximum decrease in the shutdown margin (SDM) at cycle exposures other than beginning-of-cycle (e.g., R value) will be

calculated before the restart of the plant. Address whether the R value will be calculated for the assumed operation at 105 percent and at 120 percent, or if the SDM calculation will be revised based on the actual operational plan.

TVA Response to RAI SBWB-59 (Unit 1)

Figure SBWB-59-1 shows the R-value for the core operating at both 105% and 120% power. The core loading is the same for both reload analyses, and because the void and control history impact is negligible, the SDM results are nearly identical. Since SDM is evaluated for operation at both power levels, the two reload analyses bound the effects of power level and power shape for actual operations. The core average control rod fractions at hot operating conditions for the 105% and 120% core power are provided in Figure SBWB-59-2.

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Figure SBWB-59-1

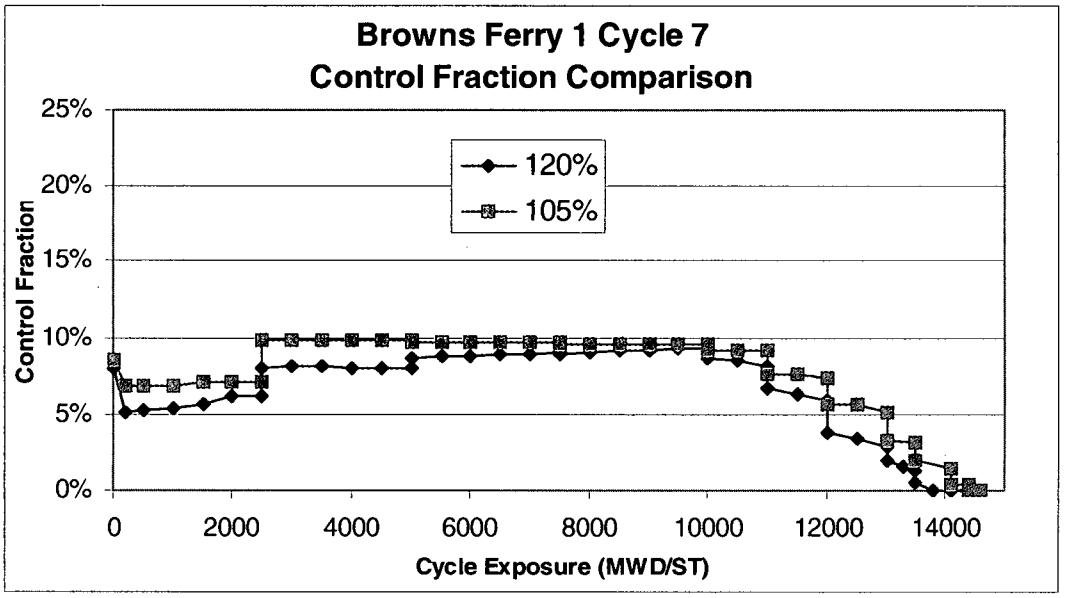

Figure SBWB-59-2

NRC RAI SBWB-60 (Unit 1)

Before implementation of the Unit 1 restart, provide the COLR and the SRLR for the finalized Cycle 7 operational plan. If separate reload SRLRs were performed, provide both of them.

TVA Response to RAI SBWB-60 (Unit 1)

The SRLR for EPU (120% OLTP) operations was transmitted to NRC on May 15, 2006 (ML061450390). The SRLR for 105% OLTP operations was submitted on January 29, 2007 (ML070320392). A COLR for the initial 105% power operation of Unit 1 will be submitted to NRC prior to unit restart.

NRC RAI SBWB-61 (Unit 1)

In a request for additional information (RAI) dated August 10, 2006, the Nuclear Regulatory Commission (NRC) staff asked in SBWB question 37d. why a local critical SDM demonstration was not warranted for Unit 1 restart. The NRC staff recognizes that disabling the banked position withdrawal sequence, which mitigates a control rod (CR) drop accident, was not prudent in order to perform the local critical SDM demonstration. However, considering the uniqueness of Unit 1 Cycle 7 an-d 8, performing the startup physics test may still be prudent and warranted.

Discuss whether the one-rod-out subcriticality test will be performed before the startup-criticality CR pulls to ensure that there is sufficient subcriticality with the highest worth CR withdrawn. Address whether this will be performed for all the analytically-determined "high worth" CRs before restart of the criticality CR pulls.

TVA Response to RAI SBWB-61 (Unit 1)

Fuel loading and core verification have been completed on Unit 1. To ensure that the control rod drives/control rods are functioning properly prior to startup, a number of tests are performed on each individual control rod, including full withdrawal of the rod to check travel, rod position indication, and stroke time, among other attributes. This affords an opportunity to identify problems, and repair control rods and associated instrumentation while the reactor is shutdown and the components are more readily accessible. All Unit 1 control rods are currently new.

As a prerequisite to this control rod testing, the analytically determined strongest rod is withdrawn and subcriticality confirmed. This one-rod-out subcriticality test has been completed. Since all of the remaining control rods will be tested as described above, a complete subcriticality check of the core will be equivalently performed prior to restart.

TVA's August 18, 2006, response to the RAI Round 8 questions, which includes a response to RAI SBWB-37, explains in detail the reasons for not performing a local critical SDM demonstration and also discusses the excess SDM criterion used for the Cycle 7 core. Accordingly, TVA will perform the insequence criticality surveillance test to verify SDM in accordance with Technical Specification Surveillance Requirement 3.1.1.1. Also, as stated in the response to question SBWB-37.e in RAI 8, TVA considers that the excess SDM design margin used for the first EPU core (Cycle 7) can be reduced for the Cycle 8 core to a value that is more typical of standard core designs.

NRC RAI SBWB-62 (Unit 1)

State all planned startup physics tests.

TVA Response to RAI SBWB-62 (Unit 1)

TVA will perform the insequence criticality surveillance test to verify SDM in accordance with Technical Specification Surveillance Requirement 3.1.1.1.

NRC RAI SBWB-63 (Unit 1)

a. Provide the rod density and the R value for the SDM calculation, based on the combined power level operation.

b. Discuss if the operation at the lower power level will decrease the additional design SDM (e.g., the 1.5-percent design margin for EPU), considering the CR patterns and the power distributions expected relative to the projected EPU operation.

c. Provide a comparative assessment of the components of the design SDM to show that operation at the lower power level will not result in lower available design margin to account for potentially higher modeling uncertainties. Specifically, provide assurances that the control of the additional reactivity through deep CRs for operation at the lower power levels will not result in a higher CR worth, such that the additional design margin is reduced.

TVA Response to RAI SBWB-63 (Unit 1)

a. The R-value and core average control rod fraction for the 105% OLTP and 120% OLTP reload analyses are provided in Figures SBWB-59-1 and SBWB-59-2, respectively. The minimum cold SDM based on 105% operation is slightly lower than for EPU operation; however, the minimum SDM is still maintained above the 1.5% design goal for both power levels.

b. Since SDM is evaluated for operation at both power levels, the two analyses bound the effects of power and power shape for actual operations.

c. Two components of the SDM are the all-rods-in reactivity (k-effective) and the maximum rod worth, which are equal to the difference between the all-rods-in reactivity and the strong-rod-out reactivity. The all-rods-in reactivity is indicative of the overall core reactivity characteristics, while the maximum rod worth is indicative of local reactivity characteristics. These two components are shown in Figures SBWB-63-1 and SBWB-63-2 for the BFN Unit 1 Cycle 7 core operated at both 105% power and 120% power. As is evident in these results, the small changes in R-value discussed in RAI SBWB-59 are the results of equally small or smaller changes in the two components. Further, there are no discernable trends in these results. Each is at times greater for the 120% core and at times greater for the 105% core. This further supports the conclusion drawn in SBWB-59 that the void and control history impacts of operations at 105% and 120% power are negligible with

regard to SDM. Although the R-value is slightly smaller for the 105% power case at the point of minimum SDM, the 1.5% design margin for SDM is maintained for operation at 105% power.

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Figure SBWB-63-1

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Figure SBWB-63-2

NRC RAI SBWB-64

In general, small break loss-of-coolant accidents tend to be relatively insensitive to stored energy; therefore, fuel type does not influence the difference in limiting breaks on Units 1, 2, and 3. Explain why the 0.5 ft^2 break is limiting for Units 2 and 3 while the 0.06 ft^2 discharge break is limiting for Unit 1. Provide peak centerline temperatures (PCTs) for breaks in the range of 0.04 - 0.07 ft^2 for Units 2 and 3.

TVA Response to RAI SBWB-64

The EPU loss-of-coolant accident (LOCA) analyses for Unit 1 and Units 2/3 were performed by GE and Areva, respectively, utilizing their proprietary analytical models. The limiting pipe break size on Unit 1 is 0.06 ft^2 and was determined using GE SAFER/GESTR-LOCA modeling. The limiting break on Units 2 and 3 was analyzed to be 0.5 ft^2 and was determined using AREVA EXEM BWR-2000 LOCA methodology. The GE and Areva LOCA methodologies were previously submitted by the respective fuel vendor to NRC as Licensing Topical Reports for staff review and approval.

The following is an explanation from each fuel vendor describing the basis for the determination of the limiting break for Unit 1 and Units 2/3 using GE SAFER/GESTR-LOCA and AREVA EXEM BWR-2000 LOCA methodologies. Considerable margin to 10 CFR 50.46 peak cladding temperature (PCT) limits is maintained in both applications.

GE Input to RAI SBWB-64

Although GE is not in a position to discuss the Areva methodology and results, this response discusses the SAFER methodology, especially those parts that affect the limiting break size. This response focuses on the GE break spectrum results, particularly the differences between the 0.5 ft^2 and 0.07 ft^2 breaks. [[

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Table SBWB-64-1

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Figure SBWB-64-1
SAFER Nodalization

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Figure SBWB-64-2
Comparison of Total Break Flow for 0.5 ft^2 and 0.07 ft^2 Breaks

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Figure SBWB-64-3
Comparison of Total Mass Loss due to Breaks and ADS for 0.5 ft^2 and 0.07 ft^2 Breaks

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Figure SBWB-64-4
Comparison of Dome Pressure for 0.5 ft^2 and 0.07 ft^2 Breaks

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Figure SBWB-64-5
Hot and Average Bundle Levels for the Nominal 0.5 ft^2 Break with Battery Failure (GE13 at 3952 MWt)

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Figure SBWB-64-6
Heat Transfer Coefficients for the Nominal 0.5 ft^2
Break with Battery Failure (GE13 at 3952 MWt)

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Figure SBWB-64-7
Hot and Average Bundle Levels for the Nominal 0.07 ft^2
Break with Battery Failure (GE13 at 3952 MWt)

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Figure SBWB-64-8
Heat Transfer Coefficients for the Nominal 0.07 ft^2
Break with Battery Failure (GE13 at 3952 MWt)

Areva Input to RAI SBWB-64

It is correct that fuel type does not significantly influence the difference in limiting breaks between Units 1, 2, and 3. The difference in limiting break size is due to differences between the approved AREVA EXEM BWR-2000 LOCA methodology and the GE LOCA methodology. Areva does not have access to the details of GE LOCA methodology, thus it is not possible to quantify the differences in methods that lead to the different predicted limiting break size. However, the following is a detailed explanation providing physical arguments that describe why the 0.5 ft^2 break is correctly identified as being the break with limiting conditions for the AREVA EXEM BWR-2000 methodology for BFN Units 2 and 3.

Figure SBWB-64-9 presents the time and break size dependence for key recirculation loop discharge side break parameters from the Areva LOCA break spectrum analysis for Units 2 and 3. [[

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The analysis results also demonstrate:

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The PCTs for 0.05 ft^2 and 0.1 ft^2 breaks are presented in Table SBWB-64-2. The 0.05 ft^2 break was the only break size analyzed in the requested range of 0.04 - 0.07 ft^2. The 0.1 ft^2 break was the next smallest break analyzed and is included to bracket the break size range requested in the RAI.

Table SBWB-64-2
TLO Recirculation Line Break Spectrum
Results for 102% EPU 105% Flow SF-BATT

Break Size and Type	PCT (°F)	
	Pump Discharge	
	Mid-Peaked	Top-Peaked
0.1 ft^2	1569	1483
0.05 ft^2	1222	1235

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Figure SBWB-64-9
Comparison of Key Event Times Relative to PCT for Browns Ferry EPU LOCA Analyses

NRC RAI SBWB-65 (Units 2 and 3)

Provide the PCT for a top-peaked axial distribution for the 0.5 ft^2 break. Explain why a top-peaked axial power distribution is less limiting for the 0.5 ft^2 break for Units 2 and 3.

TVA Response to RAI SBWB-65 (Units 2 and 3)

Table SBWB-65-1 provides PCTs for both top- and mid-peaked axial power distributions for the 0.5 ft^2 recirculation pump discharge side break from the AREVA EXEM BWR-2000 break spectrum analysis for Units 2 and 3. Figures SBWB-65-1 and SBWB-65-2 show the temperature plot for the peak temperature node of each case. The plots show that the top-peaked axial power distribution case begins clad heatup earlier and heats up to a higher temperature during the early part of the blowdown period. This earlier heatup results from the earlier uncovery of the hot node near

the top of the core. However, when core spray begins injection at about 202 seconds, the hot node for the top-peaked case near the top of the core receives more effective cooling as the core spray penetrates from the top of the core downward.

For the mid-peaked case, the core spray is less effective at penetrating down to the middle of the core to provide enhanced cooling to the hot node. The enhanced cooling for the top-peaked hot node continues for about 75 seconds until the end of blowdown. After the end of blowdown, the EXEM BWR-2000 Appendix K method requires that credit be taken only for conservative spray cooling heat transfer coefficients until reflood occurs. Once this required conservative heat transfer is applied, both cases rapidly increase in clad temperature. However, because the top-peaked hot node received better cooling during the period of core spray prior to end of blowdown, it is at a lower initial temperature at the end of blowdown than the mid-peaked case. So, while the differential magnitude of heatup after the end of blowdown is similar between cases (as it should be) the top-peaked case does not ultimately achieve as high a clad temperature prior to reflood because it begins the heatup from a lower temperature point.

The limiting (highest PCT) recirculation line break for BFN Units 2 and 3 is the 0.5 ft^2 break in the pump discharge piping with a failure of a shutdown board battery as the single failure (SF-BATT), and a mid-peaked axial power shape at 102% of EPU and 105% rated core flow.

Table SBWB-65-1
TLO Recirculation Line Break Spectrum
Results for 102% EPU 105% Flow SF-BATT*

Break Size and Type	PCT (°F)	
	Pump Discharge	
	Mid-Peaked	Top-Peaked
0.5 ft^2 split	1998	1874

* Break spectrum results only. The limiting PCT for the limiting Browns Ferry ATRIUM-10 assembly is determined in the MAPLHGR analysis and is a few degrees higher. See the response to RAI SBWB-66.

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Figure SBWB-65-1
Cladding Temperatures for 0.5 ft^2/Pump Discharge
Mid-Peaked SF-BATT 102P/105F EPU
(Limiting Break)

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Figure SBWB-65-2
Cladding Temperatures for 0.5 ft^2/Pump Discharge
Top-Peaked SF-BATT 102P/105F EPU

NRC RAI SBWB-66 (Units 2 and 3)

Provide the plot and detailed results for the 0.5 ft^2 break from the June 12, 2006, submittal with the ANP-2541 enclosure. Ensure the two-phase level in the core region is included in the package.

TVA Response to RAI SBWB-66 (Units 2 and 3)

The approved AREVA EXEM BWR-2000 methodology does not track a two-phase level in the core; therefore, the requested plot of two-phase level in the core is not available. Attempts to define a level based on mixture void fraction would not be qualified or meaningful.

The limiting (highest PCT) recirculation line break for BFN Units 2 and 3 is the 0.5 ft^2 break in the pump discharge piping with a failure of a shutdown board battery as the single failure, and a mid-peaked axial power shape at 102% of EPU and 105% rated core flow. The break spectrum PCT is 1998°F as previously shown in Table SBWB-65-1.

This PCT of 1998°F, which was calculated in the break spectrum analysis, is used to determine the limiting break conditions in the EXEM BWR-2000 methodology. The identified limiting case is then evaluated in the burnup dependent maximum average planar linear heat generation rate (MAPLHGR) limit heatup analysis to determine the limiting licensing PCT. The limiting PCT for this case from the MAPLHGR limit heatup analysis is 2007°F.

A copy of ANP-2541, "Browns Ferry Unit 2 Cycle 15 Reload Analysis," was submitted to NRC on June 12, 2006, (ML061670151). This report includes a small 10 CFR 50.46 estimated PCT reduction of 5°F for a reportable PCT of 2002°F. This PCT value was also reported to NRC on November 7, 2006 (ML063110435), in the periodic BFN 10 CFR 50.46 report. ANP-2541 provides references to the Areva break spectrum report and MAPLHGR report discussed above.

Key event times for this limiting break are provided in Table SBWB-66-1. Figures SBWB-66-1 through SBWB-66-26 provide plots of key parameters from the RELAX system and hot channel blowdown analyses. A plot of cladding temperature versus time in the hot assembly from the HUXY heatup analysis is provided in Figure SBWB-66-27.

Table SBWB-66-1
Event Times for Limiting Two Loop Operation (TLO)
Recirculation Line Break 0.5 ft^2 Split Pump Discharge -
Single Failure-Battery (SF-BATT) Mid-Peaked Axial 102% EPU 105% Flow

Event	Time (sec)
Initiate break	0.0
Initiate scram	0.5
Low-low liquid level, L2 (448 in)	16.4
Low-low-low liquid level, L1 (372.5 in)	26.7
Jet pump uncovers	35.5
Core spray permissive for ADS timer	55.7
Recirculation suction uncovers	57.2
Core spray pump at rated speed	58.7
Lower plenum flashes	71.3
ADS valves open	175.7
Core spray valve pressure permissive	193.1
Core spray valve starts to open	195.1
Core spray high-pressure cutoff	201.4
Core spray flow starts	201.4
Recirculation discharge isolation valve (RDIV) pressure permissive	222.0
RDIV starts to close	224.0
Core spray valve fully open	228.1
RDIV fully closed	260.0
Rated core spray flow	277.2
Blowdown ends	277.2
Core reflood	358.3
PCT	358.3
Bypass reflood	421.4

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Figure SBWB-66-1
Limiting TLO Recirculation Line
Break Upper Plenum Pressure

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Figure SBWB-66-2
Limiting TLO Recirculation Line Break
Total Break Flow Rate

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Figure SBWB-66-28
Limiting TLO Recirculation Line Break
Core Inlet Flow Rate

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Figure SBWB-66-4
Limiting TLO Recirculation Line Break
Core Outlet Flow Rate

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Figure SBWB-66-5
Limiting TLO Recirculation Line Break
Intact Loop Jet Pump Drive Flow Rate

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Figure SBWB-66-6
Limiting TLO Recirculation Line Break
Intact Loop Jet Pump Suction Flow Rate

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Figure SBWB-66-7
Limiting TLO Recirculation Line Break
Intact Loop Jet Pump Exit Flow Rate

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Figure SBWB-66-8
Limiting TLO Recirculation Line Break
Broken Loop Jet Pump Drive Flow Rate

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Figure SBWB-66-9
Limiting TLO Recirculation Line Break
Broken Loop Jet Pump Suction Flow Rate

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Figure SBWB-66-10
Limiting TLO Recirculation Line Break
Broken Loop Jet Pump Exit Flow Rate

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Figure SBWB-66-11
Limiting TLO Recirculation Line Break
ADS Flow Rate

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Figure SBWB-66-12
Limiting TLO Recirculation Line Break
HPCI Flow Rate

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Figure SBWB-66-13
Limiting TLO Recirculation Line Break
Core Spray Flow Rate

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Figure SBWB-66-14
Limiting TLO Recirculation Line Break
Intact Loop LPCI Flow Rate

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Figure SBWB-66-15
Limiting TLO Recirculation Line Break
Broken Loop LPCI Flow Rate

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Figure SBWB-66-16
Limiting TLO Recirculation Line Break
Upper Downcomer Mixture Level

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Figure SBWB-66-17
Limiting TLO Recirculation Line Break
Lower Downcomer Mixture Level

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Figure SBWB-66-18
Limiting TLO Recirculation Line Break
Lower Downcomer Liquid Mass

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Figure SBWB-66-19
Limiting TLO Recirculation Line Break
Intact Loop Discharge Line Liquid Mass

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Figure SBWB-66-20
Limiting TLO Recirculation Line Break
Upper Plenum Liquid Mass

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Figure SBWB-66-21
Limiting TLO Recirculation Line Break
Lower Plenum Liquid Mass

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Figure SBWB-66-22
Limiting TLO Recirculation Line Break
Hot Channel Inlet Flow Rate

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Figure SBWB-66-23
Limiting TLO Recirculation Line Break
Hot Channel Outlet Flow Rate

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Figure SBWB-66-24
Limiting TLO Recirculation Line Break
Hot Channel Coolant Temperature at the Limiting Node

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Figure SBWB-66-25
Limiting TLO Recirculation Line Break
Hot Channel Quality at the Limiting Node

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Figure SBWB-66-26
Limiting TLO Recirculation Line Break
Hot Channel Heat Transfer Coefficient at the Limiting Node

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Figure SBWB-66-27
Limiting TLO Recirculation Line Break
Cladding Temperatures

NRC RAI EEMB-118 (Unit 1)

In the July 26, 2006 response, the licensee indicated that Unit 1 is currently performing restart modifications and that the final stress analysis results, which reflect the as-built configuration, are not available for most of the reactor coolant pressure boundary and balance-of-plant systems. Provide the schedule for completion of the piping system evaluation for Unit 1. Upon completion, provide the evaluation summary for piping systems and their supports including main steam, feedwater, recirculation, residual heat removal, and torus-attached piping systems. The information should include the calculated maximum stresses and fatigue usage factors, as necessary, for piping systems and their supports similar to those provided for the Units 2 and 3 extended power uprate (EPU) evaluation.

TVA Response to RAI EEMB-118 (Unit 1)

TVA originally replied to RAI-10 question EEMB-118 by letter dated October 5, 2006 (ML062860267) and supplemented its response by letter dated December 21, 2006, which included an evaluation summary for Unit 1 piping systems. The evaluation summary included calculated maximum stresses for piping systems similar to the information provided for the EPU application for BFN Units 2 and 3 in TVA's July 26, 2006, submittal (ML062200277).

The following additional information is provided in response to a verbal request by the NRC staff.

The effects due to the Unit 1 EPU are included with other effects due to other changes in the analyses. That is, the pipe stress/pipe support analyses performed to support the Unit 1 restart effort included the effects of the revised seismic response spectra, NRC IE Bulletins 79-02 and 79-14 requirements, and EPU effects. Therefore, it is not possible to discretely identify the effects due solely to EPU on the Unit 1 piping/supports.

As an alternative, TVA proposed to use knowledge gained from Units 2/3 to help answer this question. The Units 2/3 analyses for the response spectra change and NRC IE Bulletins 79-02 and 79-14 were performed as part of the restart efforts for those units. Later, as part of the Units 2/3 EPU project, the impact of EPU was evaluated in the calculations. Thus, the Units 2/3 analyses were sequential, such that the individual effects could be seen as opposed to the Unit 1 evaluation where all factors were included at the same time.

The Units 2/3 EPU analyses identified one pipe support on one system that was overstressed and required modification. Due to the similarity between Units 1, 2 and 3, it would be expected that the same pipe support in Unit 1 would have been similarly affected if analyzed sequentially.

The Unit 2 support modified for EPU is Main Steam snubber support 2-47B400S0110. The corresponding Unit 1 support is identified as support 1-47B400-75. The maximum calculated values for the Unit 1 support and the associated allowables are shown in Table EEMB-118-36.

Table EEMB-118-36

Item	Maximum Calculated Value	Allowable Limit
Catalog items (snubber, end brackets)	18.122 kips	20.0 kips
Structural Members	12.147 ksi	19.17 ksi
Welds	15.44 ksi	15.66 ksi

NRC RAI EEMB-119 (Unit 1)

In TVA response to RAI EEMB.39 [dated July 26, 2006], the steam separators at Units 2 and 3 were evaluated for the EPU condition, since they are the most critical component next to the steam dryers affected by the increase steam flow within the reactor vessel. Confirm whether and how the identical quantitative evaluation can be applied for the Unit 1 steam separator at the EPU operation. If not, provide a summary evaluation for the Unit 1 separators.

TVA Response to RAI EEMB-119 (Unit 1)

The separators at BFN Unit 1 were evaluated identically as the separators at BFN Units 2 and 3. The separators are identical and were evaluated to the same conditions.

NRC RAI EEMB-120 (Unit 1)

In reference to EEMB.42 of July 26, 2006, transmittal, provide a summary evaluation of recirculation pumps and their supports at the EPU condition. The requested information should include the calculated stresses and cumulative usage factors (CUFs) for the critical components of the recirculation pump in comparison with the allowable limits. Also, provide a discussion of the potential for vane passing frequency vibration at the EPU condition at Unit 1.

TVA Response to RAI EEMB-120 (Unit 1)

A summary evaluation of recirculation pumps and their supports at the EPU condition is provided below. The information provided includes the maximum EPU stress, allowable stress, and ratio of "EPU Stress/Allowable Stress." The design basis code of record for BFN is the USAS B31.1-1967 code; consequently, cumulative usage factors (CUFs) have not been calculated.

RECIRCULATION PUMP NOZZLES

A summary of the highest stressed nozzle for the recirculation pump nozzles is presented in Table EEMB-120-1.

The pipe stress equations (for normal, upset, and faulted conditions) used in this determination are as defined in General Design Criteria No. BFN-50-C-7103, "Structural Analysis and Qualification of Mechanical and Electrical Systems (Piping and Instrument Tubing)."

Pipe stresses were reviewed for each of the above defined equations for all nozzles (suction and discharge) and the maximum stress selected.

Allowable stress is 0.9*(Yield stress of pump nozzle) from BFN FSAR Appendix-C, Table C.4-2, "Primary System Components - Critical Load Combinations, Locations, and Allowables (Recirculation Pumps)".

As shown in the Table EEMB-120-1, the maximum stress for all nozzles is less than the applicable allowable stress.

Table EEMB-120-1

Maximum EPU Stress (psi)	Allowable Stress (psi)	Ratio EPU Stress/Allowable Stress
11,114	21,708	0.512

RECIRCULATION PUMP SUPPORTS

A summary of only the maximum controlling value (i.e., stress, load, interaction ratio (IR)) for each of the recirculation pump supports is presented in Table EEMB-120-2.

Table EEMB-120-2

Support Number	Maximum EPU - Stress (ksi) - Load (kip) - IR	Allowable - Stress (ksi) - Load (kip) - IR	Ratio	Comment
1-47B465-497	26.341 kip	66.6 kip	0.4	standard component
1-47B465-498	50.972 kip	66.6 kip	0.77	standard component
1-47B465-499	10.405 ksi	11.076 ksi	0.94	weld
1-47B465-500	0.978	1.0	0.978	structural member combine stress IR per AISC
1-47B465-501	19.482 ksi	23.925 ksi	0.814	structural member bending stress
1-47B465-502	0.975	1.0	0.975	structural member combine stress IR per AISC
1-47B465-503	15.505 ksi	16.588 ksi	0.93	weld
1-47B465-504	11.267 ksi	16.588 ksi	0.679	weld
1-47B465-505	0.5	1.0	0.5	structural member combine stress IR per AISC
1-47B465-513	59.600 kip	66.6 kip	0.895	standard component
1-47B465-514	12.225 ksi	16.588 ksi	0.74	weld
1-47B465-515	49.389 kip	66.6 kip	0.742	standard component
1-47B465-516	18.028 ksi	18.678 ksi	0.965	structural member bending stress
1-47B465-517	25.700 kip	27.200 kip	0.945	standard component
1-47B465-518	10.498 ksi	12.240 ksi	0.858	weld
1-47B465-519	13.677 ksi	16.588 ksi	0.824	weld
1-47B465-520	12.480 ksi	16.359 ksi	0.76	weld
1-47B465-521	14.673 ksi	16.360 ksi	0.9	weld

VIBRATION

As discussed in the response to RAI EEMB.42 (Units 2 and 3) in TVA's July 26, 2006, submittal (ML062200277), vibration monitoring of the BFN Unit 2 reactor recirculation piping including monitoring at the pump impeller vane passing

frequencies has been performed. Analyses indicate that piping stresses would be acceptable for the higher pump speeds expected for EPU conditions. Similarly, vibration monitoring of the Unit 1 recirculation piping is planned during startup testing.

NRC RAI EEMB-89 (Unit 2)

In the response to EEMB.40 of the July 26, 2006, submittal, Footnote 5 indicated that the current design basis stress shown above corresponds to the material stress allowable for this equation of 40,500 pounds per square inch (psi), while the EPU stress corresponds to a location with a material stress allowable of 36,000 psi. Provide a summary of evaluation pertaining to the calculated stress for the EPU condition, which is almost equal to the allowable limit. Also specify the component material and the calculated stresses at these two maximum stress locations for current and EPU conditions for the Unit 2 feedwater loop A.

TVA Response to RAI EEMB-89 (Unit 2)

The Unit 2 Feedwater Loop A piping was reanalyzed to incorporate EPU changes to the thermal modes.

The EPU analysis increased the thermal stresses at data point SB20, causing this location to replace data point 47 as having the highest Equation 9U + 10 stress ratio. The pipe material at data point 47 is SA106, Grade B carbon steel. The material at data point SB20 is SA106, Grade A, which has a lower allowable stress. Data point 47 is on the end of the last elbow on the 12" feedwater injection line just before it taps into the 24" feedwater header. Data point SB20 is on a 1/2" line off valve 2-HCV-3-67.

The highest stressed locations for both the current design basis and EPU conditions are included in Table EEMB-89-1. Pipe stresses for the Feedwater piping are evaluated using the load combination criteria established in Table 8.0-1 of BFN-50-C-7103. The Code equations (Equation 9U, Equation 10 and Equation 11) are established using Section III of the 1971 Edition of the ASME Boiler and Pressure Vessel Code, including Summer 1973 Addenda. The Equation 9U + 10 stress combination includes stresses due to design pressure, sustained weight loading, thermal expansion, thermal anchor movement, operating basis earthquake (OBE) and OBE seismic anchor movements. Equation 11 includes stresses due to design pressure, sustained weight, thermal expansion and thermal anchor movement.

Table EEMB-89-1

	Location (Data Pt)	Material	Equation	Stress (psi)	Allowable (psi)
Current DB Max Stress	47	SA106 GR B	9U+10	40,203	40,500 (see note)
	SB20	SA106 GR A	9U+10	34,479	36,000
	SB20		11	28,581	30,000
EPU Max Stress	47	SA106 GR B	9U+10	42,419	45,000
	SB20	SA106 GR A	9U+10	35,924	36,000
	SB20		11	29,807	30,000

Note: The current design basis pipe stress calculations conservatively used an allowable stress of 40,500 psi for Equation 9U + 10. This is the allowable stress that was reported in the TVA response to RAI EEMB-40 (July 26, 2006, submittal). This value is lower than the actual 9U + 10 allowable defined in Table 8.0-1 of BFN-50-C-7103 and the UFSAR which is 45,000 psi for SA106 Gr B carbon steel. When these calculations were revised to address EPU conditions, the stress combination was compared to the 45,000 psi allowable reported for the EPU stresses in the table above.

NRC RAI EEMB-121/90 (Units 1, 2, and 3)

In reference to Section 3.4 of NEDC-33101 P, (a) confirm whether CUFs given in Table 3-4 are evaluated for 60 years, since the plant's current licensing basis includes the 20 years of life extension at the EPU condition. If so, explain why the table indicates the evaluation was done for a 40-year life of EPU operation; (b) the CUF for the feedwater nozzle at the EPU condition is less than that for the current operating condition. Confirm whether the calculation has taken account for the increase of thermal transient due to the increase in flow at the nozzle for the EPU. Also, provide a summary of calculation for the CUF for Units 2 and 3 at feedwater nozzle, which is almost equal to 1.0 for the EPU operation; (c) the CUF for the support skirt is very small in comparison with that of the current CUF. Provide the summary of calculation and the technical basis to

show that the peak stress and the number of cycles are greatly reduced for the EPU operation at the EPU condition for the Unit 1 support skirt. Also, confirm whether and how the calculated CUFs for the critical components such as main closure studs, support skirt and recirculation outlet nozzle remain unchanged for the EPU condition.

TVA Response to RAI EEMB-121/90 (Units 1, 2, and 3)

Part (a)

The CUF values given in Table 3-4 of the Unit 1 PUSAR (NEDC-33101P) and Tables 3-2a and 3-2b of the Units 2 and 3 PUSAR (NEDC-33047P) were evaluated at 40 years. Comparison of the current CUFs and EPU CUFs in these tables provides the relative differences associated with EPU only.

The fatigue usage factors for the limiting components evaluated at EPU conditions and 60 years were previously provided to the NRC in Table 4.3.1.1 of the Application for Renewed Operating Licenses for BFN Units 1, 2, and 3 dated December 31, 2003 (ML040060359).

Part (b)

Yes, increases in flow and temperature have been included. Note that the stress has increased for the feedwater nozzle as shown in Table 3-4. The CUF was evaluated to reduce conservatism by considering plant-specific cycles and a seal refurbishment at 18 years of operation and a 12-year seal refurbishment cycle after 18 years (to reduce the fatigue contribution due to rapid cycling).

The 40-year license is defined as 18 years of operation at pre-EPU conditions, and 22 years at EPU operating conditions. Rapid cycling is included in fatigue evaluations for the Feedwater Nozzles. Refurbishing the seals for the nozzle/safe end component can mitigate the effects of rapid cycling.

Blend Radius

The blend radius is the limiting CUF location for the feedwater nozzle. Calculations were performed to determine that the optimum seal refurbishment period is after 18 years of operation at pre-EPU conditions, with another refurbishment 12 years into operation at EPU conditions (i.e., after 18 and 30 years of plant operation). The rapid cycling contribution for this location was determined to be 0.8423 for the 18 years of pre-EPU operation; 0.0102

for the first 12 years of EPU operation (assuming refurbishment at 18 years); 0.0041 for the remaining 10 years of EPU operation (assuming a second refurbishment at 30 years), thus accounting for the total 40-year license. System cycling usage was determined to be 0.14. Therefore, the total of system plus rapid cycling CUF = 0.14 + 0.8423 + 0.0106 + 0.0041 = 0.997, at the blend radius. It can be seen that the largest contributor to the CUF is the rapid cycling. Therefore, if there is no leakage contributing to rapid cycling fatigue, the system cycling fatigue is very small and the nozzle will remain qualified. The blend radius is managed by an inspection program.

Safe End

For the safe end CUF, the rapid cycling was determined to be 0.1545 for the 18 years of pre-EPU operation; 0.1344 for the first 12 years of EPU operation (assuming refurbishment at 18 years); 0.0087 for the remaining 10 years of EPU operation (assuming a second refurbishment at 30 years), thus accounting for the total 40-year license. System cycling was determined to be 0.6777. Therefore, the total of system plus rapid cycling CUF = 0.6777 + 0.2976 = 0.973. The nozzle blend radius is therefore expected to show evidence of fatigue before the safe end. Since the feedwater nozzle blend radius CUF is managed with an inspection program, the safe end region is also managed (assuming that no fatigue failure is seen at the blend radius).

Part (c)

In the BFN Unit 2/3 evaluation, upon which the Unit 1 evaluation is partially based, the OLTP stresses were scaled by an uprate factor of 1.044. The CUF value was based upon two transients: (1) Loss of feedwater pump (LOFWP) alternating stress of 254.6 ksi, corresponding to 53 allowable cycles and 10 actual cycles (U = 0.189) plus (2) Heatup/cooldown alternating stress of 159.0 ksi, corresponding to 171 allowable cycles and 122 actual cycles (U = 0.715). The current CUF is therefore 0.189 + 0.715 = 0.904.

For the BFN Unit 1 analysis, a finite element analysis was performed in order to reduce conservatism for the support skirt startup/shutdown (SU/SD) and loss of feedwater pump transients. The mechanical loads were obtained from the OLTP B&W stress reports. Also, the number of actual cycles used in the BFN Unit 1 analysis was based upon cycle

counting. The SU/SD case considered 122 cycles in the Units 2/3 evaluation; the Unit 1 evaluation considered 78 cycles. The results are provided in Table EEMB-121/90-1.

Table EEMB-121/90-1
(40-Year Life)

	Stress Intensity, psi			Cycles		
Transient	Stress Range, psi	Alt Stress, psi	Alt Stress * Modulus Correction * Ke, psi	Actual Number of Cycles, n	Allowable Number of Cycles, N	Fatigue Usage, U = n/N
LOFWP+OBE+JET	131081	65541	108812	1	210	0.005
LOFWP+OBE	107259	53630	89035	4	800	0.005
SU/SD	79014	39507	65588	78	1700	0.046
LOFWP	84293	42147	69971	12	1600	0.008
OBE	38249	19125	31751	45	20000	0.002
Total						0.066

For the Units 2/3 recirculation outlet nozzle, support skirt, and main closure studs, the operating parameters (flow, temperature, pressure) did not increase for EPU operating conditions compared to CLTP; therefore, the EPU stress and CUF (recirculation outlet nozzle = 0.779, support skirt = 0.904, and main closure studs = 0.762) remain unchanged from the current values.

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)

RESPONSE TO ROUND 12 REQUEST FOR ADDITIONAL INFORMATION

GE AND AREVA AFFIDAVITS

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 to GE letter GE-ER1-AEP-07-358, Larry King (GE) to J. Valente (TVA), *GE Responses to NRC Requests for Additional Information – EEMB-119, EEMB-120, EEMB-121/90, SBWB-57, SBWB-58, SBWB-59, SBWB-63, and SBWB-64,* dated February 22, 2007. The proprietary information in Enclosure 1, *GE Responses to NRC Request for Additional Information – EEMB-119, EEMB-120, EEMB-121/90, SBWB-57, SBWB-58, SBWB-59, SBWB-63, and SBWB-64*, is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a, and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions from evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for the power uprate of a GE BWR, utilizing analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of and applied to perform evaluations of the transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the underlying evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In

addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 22nd day of February 2007.

George B. Stramback

George B. Stramback
General Electric Company

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the Engineering Information Record, 51-9043400-000, "Response to NRC RAI – Round 12 for Browns Ferry EPU: SBWB-64, SBWB-65, SBWB-66," dated February 2007, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 23rd
day of February, 2007.

Danita R. Kidd
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/08

ENCLOSURE 4

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 12 REQUEST FOR ADDITIONAL INFORMATION

LIST OF REGULATORY COMMITMENTS

One regulatory commitment was made in the response to RAI SBWB-60:

A Core Operating Limits Report (COLR) for the initial 105% power operation of Unit 1 will be submitted to NRC prior to unit restart.