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

March 6, 2008

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

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) CHANGE TS-418 AND TS-431 - EXTENDED POWER UPRATE (EPU) - RESPONSE TO ROUND 16 REQUEST FOR ADDITIONAL INFORMATION (RAI) - SRXB-74/86 AND SRXB-87 THROUGH SRXB-90 (TAC NOS. MD5262, MD5263, AND MD5264)

By letters dated June 28, 2004 and June 25, 2004 (ADAMS Accession Nos. ML041840109 and ML041840301), TVA submitted license amendment applications to the NRC for EPU operation of BFN Unit 1 and BFN Units 2 and 3, respectively. The pending EPU amendments would increase the maximum authorized power level for all three units by approximately 14 percent from 3458 megawatts thermal (MWt) to 3952 MWt.

On February 6, 2008, the NRC staff issued a Round 16 RAI (ML080370225) regarding the BFN Unit 1 and BFN Units 2 and 3 license amendment requests. Enclosure 1 to this letter provides TVA's response to the Round 16 RAI questions SRXB-74/86 and SRXB-87 through SRXB-90. Round 16 RAI question SRXB-73 was answered on February 21, 2008. Round 16 RAI EMEB-167/134, which is a steam dryer question, will be

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answered by March 31, 2008, in the RAI Round 15 Group 3 steam dryer response.

Enclosure 1 is a proprietary response to the RAI and contains information that AREVA NP, Inc.(AREVA) considers to be proprietary in nature and subsequently, pursuant to 10 CFR 9.17(a)(4), 2.390(a)(4) and 2.390(d)(1), AREVA requests that such information 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 an affidavit from 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).

No new regulatory commitments are made in this submittal. If you have any questions regarding this letter, please contact Tony Langley at (256)729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 6th day of March, 2008.

Sincerely,

S. M. Douglas

S. M. Douglas
Interim Site Vice President

Enclosures:

1. Response to Round 16 Request for Additional Information SRXB-74/86 and SRXB-87 through SRXB-90 (Proprietary Information Version)
2. Response to Round 16 Request for Additional Information SRXB-74/86 and SRXB-87 through SRXB-90 (Non-proprietary Information Version)
3. AREVA Affidavit

Enclosures:
cc (w/o 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-418 AND TS-431
EXTENDED POWER UPRATE (EPU)

RESPONSE TO ROUND 16 REQUEST FOR ADDITIONAL INFORMATION
SRXB-74/86 AND SRXB-87 THROUGH SRXB-90

(NON-PROPRIETARY INFORMATION VERSION)

This enclosure provides TVA's response to NRC's February 6, 2008, Round 16 Request for Additional Information (RAI)(ADAMS Accession No. ML080370225) questions SRXB-74/86 and SRXB-87 through SRXB-90.

NRC RAI SRXB-74/86 (Unit 1/Units 2 and 3)

Pellet clad interaction (PCI) and stress corrosion cracking (SCC) phenomena can cause clad perforation resulting in leaking fuel bundles and resultant increased reactor coolant activity. Therefore, the staff requests the licensee to provide the following additional information regarding PCI/SCC for Units 1, 2, and 3 at EPU conditions:

a. Describe any differences in operating procedures associated with PCI/SSC at EPU conditions versus pre-EPU operations.

b. From the standpoint of PCI/SCC, discuss which of the Anticipated Operational Occurrences (AOOs), if not mitigated, would most affect operational limitations associated with PCI/SSC.

c. For the AOOs in part b), discuss the differences between the type of required operator action, if any, and the time to take mitigating actions between pre-EPU and EPU operations.

d. If the EPU core will include fuels with non-barrier cladding which have less built-in PCI resistance, then demonstrate by plant-specific analyses that the peak clad stresses at EPU conditions will be comparable to those calculated for the current operating conditions.

e. Describe operator training on PCI/SCC operating guidelines.

TVA Response to SRXB-74/86 (Unit 1/Units 2 and 3)

a. A cycle-specific report "Core Design and Operating Restrictions to Reduce PCI Fuel Failure Probability" is prepared by the TVA Nuclear Fuels group, which includes details of the core design, fuel conditioning restrictions, deconditioning parameters, operating guidance for power changes and monitoring, and in the case of failed fuel, guidance for suppressing leaking bundles. The PCI operating guidelines from this report are incorporated into Appendix T of plant procedure 0-TI-248, "Station Reactor Engineer," which is the primary plant document used for establishing and monitoring PCI limitations. PCI operating restrictions are based on each specific type of fuel assembly and do not vary based on whether the operating cycle is non-EPU or EPU.

b. The analyzed AOO, which if not mitigated, would most affect operational limitations associated with PCI/SCC is the Loss of Feedwater Heater (LFHW) event since the transient involves a global increase in core power, which would result in a large number of fuel nodes exceeding their preconditioned envelope. A feedwater heater can be lost if the steam extraction line to a heater isolates, which results in the heat supply to the heater being removed, producing a gradual cooling of the feedwater. The reactor vessel receives cooler feedwater, which results in an increase in core inlet subcooling and an increase in core power, a change in power distribution, and a decrease in bundle Critical Power Ratio (CPR).

The LFWH is an analyzed Updated Final Safety Analysis Report (UFSAR) transient and is evaluated as part of the reload licensing process to determine the impact on CPR and on the design basis fuel thermal and mechanical design limits. The UFSAR LFWH analysis assumes a decrease in feedwater temperature of 100°F resulting from the loss of a heater string.

c. The plant has a set of abnormal operating instructions (AOIs) for the loss of combinations of high pressure and low pressure feedwater heaters. On the isolation of a heater, the first step in the each of the AOI procedures is for the operator to reduce thermal power to 5% below the initial power level. So if the plant was operating at 100% power, the first operator action would be to reduce power to 95%. The subject AOIs also have a Caution Statement that the failure to reduce power, if the fuel was operating near or at the preconditioning envelope in any region of the core, could result in fuel damage. After the power reduction, the procedure specifies that the operator will adjust reactor power and flow to stay within thermal limits as directed by the Reactor Engineer or Unit Supervisor.

Recent plant operating experience was reviewed for cases where there was a loss of feedwater heaters to determine operator compliance with subject AOIs. There were two such events involving the unexpected isolation of a feedwater heater in the last two years. On June 17, 2007, Unit 2 experienced a loss of extraction steam on low pressure heater C3 and on July 22, 2006, Unit 3 had a loss of low pressure heater A3. The loss of a single number 3 heater has very little effect on feedwater temperature and reactor power. In both events, operators entered the correct AOI and reduced thermal power within about 3 1/2 minutes and 2 1/2 minutes, respectively, by reducing reactor recirculation system pump flow.

Plant response to a LFWH event would be similar for both current licensed power levels and for EPU. To minimize the impact of such an event on the fuel due to PCI/SCC, the appropriate operator action would be to promptly reduce power to within the preconditioning envelope. Therefore, the instructions in the AOIs to promptly reduce power to 5% below the initial power are the correct operator actions to take for non-EPU or EPU operation. Following the power reduction, the process computer core analysis programs would be used to check core thermal limits and to confirm the effectiveness of the power reduction with regard to the preconditioning envelope. There is high confidence based on operating experience and training that the operator would respond in accordance with the loss of feedwater heaters AOIs.

d. Currently, Unit 1 has an EPU-capable core comprised of 600 GE14 and 164 GE13 fuel assemblies. Unit 2 has an EPU-capable core consisting 653 ATRIUM-10 assemblies and 111 GE14 assemblies. The Unit 3 core is rated at current licensed power and consists of 169 GE14 and 595 ATRIUM-10 assemblies. The GE fuel types are all barrier fuel and the ATRIUM-10 fuel assemblies are all non-barrier. Starting with the Spring 2009 Unit 2 refuel outage, replacement ATRIUM-10 fuel assemblies will also be barrier fuel.

XEDOR is a tool for power maneuvering guidance currently under development by AREVA. It contains a reduced order stress model based on AREVA's fuel performance code RODEX4 and has been incorporated in MICROBURN-B2 with pin power reconstruction. The analysis is applied to every node of every rod in the core so that clad hoop stresses are calculated based upon time variations of power and fast neutron flux from the MICROBURN-B2 solution. The models are currently under evaluation by EPRI as part of the Zero Failures by 2010 Initiative (with Anatech code FALCON). The XEDOR models were presented at the Top Fuel Meeting in San Francisco (October 2007) and recently at an EPRI PCI guideline meeting in St. Petersburg, Florida (February 2008).

Various licensing basis AOO's were evaluated to assess the impact of PCI/SCC. The primary event of concern as discussed in b. is the LFWH event due to the core wide increase in power, which has an impact on all of the rods in the core. To address the NRC RAI, the peak clad stress for a licensing basis LFWH event has been analyzed with XEDOR for BFN Unit 3 Cycle 14, which is the next Unit 3 operating cycle (pre-EPU) core design. The XEDOR analysis was also performed for an equilibrium EPU cycle. Pre-EPU and EPU analyses were performed to provide a comparison of fraction of rods experiencing high levels of clad stress.

The analysis ignores any reduction in power for a control blade position change due to cycle startup, control blade adjustments, and control blade sequence exchanges by allowing the resultant clad stresses to relax for 40 hours in order to simulate the clad stresses that would actually be experienced in a realistic sequence exchange. Another depletion of one week is added to allow the clad stress relaxation to reach an equilibrium point. This latter point represents the conditions in the reactor for 95% of the operating time. The former point is to represent the worst anticipated conditions. These pairs of conditions are repeated for each of the planned sequence exchanges.

The LFWH event is modeled to occur at each of the points described above, which increase the inlet subcooling with a corresponding increase in core power by nearly 12% for both the pre-EPU and EPU cases. The primary purpose of this analysis is to evaluate the relative response of the pre-EPU core and the EPU core design. The results are compared in Figures SRXB-86.1 through SRXB-86.14. These histogram plots show the percentage of peak clad stresses in various stress ranges after the LFWH event before and after clad stress relaxation for various cycle exposures. Cycle exposure points are approximate for the two cycle designs since the timing for control rod sequence exchanges is different.

The plots were designed to show comparable conditions corresponding to a control rod sequence exchange. The figures only display the percentage of the rods for clad stresses above 75 megapascals (MPa) (the first pair of bars represent the population of fuel rods with a peak clad stress between 75 and 100 MPa) in order to improve the resolution of the data at the high stress end of interest. These plots indicate that there is minimal probability of significant failure due to PCI/SCC phenomena, which is considered likely above 400 MPa if the clad stress is maintained for a long time (on the order of one hour). Differences in core state-points (as shown by the cycle exposure) driven primarily by the control rod pattern show much larger variation than differences due to the core power level. Some cases show more fuel rods with high clad stresses for the EPU conditions, while other cases show more fuel rods with high clad stress for the pre-EPU conditions. In most cases the EPU state-points demonstrate slightly larger percentages of rods above high clad stress threshold values than those for the current pre-EPU cycle state-point. The maximum clad stress value calculated for the pre-EPU cycle was [ ]. The corresponding value for the EPU cycle is [ ]. When considering the fully relaxed condition, which represents the bulk of operation time, the clad maximum stress value for the pre-EPU cycle was calculated to be [ ] and [ ] for the EPU cycle.

This analysis demonstrates that the peak clad stresses at EPU conditions are comparable to those for current operation. It also demonstrates that the fraction of fuel experiencing high clad stresses that would likely cause failures during an anticipated LFWH, even if unmitigated, are very small. These results are consistent with previous analyses performed for power uprate in other plants.

Figure SRXB-86.1 BFN Fuel Rod Clad Maximum Stress Analysis
0 MWd/MTU (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.2 BFN Fuel Rod Clad Maximum Stress Analysis
0 MWd/MTU (LFWH Occurs After Stress Relaxation)

Figure SRXB-86.3 BFN Fuel Rod Clad Maximum Stress Analysis 3500 MWd/MTU (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.4 BFN Fuel Rod Clad Maximum Stress Analysis 3500 MWd/MTU (LFWH Occurs After Stress Relaxation)

Figure SRXB-86.5 BFN Fuel Rod Clad Maximum Stress Analysis
6000 MWd/MTU (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.6 BFN Fuel Rod Clad Maximum Stress Analysis
6000 MWd/MTU (LFWH Occurs After Stress Relaxation)

Figure SRXB-86.7 BFN Fuel Rod Clad Maximum Stress Analysis 7500 MWd/MTU (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.8 BFN Fuel Rod Clad Maximum Stress Analysis 7500 MWd/MTU (LFWH Occurs After Stress Relaxation)

Figure SRXB-86.9 BFN Fuel Rod Clad Maximum Stress Analysis
10000 MWd/MTU (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.10 BFN Fuel Rod Clad Maximum Stress Analysis
10000 MWd/MTU (LFWH Occurs After Stress Relaxation)

Figure SRXB-86.11 BFN Fuel Rod Clad Maximum Stress Analysis 13000 MWd/MTU (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.12 BFN Fuel Rod Clad Maximum Stress Analysis 13000 MWd/MTU (LFWH Occurs After Stress Relaxation)

Figure SRXB-86.13 BFN Fuel Rod Clad Maximum Stress Analysis Last Sequence Exchange (LFWH Occurs Before Stress Relaxation)

Figure SRXB-86.14 BFN Fuel Rod Clad Maximum Stress Analysis Last Sequence Exchange (LFWH Occurs After Stress Relaxation)

e. Operator training on fuel limits is provided for licensed Reactor Operators (RO) and Senior Reactor Operator (SRO) candidates in initial training as part of Generic Fundamentals, Core Thermal Limits. In this training segment, fuel preconditioning limitations and their role in operations is taught in a manner similar to that for Technical Specifications core limits. The following objectives regarding fuel preconditioning are covered:

- Describe PCI
- List the causes of PCI
- Describe the purpose of the pellet-to-clad gap
- Identify the possible effects of fuel densification
- Describe the effects of iodine and cadmium on PCI
- Explain the purpose of preconditioning operating recommendations
- Identify how the preconditioning rules minimize the adverse effects of PCI

In addition, a cycle-specific core design lesson plan is developed using the "Core Design and Operating Restrictions to Reduce PCI Fuel Failure Probability" report referenced in response a. above. Material in the lesson plan includes specific thresholds at which preconditioning should be done and ramp rates at which to precondition. This core design and preconditioning training is provided in the licensed operator requalification training program, which is attended by both RO and SRO personnel. Cycle-specific core design lesson plans are taught prior to each unit's refuel outage in preparation for the next cycle of operation.

NRC RAI SRXB-87 (Units 2 and 3 only)

To address the adequacy of benchmark data associated with neutronic power prediction methods, the staff understands that the issue was addressed by the fuel vendor by increasing the power distribution uncertainties and propagating them into the safety limit minimum critical power ratio calculation. Provide the following additional information:

a. Discuss the applicability of this approach to projected Units 2 and 3 operations using ATRIUM-10 fuel and AREVA methodologies.

b. Justify the use of the local and radial power distribution uncertainties based on Quad Cities gamma scans in light of the harder neutron spectrum present in EPU cores.

TVA Response to SRXB-87 (Units 2 and 3 only)

The methodology described in Reference SRXB-87.1 calculates radial bundle power uncertainty ($\delta P'_{ij}$) from separately determined uncertainty components. Three uncertainty components used to calculate $\delta P'_{ij}$ are:

- the deviation between the CASMO-4/MICROBURN-B2 (C4/MB2) calculated radial TIP response and the measured radial TIP response ($\delta T'_{ij}$),
- radial TIP measurement uncertainty (δT^{m}_{ij}), and
- radial synthesis uncertainty (δS_{ij})

These uncertainty components are determined using traversing incore probe (TIP) measurements, which are taken at or near full power conditions for Local Power Range Monitor (LPRM) calibration.

The BFN specific value of $\delta T'_{ij}$ was calculated in accordance with the Reference SRXB-87.1 methodology using BFN gamma TIP measurements and is **[ ]**. BFN is a D-Lattice plant. For comparison, Reference SRXB-87.1 reports a $\delta T'_{ij}$ of **[ ]** for D-Lattice plants.

The BFN specific $\delta T'_{ij}$ database is shown versus cycle number in Figure SRXB-87.1, versus power to flow ratio in Figure SRXB-87.2, and versus core void in Figure SRXB-87.3. Figures SRXB-87.1 and SRXB-87.2 represent the same data. The database includes 98 full core gamma TIP measurements: 46 for Unit 2 Cycles 13 through 15 (through February 2008), and 52 for Unit 3 Cycles 11 through 13 (to September 2007). Figure SRXB-87.3 represents the database consisting of Unit 2 Cycles 14 and 15, and Unit 3, Cycles 12 and 13. Void fraction data for Unit 2 Cycle 13 and Unit 3 Cycle 11 was not readily available.

Figures SRXB-87.1 through SRXB-87.3 clearly demonstrate that the D-lattice radial TIP uncertainty reported in the Reference SRXB-87.1 topical report is very conservative for BFN. Figures SRXB-87.1 through SRXB-87.3 also clearly demonstrate there is no correlation in the BFN specific uncertainty component due to the core power to flow ratio, or core average void fraction. Operation at the maximum core power and minimum core flow conditions allowed for EPU operations corresponds to a power to flow ratio of 38.95 MW-th/Mlb/hr, which is within the range of the data already taken.

The δT^{m}_{ij} is comprised of random instrument error and geometric measurement uncertainty caused by variations in the physical TIP location. A BFN specific radial TIP measurement uncertainty (δT^{m}_{ij}) was calculated in accordance with the Reference SRXB-87.1 methodology using BFN gamma TIP measurements and is [ ]. For comparison, Reference SRXB-87.1 reports a δT^{m}_{ij} of [ ] for D-Lattice plants. The BFN gamma TIP system is far less sensitive than neutron TIP systems to variations in TIP location within the corner water gap between fuel assemblies. Because δT^{m}_{ij} is determined by comparing TIP measurements in symmetrically operated core locations, it is independent of the C4/MB2 core model and core operating conditions.

The δS_{ij} is the uncertainty associated with update of calculated power by the core monitoring system to more closely match incore instrumentation. A BFN specific radial synthesis uncertainty (δS_{ij}) was calculated in accordance with the Reference SRXB-87.1 methodology using BFN gamma TIP measurements and is [ ]. For comparison, Reference SRXB-87.1 reports a δS_{ij} of [ ] for D-Lattice plants. δS_{ij} is a function of the core monitoring system update algorithm and is independent of core operating conditions. [,] a comparison of δS_{ij} to core operating conditions is not provided.

Utilizing the BFN specific values of $\delta T'_{ij}$, δT^{m}_{ij} and δS_{ij} results in a measured assembly power distribution uncertainty of [ ]. BFN Safety Limit Minimum Critical Power Ratio (SLMCPR) analyses are based on the radial bundle power uncertainty value of [ ] reported in the Reference SRXB-87.1 topical report rather than the BFN specific value of [ ]. The BFN specific value is conservative relative to the topical report value by [ ] due primarily to BFN implementation of gamma TIPs for LPRM calibration. Both the BFN specific and topical report bundle power uncertainty values are additionally very conservative relative to their respective TIP measurement databases due to the use of a correlation coefficient to increase calculated power uncertainty above calculated TIP uncertainty, contrary to measured data that support decreasing calculated power uncertainty below calculated TIP uncertainty. Even if a 50% reduction was assumed in the correlation coefficient, the BFN specific evaluation of the power uncertainty would be conservative relative to the value used in the SLMCPR analysis. Therefore, increasing the power distribution uncertainty is not necessary for the SLMCPR analysis of BFN.

The Reference SRXB-87.1 topical report database includes TIP measurements of cores containing many different fuel designs and identifies no correlation between C4/MB2 uncertainty and fuel design. Figure SRXB-87.1 demonstrates there is no significant variation in uncertainty determined from the BFN gamma TIP measurements for various mixes of fuel types. These measurements include mixed GE13 and GE14 cores operated in Unit 2 Cycle 11 and Unit 3 Cycle 13. Mixed cores of GE14 and ATRIUM-10 fuel were operated in Unit 2 Cycles 14 and 15 as well as Unit 3 Cycles 12 and 13.

C4/MB2 local power distributions are compared to bundle gamma scan data as reported in Tables 8.3, 8.4, and 8.5 of Reference SRXB-87.1 for 10x10 and other orthogonal lattice designs. These results indicate that there is no degradation in the uncertainty for 10x10 fuel relative to the other designs.

Reference SRXB-87.1

EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,* Siemens Power Corporation, October 1999.

Figure SRXB-87.1 BFN $\delta T'_{ij}$ Gamma TIP Response vs. Cycle Number

Figure SRXB-87.2 BFN $\delta T'_{ij}$ Gamma TIP Response vs. Power/Flow Ratio

Figure SRXB-87.3 BFN $\delta T'_{ij}$ Gamma TIP Response vs. Core Average Void Fraction

NRC RAI SRXB-88 (Units 2 and 3 only)

To address the adequacy of void-quality correlation bias and uncertainties, the staff understands that a plant specific calculation can be performed to assess the impact of the uncertainties on the operating limit minimum critical power ratio (OLMCPR). Provide the following additional information:

a. Discuss how the void-quality correlation bias and uncertainties are addressed for the projected Units 2 and 3 operation at EPU conditions.

b. Determine the net impact on the OLMCPR from a bias in the void-quality correlation within the uncertainty range based on full-scale-test data.

TVA Response to SRXB-88 (Units 2 and 3 only)

a. The AREVA analysis methods and the correlations used by the methods are applicable for both pre-EPU and EPU conditions as discussed in responses to previous BFN Unit 2/3 RAIs (SRXB-A.15, SRXB-A.26 through SRXB-A.29, SRXB-A.35), which were submitted to NRC by TVA on March 7, 2006 (ML060680583). The approach for addressing void-quality correlation bias and uncertainty remains unchanged and is applicable for BFN EPU operation. The approach for addressing void-quality correlation bias and uncertainty is described below.

   The **[ ]** void-quality correlation has been qualified by AREVA against both the FRIGG void measurements and ATRIUM-10 measurements. Despite the different geometrical configurations between FRIGG and ATRIUM-10, the **[ ]** correlation compares very well to the measured data as illustrated in Figure SRXB-88.1.

   The OLMCPR is determined based on the SLMCPR methodology and the transient analysis (ΔCPR) methodology. Void-quality correlation uncertainty is not a direct input to either of these methodologies; however, the impact of void-correlation uncertainty is inherently incorporated in both methodologies as discussed below.

   The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculations performed to determine the number of rods in boiling transition. One of the uncertainties considered in the SLMCPR methodology is the bundle power uncertainty. This uncertainty is determined through comparison of calculated to measured core power distributions. Any miscalculation of void conditions will increase the error between the calculated and measured power distributions and be reflected in the bundle power uncertainty. Therefore, void-quality correlation uncertainty is an inherent component of the bundle power uncertainty used in the SLMCPR methodology.

The transient analysis methodology is not a statistical methodology and uncertainties are not directly input to the analyses. The transient analysis methodology is a deterministic, bounding approach that contains sufficient conservatism to offset uncertainties in individual phenomena. Conservatism is incorporated in the methodology in two ways: (1) computer code models are developed to produce conservative results on an integral basis relative to benchmark tests, and (2) important input parameters are biased in a conservative direction in licensing calculations.

The transient analysis methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analysis methodology. Therefore, uncertainty in the void-quality correlation is inherently incorporated in the transient analysis methodology.

Based on the above discussions, the impact of void-quality correlation uncertainty is inherently incorporated in the analytical methods used to determine the OLMCPR. Biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. No additional adjustments to the OLMCPR are required to address void-quality correlation uncertainty.

b. A sensitivity calculation was previously performed for another plant to assess the impact of a bias in the void-quality correlation on the OLMCPR. The sensitivity calculation used an alternate void-quality correlation (Ohkawa-Lahey) that results in the prediction of lower void fractions than the **[ ]** correlation. The Ohkawa-Lahey predicted exit void fraction data is closer to the low end of the measured data (~ 2% to 3% bias relative to **[ ]**). These sensitivity calculations demonstrated that the void-quality correlation bias had small and offsetting impacts on SLMCPR and ΔCPR; there was no impact on the OLMCPR.

A BFN plant specific calculation was performed for a proposed EPU core design for Unit 3 Cycle 14 with the Ohkawa-Lahey alternate void-quality correlation. The BFN calculation demonstrated that the change in the SLMCPR (0.0017) and in the ΔCPR (0.0001) were small and did not impact the OLMCPR.

Figure SRXB-88.1 Void Fraction Correlation Comparison to FRIGG and ATRIUM-10 Test Data

NRC RAI SRXB-89 (Units 2 and 3 only)

To address the effect of bypass boiling on the stability oscillation power range monitor (OPRM) setpoints, a setpoint setdown was performed. Provide the following additional information:

a. Discuss how the bypass boiling effect is addressed for Units 2 and 3 OPRM setpoints.

b. Determine a method for conservatively accounting for the effect of bypass void formation on OPRM and average power range monitor sensitivity.

TVA Response to SRXB-89 (Units 2 and 3 only)

The impact of localized bypass boiling is a reduction of the LPRM signal due to the decreased local moderation of the fast flux. [

] Therefore, no degradation in the OPRM signal is expected due to bypass boiling and no additional conservatism above and beyond the Option III licensing basis is required.

NRC RAI SRXB-90 (Units 2 and 3 only)

Provide the following information regarding the AREVA LOCA analyses:

- the flow area above the hot bundle exit,
- power of the hot bundle, and
- perform an analysis assuming little or no downflow.

TVA Response to SRXB-90 (Units 2 and 3 only)

The flow area used at the hot bundle exit (Junction 10 from Figure SRXB-90.1) in the NRC-approved EXEM BWR-2000 methodology is the [

] For ATRIUM-10 Fuel in the BFN EPU Loss-of-Coolant Accident (LOCA) model, [

] The power of the hot bundle used in the BFN EPU LOCA model is [

].

[

] Therefore, to accommodate the NRC request, the 0.05 ft^2 top-peaked small break LOCA analysis was repeated with the modification that the injection of Low Pressure Core Spray (LPCS) was moved from the upper plenum (node 1 in Figure SRXB-90.2) to the bypass (node 9 in Figure SRXB-90.2). This allowed for the injection of water from the LPCS, which is needed to refill the lower plenum, without LPCS being available for CCFL into the core. Review of the results confirmed there was no top-down cooling from liquid entering the top of the bundle from the upper plenum after the time when LPCS flow starts (the LPCS did not fill the bypass and then flow into the upper plenum). The peak clad temperature (PCT) for this analysis is 1465°F.

Figures SRXB-90.3 through SRXB-90.5 show that the selected modeling is performing as intended. Figure SRXB-90.3 presents the liquid level in the bypass and demonstrates that some of the injected Emergency Core Cooling System water is held up in the bypass. Thus, all of the LPCS injection does not instantaneously drop into the lower plenum. The figure also shows that the bypass does not fill completely prior to reflood. Therefore, no injected LPCS water can spill into the upper plenum to drive downflow of liquid trough the core prior to reflood.

Figures SRXB-90.4 and SRXB-90.5 present the liquid mass flow rate at the hot bundle and average core exit junctions, respectively. These figures show that there is no liquid down flow from the upper plenum to the core after LPCS injection begins at 543 seconds.

Moving the LPCS injection and the associated absence of liquid water in the upper plenum may alter the system response in ways that are not directly related to the absence of countercurrent flow at the core outlet. Additionally, it should be recognized that moving the LPCS injection results in a model that no longer reflects the ECCS configuration of the BFN plants. Nonetheless, injection of LPCS into the bypass instead of the upper plenum is the closest modeling achievable to the requested analysis that is possible without RELAX code modifications.

Since LPCS is really injected into the upper plenum and the AREVA CCFL model has been shown by testing to be applicable for ATRIUM-10 fuel, the result obtained using the approved methodology (PCT = 1235°F) is a more appropriate result for these break conditions.

Figure SRXB-90.1 RELAX LOCA Hot Channel Nodal Diagram for Top-Peaked Axial Shapes

Figure SRXB-90.2 RELAX LOCA System Nodal Diagram

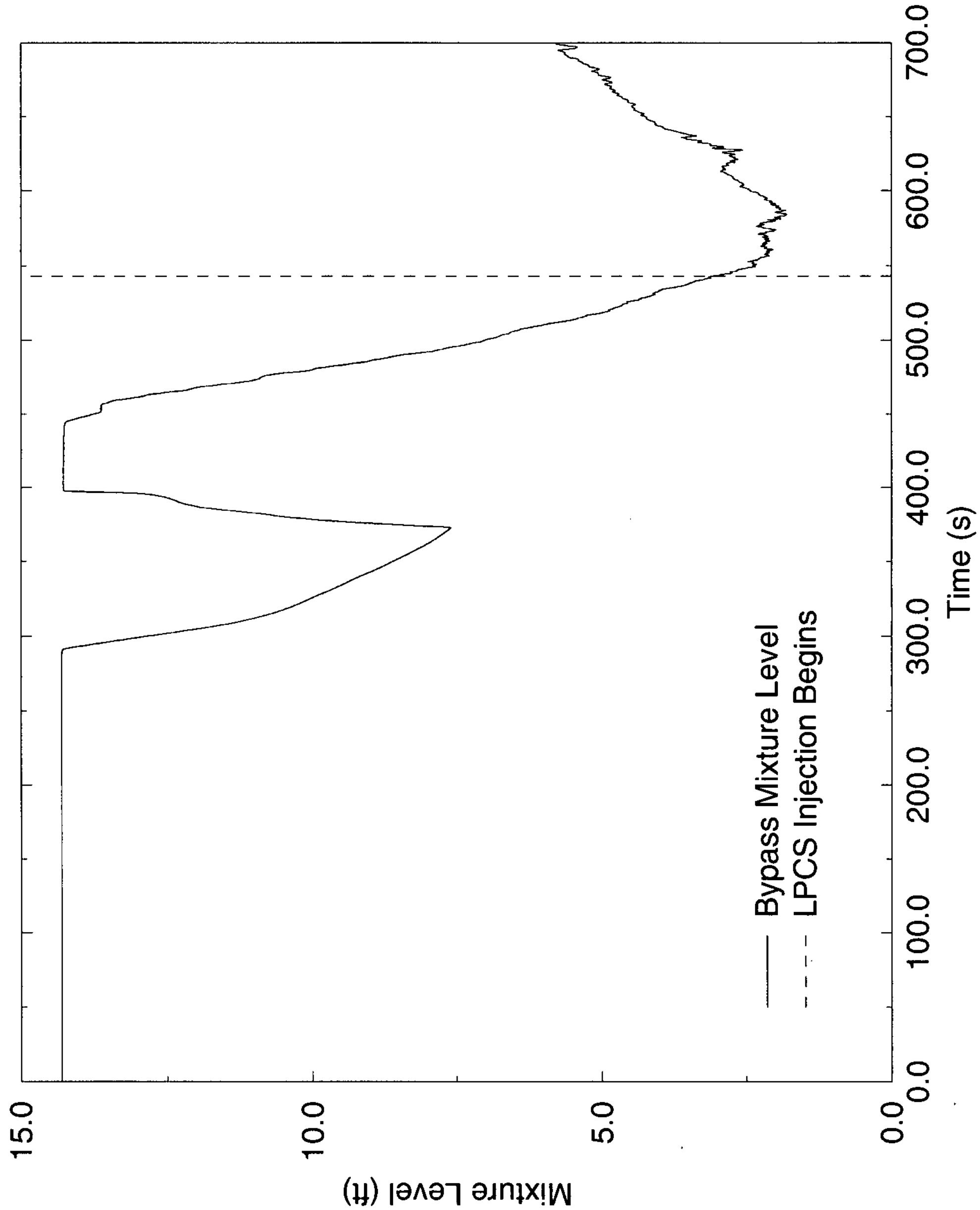

Figure SRXB-90.3 Bypass Mixture Level for .05 FT^2/PD TOP SF-BATT 102P/105F EPU With Bypass Injection of LPCS

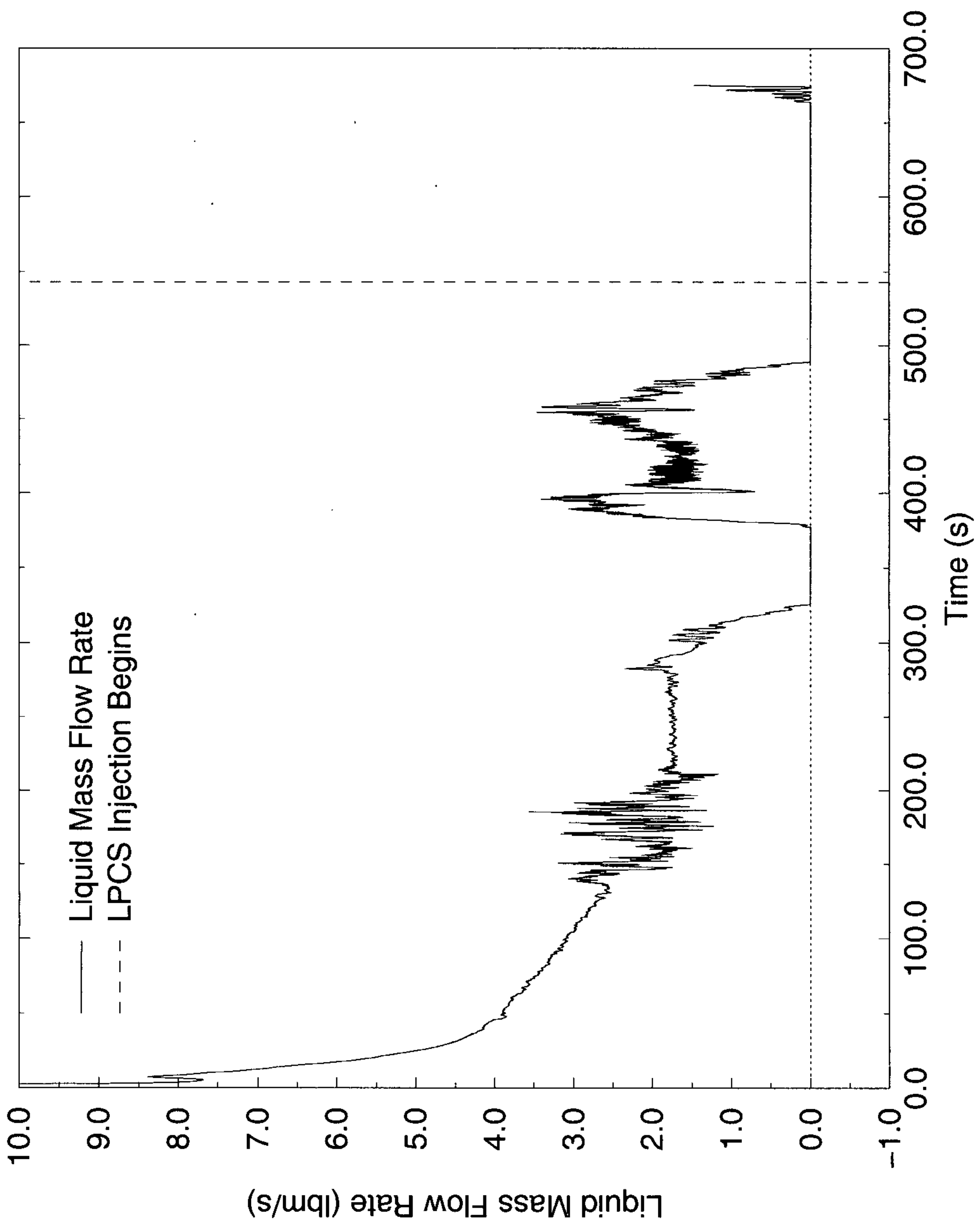

Figure SRXB-90.4 Hot Channel Exit Liquid Mass Flow Rate for .05 FT^2/PD TOP SF-BATT 102P/105F EPU With Bypass Injection of LPCS

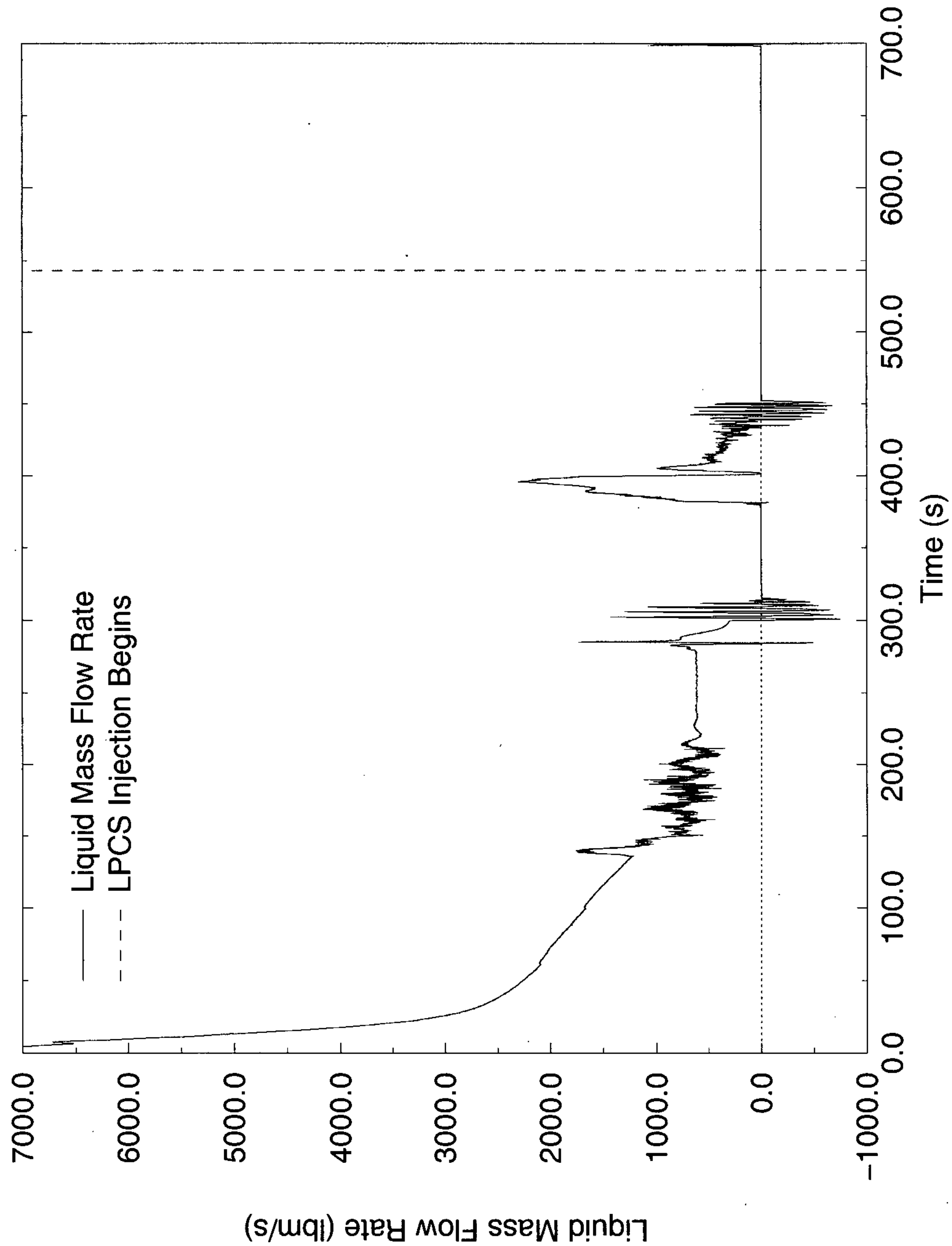

Figure SRXB-90.5 Average Core Exit Liquid Mass Flow Rate for .05 FT²/PD TOP SF-BATT 102P/105F EPU With Bypass Injection of LPCS

ENCLOSURE 3

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

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

RESPONSE TO ROUND 16 REQUEST FOR ADDITIONAL INFORMATION
SRXB-74/86 AND SRXB-87 THROUGH SRXB-90
AFFIDAVIT

This enclosure provides AREVA's affidavit for Enclosure 1.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) 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 "Responses to NRC RAI – Round 16 for Browns Ferry EPU SRXB-74/86, SRXB-87, SRXB-88, SRXB-89, and SRXB-90," dated March of 2008 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 6th
day of March, 2008.

____

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010