September 22, 2006

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 No. 50-259
Tennessee Valley Authority	)	

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT 1 – EXTENDED POWER UPRATE (EPU) (TAC NO. MC3812)

By letter dated June 28, 2004 (ADAMS Accession No. ML041840109), TVA submitted an application to the NRC for EPU of BFN Unit 1. The proposed license amendment increases the maximum power level authorized by Section 2.C. (1) of the license from 3293 megawatts thermal (MWt) to 3952 MWt, an approximate 20% increase in thermal power.

This supplement to TVA's application for the extended power uprate of BFN Unit 1 provides for interim operation at 105% of original licensed thermal power (OLTP) until such time as certain steam dryer analyses can be completed. Operation at 105% OLTP will support the collection of operational data and the analysis of steam dryer fluctuating loads prior to power ascension to 120% OLTP.

NRC approval of the requested increase in reactor thermal power level will allow TVA to implement operational changes to generate and supply a higher steam flow to the turbine generator. Higher steam flow is accomplished by increasing the reactor power along specified control rod and core flow lines and by increasing reactor operating pressure. This increase in steam flow will enable increasing the electrical output of the plant.

OLTP for Browns Ferry Unit 1 was based on the vendor-guaranteed power level of 3293 MWt for the reactor.

In license amendment request TS-431, TVA provided analytical information to support the Unit 1-specific EPU request. This documentation included the Power Uprate Safety Evaluation Report (PUSAR), supplemented with evaluations and calculations as necessary to demonstrate acceptability of operation at 120% OLTP. Because of uncertainties in the method proposed for predicting steam dryer stresses, the NRC staff is unable to conclude that the modified BFN steam dryer stress analysis provides reasonable assurance that steam dryer integrity will be maintained at EPU conditions. To provide adequate confidence in calculated stresses, a steam dryer analysis that is based on plant-specific data is needed prior to exceeding 3458 MWt. Substantial operating history exists from BFN Units 2 and 3 (approximately 14 reactor-years combined) at 3458 MWt, which demonstrates the steam dryer is not expected to experience degradation at 3458 MWt. Therefore, TVA proposes achieving EPU for BFN Unit 1 in a two-step approach:

- Step 1: **Unit 1 operation licensed for 105% OLTP**

  Approval for operation at 3458 MWt (105% OLTP) with a 30 psig increase in reactor steam dome pressure to obtain steam dryer pressure data. This is an interim step to allow steam line data to be obtained, if it is not available from another BFN unit.

- Step 2: **Unit 1 operation licensed for 120% OLTP**

  Plant-specific data obtained from a BFN operating unit (Unit 1, 2, or 3) at 3458 MWt would be used as an input to the BFN steam dryer stress analysis to provide reasonable assurance of steam dryer integrity for operation at greater than 105% OLTP.

With NRC approval of the license amendment for 120% OLTP operation, power ascension from 105% to 120% OLTP would commence in accordance with a steam dryer monitoring plan and a new license condition that provides for monitoring and evaluating plant data, and taking prompt action in response to potential adverse flow effects on the steam dryer. The steam dryer monitoring plan was provided in the response to RAI EEMB.D.7 in TVA to NRC letter dated July 26, 2006 (ML062200277).

The analyses and evaluations previously performed for EPU operation at 120% OLTP in the PUSAR support and/or bound operation at lesser power levels, subject to the operating restrictions and limitations that would be applicable. That is, the analyses demonstrate that operation anywhere in the power range from 0 to 120% OLTP is acceptable. Due to the change in power level, a Supplemental Reload Licensing Report will be generated to reflect the interim power level as delineated in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

Details and references from the following discussion are provided in Enclosure 4. On September 8, 1998, the NRC issued TS Amendments 254 and 214 for BFN Units 2 and 3, authorizing operation of BFN Units 2 and 3 at 105% OLTP (Reference 1). TVA has performed a review of the NRC Safety Evaluation (SE) issued for those 5% uprate amendments to verify that the information in the SE is supported and/or bounded by the information for a Unit 1 20% uprate with 30 psi reactor pressure increase. General Electric has reviewed their input into the 120% OLTP PUSAR, and has concluded that the evaluations performed would bound operation at 105% OLTP. The basis for the evaluations is provided by a combination of one or more of the following sources:

- Unit 1 Safety Analysis Report for Extended Power Uprate (Enclosure 4 of Reference 2);
- Responses to Requests for Additional Information for a 20% uprate for BFN Unit 1 (References 4 through 25); and
- Other BFN Unit 1 submittals.

The review has verified that the PUSAR information provided for a 20% uprate is directly applicable to the condition/result for a 5% uprate. The review concluded that there are no new conditions created by the BFN Unit 1 restart and three unit operation which affect the information.

As noted in Enclosure 1, the plant conditions that will exist for BFN Unit 1 at 105% OLTP are slightly different than those for BFN Units 2 and 3 operation at 105% OLTP. This is due to the installation of the plant modifications on BFN Unit 1 to support operation at 120% OLTP; EPU modifications of Units 2 and 3 have not yet been completed. In all cases, the resulting conditions at 105% OLTP have been evaluated at the more severe condition of 120% OLTP and found to be acceptable.

As delineated in section II of Enclosure 1, specified operating parameters are being changed to support the increase in reactor power and the associated 30 psi increase in reactor pressure. Generally, these changes can be classified as:

- A change in reactor thermal power or thermal power related parameter to reflect the increase to 105% OLTP.

- Pressure-related setpoints are being scaled for the 30 psi increase.

- The changes in Unit 1 operating parameters identified in the response to NRC Item of Interest 13 of Enclosure 5 are representative of the projected operating heat balance at 3458 MWt and closely match the reactor heat balance contained in Table 1-2 and Figure 1-1 from NEDC-32751P, "Power Uprate Safety Analysis for the Browns Ferry Nuclear Plant Units 2 & 3," September 1997.

Uprated operation of Unit 1 will involve a slightly higher reactor vessel dome pressure (30 psi increase) to establish desired inlet pressure conditions at the turbine. A steam dome operating pressure of 1035 psig (the same as BFN Units 2 and 3) will account for the larger pressure drop through the steam lines at higher flow rates and will provide sufficient pressure control and turbine flow capability. Because the higher steam dome pressure is maintained from 105% to 120% OLTP, the pressure setpoints selected for EPU will remain the same and will ensure that safety and operating margins are met over the entire range of uprated power levels.

Because BFN Unit 1 will be modified and analyzed for operation at 120% OLTP, TVA requests that the transition from step 1 to step 2 only be contingent upon NRC review and acceptance of the steam dryer stress report. All other safety evaluations that support operation at 105% OLTP would remain valid for operation at 120% OLTP. The BFN Unit 1 steam dryer will be modified prior to restart to provide structural reinforcement determined to be necessary for EPU operation. These modifications include the replacement of the ½” hood face plates, and 3/8” cover plates with 1” thick plate. Dryer tie-bars will also be replaced with an updated design to provide additional reinforcement. Improved weld designs will be utilized to provide additional reinforcement for the vane bank plate that joins the hood face plates.

To demonstrate that structures, systems, and components will perform satisfactorily upon restart and under power uprate conditions, TVA will conduct a startup test program for Unit 1 as described in the following:

- PUSAR section 10.4 and Enclosure 8 to the June 28, 2004, license amendment application provided the BFN Extended Power Uprate Startup Test Program. This testing plan was supplemented by TVA response to NRC RAIs in letters dated February 23, 2005 (ML0050560150) and April 25, 2005 (ML051170244). This test program includes testing from less than 90% OLTP to 120% OLTP.

- TVA letter to the NRC dated August 15, 2005 (ML052280327) provided a detailed description of the modifications completed or in progress in support of BFN Unit 1 restart and a description of the testing planned as part of the

BFN Unit 1 Restart Test Program. Included was a description of the power ascension testing planned for BFN Unit 1.

- TVA will perform the applicable portions of the Restart Test Program planned to occur up to and including operation at 105% OLTP. The remainder of the program above 105% OLTP will be conducted following NRC approval of operation at 120% OLTP.

As part of the two-step approach to EPU, TVA will perform two large transient tests:

1. A large transient test that simulates the rejection of generator load will be performed at 105% OLTP. This test and the Unit 1 Restart Test Program will demonstrate the satisfactory performance of equipment that was modified for plant restart and power uprate. This large transient test will provide additional confidence that the transient analysis properly reflects the expected behavior of the interacting systems.

2. An MSIV Closure with valve position scram large transient test will be performed at 115% to 120% OLTP.

Note that the above proposed large transient testing (LTT) at 105% OLTP and 120% OLTP supersedes TVA's commitment for LTT in TVA's letter of July 31, 2006.

The sequence of events following NRC approval of BFN Unit 1 operation at 105% OLTP is expected to be as follows:

- NRC approves BFN Unit 1 operation at 105% OLTP (3458 MWt) with accompanying 30 psi increase in reactor pressure.

- BFN Unit 1 restarts; proceeds to operation at 105% OLTP.

- Steam dryer data gathered.

- Steam dryer analysis completed.

- The first Large Transient Test (i.e., generator load reject) will be completed within 30 days of reaching 105% OLTP. This is within 15% of EPU (120% OLTP) and thus satisfies the requirements of Appendix L of the NRC approved General Electric (GE) Licensing Topical Report (LTR) NEDC-32424P-A, "Generic Guidelines for General

Electric Boiling Water Reactor Extended Power Uprate," (ELTR-1).

- Results of steam dryer analysis submitted to NRC with request for approval of 120% OLTP (3952 MWt). This request will rely on the previously submitted License Amendment Request and all supporting submittals.

- NRC approves BFN Unit 1 operation at 120% OLTP.

- BFN Unit 1 proceeds to 120% OLTP in 5% OLTP increments as delineated in earlier correspondence. (The power ascension monitoring plan for the steam dryer and main steam lines is described in the response to RAI EEMB.D.7 in TVA letter to NRC dated July 26, 2006 (ML062200277).)

- The second Large Transient Test (i.e., MSIV closure) will be completed within 30 days of reaching 115% to 120% OLTP.

Five additional BFN Unit 1 or BFN Units 1, 2, and 3 amendment requests have been submitted to the NRC, that following approval, would implement changes that are assumed for BFN Unit 1 restart to 105% OLTP. These changes would make the BFN Unit 1 TS consistent with the BFN Units 2 and 3 TS in these respective areas. Accordingly, TVA requests that these BFN Unit 1 TS changes be issued prior to or concurrent with the amendment being requested by this Supplement to TS-431. These include:

- BFN Unit 1 TS Change 430 – Power Range Neutron Monitor Upgrade With Implementation of Average Power Range Monitor and Rod Block Monitor Technical Specification Improvements and Maximum Extended Load Line Limit Analyses, submitted to NRC by TVA letter dated November 10, 2003;

- BFN Unit 1 TS Change 433 – 24 Month Fuel Cycle, submitted to NRC by TVA letter dated August 16, 2004;

- BFN Unit 1 TS Change 443 – Oscillation Power Range Monitor (OPRM), submitted to NRC by TVA letter dated January 6, 2006;

- BFN Units 1, 2, and 3 TS Change 447 - Extension of Channel Calibration Surveillance Requirement Performance Frequency and Allowable Value Revision, submitted to NRC by TVA letter dated August 16, 2004; and

- BFN Unit 1 TS Change 455 - Safety Limit Minimum Critical Power Ratio (SLMCPR) - Cycle 7 Operation, submitted to NRC by TVA letter dated May 1, 2006.

Enclosure 1 is a description and evaluation of the proposed changes to the BFN Renewed Operating License and TS to allow operation of BFN Unit 1 at 105% OLTP. Included in this enclosure is TVA's determination that the proposed changes do not involve a significant hazards consideration.

Enclosure 2 contains the affected page list and copies of the Unit 1 Renewed Operating License and TS pages marked-up to show the proposed changes for 105% RTP and associated 30 psi increase in reactor pressure.

Enclosure 3 contains the affected page list and copies of the Unit 1 Renewed Operating License and TS pages revised to show the proposed changes for 105% RTP and associated 30 psi increase in reactor pressure.

Enclosure 4 provides a correlation between the sections of the Safety Evaluation issued for amendments 254 and 214 for 5% OLTP uprates for BFN Units 2 and 3, respectively, and the information submitted for BFN Unit 1 operation at 120% OLTP. Amendments 254 and 214 authorized operation of BFN Units 2 and 3 at 105% OLTP and were issued in response to TVA’s request TS-384.

Enclosure 5 provides change considerations of interest to the NRC and provides TVA’s response to each.

Enclosure 6 provides a listing of the new regulatory commitments made in this submittal.

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 22th day of September, 2006.

Sincerely,

Original signed by:

William D. Crouch
Manager of Licensing
and Industry Affairs

Enclosures:

1. TVA Evaluation of the Proposed Change
2. Marked-Up Renewed Operating License and TS Pages
3. Revised Renewed Operating License and TS Pages
4. Correlation Between SER for Units 2 and 3 Amendment for 105% OLTP Operation and Information Submitted for BFN Unit 1 Operation at 120% OLTP
5. Change Considerations of Interest to the NRC
6. New Regulatory Commitments Made

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT 1 – EXTENDED POWER UPRATE (EPU)

TVA EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION

By letter dated June 28, 2004 (ADAMS Accession No. ML041840109), TVA submitted an application to the NRC for EPU of BFN Unit 1. The proposed license amendment increases the maximum power level authorized by Section 2.C.(1) of the license from 3293 megawatts thermal (MWt) to 3952 MWt, an approximate 20% increase in thermal power.

NRC approval of the requested increase in reactor thermal power level will allow TVA to implement operational changes to generate and supply a higher steam flow to the turbine generator. Higher steam flow is accomplished by increasing the reactor power along specified control rod and core flow lines and by increasing reactor operating pressure. This increase in steam flow will enable increasing the electrical output of the plant.

This supplement requests an amendment to Renewed Operating License DPR-33 for BFN Unit 1. The proposed TS changes requested with this supplement represent those required to support increasing the BFN Unit 1 licensed thermal power from 3293 MWt to 3458 MWt with an attendant 30 psi increase in reactor pressure. This represents an approximate 5% increase above the original RTP of 3293 MWt. This is commonly referred to as a stretch power uprate (PU) application.

This supplement to the amendment request provides for a two-step process to confirm structural integrity of the steam dryer prior to operation above 3458 MWt:

1. approval for operation at 3458 MWt to allow gathering of plant data to evaluate the modified steam dryer; and
2. analysis of the steam dryer provided to the NRC for approval of operation at 3952 MWt.

The technical bases for this request follows the guidelines contained in GE Licensing Topical Reports (LTRs) for Extended Power Uprate Safety Analysis NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-1) and NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-2). The ELTR process was initially developed during the EPU efforts at plant Hatch and Monticello. Several other BWRs have submitted similar requests for up to a 20% increase in original license thermal power.

The modifications required for operation at 120% OLTP, except for those related to the TS changes being requested by this supplement, will be installed prior to BFN Unit 1 restart. The modifications for operation at 120% OLTP are compatible for operation at 105% OLTP since operation at 105% OLTP is simply a de-rate condition compared to operation at 120% OLTP.

The EPU submittal contains a background, description, and justification of the requested change to the BFN Unit 1 licensing bases to include credit for containment overpressure for ensuring the adequacy of Emergency Core Cooling System (ECCS) pump net positive suction head (NPSH) following design basis accidents. The current licensing basis for BFN Unit 1 does not now include credit for containment overpressure. The need for containment overpressure credit for BFN Unit 1 is consistent with the need for BFN Units 2 and 3. The basis for containment overpressure has been updated as part of the BFN EPU submittals. Analysis for containment overpressure at 120% OLTP bounds the need for containment overpressure at 105% OLTP. TVA letters to NRC dated August 4 and August 31, 2006, provided response to RAIs and supporting evaluation, respectively, concerning containment overpressure. The August 4 letter states:

- "The results conclude that adequate NPSH and containment overpressure are available for the full spectrum of events analyzed."

- "Figures ACVB.56/54-1 through 6 show the COP required and COP available for each case as a function of time."

To support EPU implementation, TVA proposed a change to the upper bound peak cladding temperature. This change was addressed in PUSAR section 4.3 and Table 4.5. These PUSAR sections also addressed 105% OLTP.

II. PROPOSED CHANGES

TVA is requesting an increase in the licensed reactor thermal power for BFN Unit 1 from 3293 MWt to 3458 MWt. This represents an increase of approximately 5% from the current licensed thermal power. TVA is also requesting a 30 psi increase in reactor pressure.

Implementation of this power uprate requires revision to the BFN Renewed Operating License and a number of the BFN Unit 1 Technical Specifications. Table E1-1 provides a listing of proposed changes to these documents necessary to support implementation of the power uprate. The table identifies the applicable document section and provides a brief description of each change. The power uprate safety analyses were conservatively performed for a 20% increase in reactor power; however, this request for license amendment and revised Technical Specifications is for a 5% increase.

Table E1-1

Changes to Browns Ferry Unit 1 Renewed Operating License and TS

#	Document Section	Description of Change
1	License Condition 2.C.(1)	Revise the value of the Maximum Power Level to the uprated power level of 3458 MWt.
2	1.1, Definitions RATED THERMAL POWER (RTP)	Revise the value of the Rated Thermal Power (RTP) definition to the uprated power level of 3458 MWt.
3	SR 3.1.7.5 SLC Surveillance	Revise minimum required amount of Boron-10 required from $\geq$ 186 pounds to $\geq$ 203 pounds.
4	SR 3.1.7.7 SLC Surveillance	Revise the Standby Liquid Control System pump discharge pressure requirement from 1275 psig to 1325 psig.
5	Table 3.3.1.1-1, Function 2.b	Revise the Allowable Value formula for the Flow Biased Simulated Thermal Power – High.
6	Table 3.3.1.1-1, Note (b)	Revise the Allowable Value formula Flow Biased Simulated Thermal Power - High for single loop operation.

Table E1-1 Changes to Browns Ferry Unit 1 Renewed Operating License and TS		
#	**Document Section**	**Description of Change**
7	Table 3.3.1.1-1, Function 3	Revise the Allowable Value for the Reactor Vessel Steam Dome Pressure scram function from ≤ 1055 psig to ≤ 1090 psi.
8	SR 3.3.4.2.3, Surveillance Requirement	Revise the ATWS-RPT Reactor Steam Dome Pressure from 1146.5 psig to 1175 psig.
9	SR 3.4.3.1, Surveillance Requirement	Revise the S/RV setpoint pressures from 1105, 1115, and 1125 psig to 1135, 1145, and 1155 psig, respectively.
10	3.4.10, LCO	Revise the Reactor Steam Dome pressure LCO from ≤ 1020 psig to ≤ 1050 psig.
11	SR 3.4.10.1, Surveillance Requirement	Revise the surveillance requirement Reactor Steam Dome Pressure from ≤ 1020 psig to ≤ 1050 psig.
12	SR 3.5.1.7, Surveillance Requirement	Revise the HPCI surveillance test reactor pressure range from ≤ 1010 psig and ≥ 920 psig to ≤ 1040 psig and ≥ 950 psig.
13	SR 3.5.3.3, Surveillance Requirement	Revise the RCIC surveillance test reactor pressure range from ≤ 1010 psig and ≥ 920 psig to ≤ 1040 psig and ≥ 950 psig.
14	5.5.12, Primary Containment Leakage Rate Testing Program	Revise the peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, to 48.5 psig.

Each Renewed Operating License and TS change is evaluated below:

Table E1-1, Items #1 and #2

The increase and redefinition of rated thermal power (RTP) for BFN Unit 1 reflects the 105% OLTP increase. The increase is bounded by the analyses performed for 120% extended power uprate (EPU). The technical basis follows the guidelines contained in General Electric (GE) Licensing Topical Reports (LTR) NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-1) and NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR-2). The generic evaluations are supplemented by plant-specific analyses documented in NEDC -33101P, "Browns Ferry Unit 1 Safety Analysis Report for Extended Power Uprate," (PUSAR).

ELTR-1 and ELTR-2 provide an NRC-accepted methodology for requesting power uprates up to 120% OLTP. Together with the analyses documented in the PUSAR, it was determined that the 105% OLTP power uprate can be achieved with no significant hazards impact.

Table E1-1, Item #3

The BFN U1 Cycle 7 core design for 3952 MWt requires increasing the minimum reactor boron concentration from 660 ppm to 720 ppm. Initially, this same core will be operated de-rated to 3458 MWt. Even though operation at 3458 MWt does not require additional boron, the minimum quantity of boron-10 is being increased to maintain BFN Unit 1 consistent with the analysis. Thus, the TS minimum quantity of boron-10 is increased from 186 pounds to 203 pounds. This change is the same as the one requested per EPU.

Table E1-1, Item #4

As discussed in Section 6.5 of the PUSAR, the surveillance test pressure is based on the maximum SLC System injection pressure and allowances for system test inaccuracies. Therefore, this pressure is increased from the current licensed thermal power value of 1275 psig to 1325 psig to account for the increase in system injection pressure at power uprate conditions. Increasing the test pressure by 50 psi assures the continued capability of the positive displacement pumps to deliver design rated flow at operating pressures expected at the uprated conditions. This change, therefore, maintains the original intent of surveillance requirement (SR) 3.1.7.7.

Table E1-1, Items #5 and #6

The allowable value for the APRM Flow Biased Simulated Thermal Power - High scram setpoint intercept is changed from 58% to 66% RTP, consistent with the revised analytical limit for 105% OLTP. The flow-biased APRM simulated thermal power monitor analytical limit and scram setpoints are lowered proportionally to the increase in rated power, such that they remain substantially unchanged in terms of absolute power and core flow. As discussed in PUSAR Section 5.3.5, this proposed change ensures that the pre-uprate design margins and licensing basis are preserved for operation at the uprated power.

Table E1-1, Item #7

The allowable value for the reactor vessel steam dome pressure scram function was increased 35 psi (as opposed to 30 psi) from ≤ 1055 psig to ≤ 1090 psig in order to place the instrument in the middle of its calibration range. The chosen allowable value is acceptable based on the analytical limit for the parameter.

The reactor vessel steam dome high pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for power uprate is increased to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and the steam flow capability of the turbine. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating condition and the steam flow capability of the turbine control valves at their maximum stroke. An operating steam dome pressure of 1035 psig, which is 30 psi higher than the current operating dome pressure, is expected. Therefore, the high pressure scram is increased by approximately 35 psi to preserve existing margins to reactor scram.

The high pressure scram terminates a pressurization transient not terminated by direct scram or high neutron flux scram. The setting is maintained above the nominal reactor vessel operating pressure and below the specified analytical scram limit used in the safety analyses. The revised high pressure scram setpoint will preserve the hierarchy of pressure setpoints. This means that the high pressure scram setpoint will remain below the opening setpoint of the safety-relief valves (S/RVs). The S/RV nominal setpoints are also increased 30 psi, as described in Table E1-1, item 9.

Table E1-1, Item #8

The allowable value was increased from 1146.5 psig to 1175.0 psig, a 28.5 psi increase. An increase of 30 psi would have yielded an allowable value of 1176.5 psig. Due to human factors consideration, the value of 1175 psig was chosen. An allowable value of 1175 psig is consistent with that selected for the BFN Units 2 and 3 5% OLTP power uprates, and below the analytical limit of 1177 psig.

The ATWS-RPT high pressure setpoint initiates a trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not (but should) occur. The analytical limit for the ATWS-RPT high pressure setpoint was increased 30 psi in the power uprate ATWS safety evaluations to account for the 30 psi increase in vessel operating pressure, S/RV setpoints, etc. The analyses demonstrate that the ATWS criteria are met with the higher analytical limits. Therefore, the allowable value is increased consistent with the analytical limit used in the safety analysis. Raising the ATWS-RPT high pressure setpoint to correlate with the increased operating pressure and analytical limit will tend to prevent unnecessary recirculation pump trips following pressurization transients with reactor scram (e.g., turbine trip or load rejection with bypass). Recirculation pump operation following a scram allows for better mixing of the reactor coolant and reduces thermal stratification in the vessel.

<u>Table E1-1, Item #9</u>

The S/RVs are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. The S/RV lift setpoints are increased to accommodate the increase in operating pressure that accompanies 105% OLTP power uprate. The increase in S/RV setpoints ensures that adequate margins are maintained so that the increase in dome pressure during normal operation does not result in an increase in the number of unnecessary S/RV actuations. The setpoint increase also maintains the hierarchy of pressure setpoints described in Table E1-1, item 7. above. Transient evaluations include a positive 3% tolerance to the nominal setpoints. Transient peak vessel pressure increases at uprated conditions, but remains below the 1375 psig American Society of Mechanical Engineers Code limit.

The adequacy of BFN Unit 1's S/RVs to operate at uprated temperatures and pressures was evaluated per the methodology of Section 5.6.8 of ELTR-1. The results are provided in PUSAR Section 3.2. The reactor operating pressure and temperature increases of less than 40 psi and 5° F, respectively, used in that evaluation bound the uprated operating conditions.

The impact of power uprate on the containment dynamic loads due to S/RV discharge has also been evaluated. The vent thrust loads with power uprate were calculated to be less than the loads used in the containment analysis. The effect of power uprate on S/RV air-clearing, the discharge line, the pool pressure boundary, and submerged structure drag loads was also considered. The small increase in the setpoint pressure is well within the margin in the S/RV loads defined in the Mark I Long Term Torus Integrity Program. Therefore, power uprate does not impact the S/RV load definitions used in the containment analysis.

Table E1-1, Items #10 and #11

The maximum nominal operating dome pressure for power uprate is changed to be consistent with Units 2 and 3. The operating pressure will be increased approximately 30 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow.

Table E1-1, Items #12 and #13

The reactor operating pressure range for HPCI and RCIC surveillance tests at high pressure is increased to correspond with the increase in normal reactor operating pressure that accompanies the 105% OLTP power uprate. The change is needed to provide a more appropriate test range for the higher reactor operating pressure. The requested changes will allow the 92 days demonstration of HPCI and RCIC capability to be performed at normal reactor operating pressures, which meets the original intent of the TSs. The pre-uprate flow rates remain valid for power uprate conditions.

Table E1-1, Item #14

The Browns Ferry 10 CFR 50, Appendix J, test program required by Technical Specifications requires periodic containment pressurization testing (Type A test), testing of containment penetrations (Type B test), and containment isolation valve and test boundary testing (Type C test) to the calculated peak containment pressure (Pa). Calculated peak containment pressure changes to 48.5 psig for the power uprate, as shown in PUSAR Table 4-1. Therefore, the 10 CFR 50, Appendix J, test program is revised to reflect this calculated peak containment pressure value.

III. BACKGROUND

An increase in the electrical output of a BWR unit is accomplished primarily by generating and supplying higher steam flow to the turbine generator. A higher steam flow is achieved by increasing the reactor power along specified control rod and core flow lines and increasing the reactor operating pressure. As delineated in section II, a limited number of operating parameters are being changed and modifications are being made to some equipment in order to support the new reactor power.

Most BWR plants, including BFN, have an as-designed equipment and system capability to accommodate steam flow above the original rating. In addition, continuing improvements in the analytical techniques (computer codes and data) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs have resulted in an increase in the design and operating margins between calculated safety analysis results and the licensing limits. These available safety analyses margins, combined with the as-designed equipment, system and component capabilities, provide BWR plants the capability to increase their thermal power ratings between 5% and 20%.

In a September 8, 1998 NRC letter (ML020100022), NRC issued Amendments 254 and 214 for Units 2 and 3, respectively, which authorized increasing the maximum power level for each unit from 3293 MWt to 3458 MWt. This power increase was primarily achieved by an increase in the reactor power along the existing rod and core flow control lines and an approximate 30 psig increase in reactor dome pressure. As BFN Unit 1 was in its current extended outage, that uprate was not requested for BFN Unit 1 at that time.

BFN Unit 1 has significant design margin allowing the plant to be safely uprated beyond the original licensed power level. For the BFN Unit 1 uprate to 105% OLTP, a higher steam flow is achieved by increasing the reactor power along specified control rod and core flow lines, and increasing reactor operating pressure approximately 30 psig. Following modifications, sufficient steam flow will be available to meet turbine demands at 120% OLTP uprated conditions. A listing of currently planned modifications (other than steam dryer modifications) was provided in Enclosure 7 to the June 28, 2004 (ML041840109) EPU submittal.

Detailed evaluations / analyses of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed. It was determined that BFN Unit 1 can be safely operated at 3952 MWt. The evaluations / analyses bound the increase to 3458 MWt being requested by this supplement.

IV. TECHNICAL ANALYSIS

The analyses and evaluations previously performed for EPU operation at 120% OLTP support operation at lesser power levels, subject to the operating restrictions and limitations that would be applicable at the lower power level. That is, analyses demonstrate that operation anywhere in the power range from 0 to 120% OLTP is acceptable.

Analysis Basis

BFN Unit 1 is currently licensed at 3293 MWt, and the original safety analyses were based on this value. The ECCS-LOCA and the containment safety analyses were based on a power level of 1.02 times the licensed power level. The EPU safety analyses are based on a power level of 1.02 times the EPU power level (120% OLTP), unless the 2% power factor from the Regulatory Guide 1.49 is already accounted for in the analysis. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Margins

The EPU analysis basis ensures that the power dependent margins prescribed by the Code of Federal Regulations (CFR) are maintained by meeting the appropriate regulatory criteria. NRC-approved or industry-accepted computer codes and calculation techniques were used to perform the calculations that demonstrate meeting the acceptance criteria. Similarly, design margins specified by application of the American Society of Mechanical Engineers (ASME) design rules are maintained, as are other margin-ensuring criteria used to judge the acceptability of the plant. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Fuel Thermal Limits

No change is required in the basic fuel design to achieve Reactor Thermal Power plant licensing limits. No increase in allowable peak bundle power is requested. The current fuel operating limits will still be met. Fuel thermal limit monitoring thresholds are adjusted to account for the power change. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC. No new fuel design is required. Future fuel designs will meet acceptance criteria approved by the NRC. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Makeup Water Sources

The BWR design concept includes a variety of means to inject water into the reactor vessel during postulated events. There are numerous safety-related and non-safety-related cooling water sources. The safety-related cooling water sources maintain core integrity by providing adequate cooling water.

Power uprate does not change the available water sources. The increase in reactor steam dome pressure of approximately 30 psig does increase slightly the discharge pressure requirement for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. Evaluation has determined that these systems are capable of meeting their design flow requirements at the higher discharge pressure conditions. NRC accepted methods have been used for analyzing the Emergency Core Cooling System (ECCS) performance during a LOCA. Analyses have been performed for operation at both 105% and 120% OLTP and demonstrated to meet applicable limits.

Containment Overpressure

Adequate Net Positive suction head (NPSH) margin is required during the LOCA to assure adequate pump operation. The NPSH margins for the low pressure ECCS pumps were evaluated for limiting conditions following a design basis LOCA. The limiting NPSH conditions occur during both short-term and long-term pump operation and depend on the total pump flow rates, debris loading on the ECCS suction strainers, and suppression pool temperature.

The NPSH evaluation shows that BFN Unit 1 requires containment overpressure credit to ensure adequate NPSH for ECCS pumps. BFN Units 2 and 3 previously requested and received containment overpressure credit of 3 psig for the short term following a LOCA, and 1 psig for the long term following a LOCA (ML020090448). That credit was needed as part of resolution of NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors." As with BFN Units 2 and 3, resolution of NRC Bulletin 96-03 for BFN Unit 1 also requires containment overpressure credit to ensure adequate NPSH for its ECCS pumps. BFN Unit 1 does not now include credit for containment overpressure. The need for containment overpressure credit for BFN Unit 1 is consistent with the need for BFN Units 2 and 3. The basis for containment overpressure has been updated as part of the BFN EPU submittals. The analysis for containment overpressure at 120% OLTP bounds the need for containment overpressure at 105% OLTP. TVA letters to NRC dated August 4 and August 31, 2006, provided response to RAIs and supporting evaluation, respectively, concerning containment overpressure. The August 4 letter stated:

- "The results conclude that adequate NPSH and containment overpressure are available for the full spectrum of events analyzed."
- "Figures ACVB.56/54-1 through 6 show the COP required and COP available for each case as a function of time."

Design Basis Accidents

Design Basis Accidents (DBAs) are very low probability hypothetical events whose characteristics and consequences are considered in the design of the plant to ensure that the consequences of these postulated events are within acceptable regulatory limits. For BWR licensing evaluations, capability evaluations demonstrate the ability to cope with the range of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and the most limiting small lines. This break range bounds the full spectrum of large and small, high and low energy line breaks; and the success of plant systems to mitigate the accidents, while accommodating a single active equipment failure in addition to the postulated LOCA. These assessments are as follow:

- **Challenges to Fuel**

  The ECCS Performance Evaluation described in ELTR-1, Section 5.3.1 (Reference 2), and PUSAR Section 4.3 was conducted through application of the 10 CFR 50, Appendix K evaluation models, and demonstrates the continued conformance to the acceptance criteria of 10 CFR 50.46. The results indicated increased peak cladding temperature consequences for EPU are insignificant compared to the large amount by which the results are below the regulatory criteria. Analyses have been performed for operation at both 105% and 120% OLTP and demonstrated to meet applicable limits.

- **Challenges to the Containment**

  The primary criteria of merit for challenges to containment are the maximum containment pressure during the course of LOCA and maximum suppression pool temperature for long-term cooling. The effect of EPU for containment pressure and temperature determined by analyses confirm the acceptability of plant operation at EPU RTP; refer to PUSAR Table 4-1. Containment pressure remains below the structural design. The drywell airspace temperature exceeds the containment structural design basis temperature of 281° F for a short time period, less than one minute, following a design basis accident. This short duration is not sufficient for the average shell temperature to exceed the containment structural design. Further, the accident has been evaluated at EPU conditions to ensure that the capacity of the Hardened Wetwell Vent System will provide sufficient relief capacity. Therefore, the effects of EPU on the conditions that affect the containment dynamic loads are acceptable for EPU operation. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

  The change in short-term containment response is negligible. There is increased post shutdown residual heat associated with EPU; therefore, the containment long-term response changes. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

- **Design Basis Accident Radiological Consequences**

  A full scope Alternative Source Term (AST) analysis performed at EPU conditions, established acceptance criteria consistent with that required by 10 CFR 50.67. The analysis demonstrated that post-accident control room and offsite doses are within regulatory limits. TVA submitted this under a separate license amendment request (ML022200382). The requested amendment was issued September 27, 2004. As part of the issuance of the AST license amendment, NRC established a license condition for only BFN Unit 1. This license condition requires TVA to confirm that the conclusions made in TVA's letter dated September 17, 2004, for the turbine building remain acceptable using seismic demand accelerations based on dynamic seismic analysis prior to the restart of Unit 1. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Transient Analyses

The effects of plant transients are evaluated by investigating a number of postulated disturbances of process variables and malfunctions or failures of equipment. These events are primarily evaluated against the Safety Limit Minimum Critical Power Ratio (SLMCPR). The SLMCPR is determined using NRC-approved methods [see Figure 5-3 of ELTR-2 (Reference 3)], and determined on a cycle-specific basis. As an acceptance criteria in the transient analyses, ensuring that the SLMCPR is not exceeded ensures that for analyzed transients, 99.9% of the rods in the core will not undergo boiling transition. The most limiting transient is slightly more severe when initiated from the EPU RTP level, and results in a slightly (≤ 0.03) larger change in MCPR than that initiated from the current power level. The Operating Limit MCPR is increased appropriately to ensure that the SLMCPR is not infringed upon, if any transient is initiated from the EPU RTP level. Additionally, the limiting transients are analyzed for each specific fuel cycle. EPU does not result in exceeding any licensing acceptance criterion. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Combined Effects

DBAs are postulated using deterministic regulatory criteria to evaluate challenges to the fuel, containment, and off-site radiation dose limits. The off-site and on-site dose evaluations are performed using AST methodology. The DBA producing the highest PCT results in fuel damage bounded by the assumptions used in the off-site and on-site dose evaluation. The DBA producing the maximum containment pressure results in leakage rates to the atmosphere also bounded by the assumptions used in the off-site and on-site dose evaluation. Thus, the doses calculated are conservative compared to the combined effect of the bounding DBA evaluations. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Non-LOCA Radiological Release Accidents

As part of the TVA AST submittal process (ML022200382), plant specific radiological analyses were updated for the applicable BFN Unit 1 events at EPU conditions. The dose consequences for these radiological release events remain within regulatory limits. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Equipment Qualification

BFN Unit 1 equipment and instrumentation have been evaluated under the normal and accident environmental conditions associated with operation at EPU. As part of restart activities, BFN Unit 1 equipment and instrumentation have been evaluated to ensure that applicable environmental qualification criteria have been met. That effort established environmental profiles at EPU conditions, and either validated the existing qualification of BFN Unit 1 equipment and instrumentation, or identified the need for replacement. Any required replacements are being performed as part of ongoing BFN Unit 1 restart activities. The analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Balance-of-Plant

Balance-of-plant (BOP) systems/equipment have been reviewed. BFN Unit 1 plant-specific evaluations justify EPU operation for BOP systems/equipment with some modifications. The PUSAR analyses performed for operation at 120% OLTP bound the requested operation at 105% OLTP.

Individual Design Basis Changes

The BFN Unit 1 PUSAR provided in Enclosure 4 of the license amendment application was not annotated to identify the individual design basis changes that require prior NRC approval. To assist in the regulatory review of the Browns Ferry EPU license amendment request for Unit 1 operation at 105% OLTP, TVA reviewed the application to identify design/licensing bases changes, that if made independent of the 105% OLTP application, might require NRC review and approval in accordance with 10 CFR 50.59. Based on this review, TVA identified several changes potentially falling into this category. TVA requests that these changes be specifically addressed in the Safety Evaluation for the 105% OLTP amendment. These changes and the location of the justification for each are as follow:

SUBMITTAL REFERENCE	DESCRIPTION
PUSAR Section 3.8	Decrease in RCIC operation time utilizing CST reserve volume
PUSAR Section 3.9.1	Increase in shutdown cooling time to achieve 125°F
PUSAR Section 4.1.5	Decrease in relieving capacity of HWWV
PUSAR Section 4.2.5	Change in ECCS NPSH margin/containment overpressure credit
PUSAR Section 4.3	Change in limiting PCT event
PUSAR Section 4.7	Decrease in onsite nitrogen storage requirements
PUSAR Section 6.7.1	Appendix R analyses: • Reduction in time to open 3 MSRVs • Reduction in time to isolate HPCI
PUSAR Section 7.2	Reduction in retention time of condensate in the condenser hotwell

V. REGULATORY SAFETY ANALYSIS

The Tennessee Valley Authority (TVA) is submitting an amendment request to License DPR-33 for BFN Unit 1. This supplement requests revision of the BFN Unit 1 renewed operating license and TS to support safe operation at an increased licensed RTP of 3458 MWt, and to allow credit for containment overpressure in ensuring adequate Net Positive Suction Head (NPSH) for the BFN Unit 1 low pressure Emergency Core Cooling System (ECCS) pumps following a design basis accident. The requested power level is approximately 5% above the current RTP of 3293 MWt. The analysis for containment overpressure at 120% OLTP bounds the need for containment overpressure at 105% OLTP. TVA letters to NRC dated August 4, and August 31, 2006, provided response to RAIs and supporting evaluation, respectively, concerning containment overpressure. The August 4 letter states "The results conclude that adequate

NPSH and containment overpressure are available for the full spectrum of events analyzed.”

No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: **No**.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increased power level, because BFN Unit 1 continues to comply with the regulatory and design basis criteria established for plant equipment. An evaluation of the Boiling Water Reactor probabilistic risk assessments concludes that the calculated core damage frequency does not significantly change due to operation at 105% OLTP. Scram setpoints (equipment settings that initiate automatic plant shutdowns) are established such that there is no significant increase in scram frequency due to operation at 105% OLTP. No new challenges to safety-related equipment result from operation at 105% OLTP.

The probability of Design Basis Accidents occurring is not affected by taking credit for containment overpressure in ensuring adequate NPSH for the BFN Unit 1 low pressure ECCS pumps. NRC Bulletin 96-03 requested that BWR owners implement appropriate measures to minimize the potential clogging of the Emergency Core Cooling System (ECCS) suppression chamber strainers by potential debris generated by a LOCA. TVA installed new, high-capacity passive strainers on BFN Unit 1 of the same design as BFN Units 2 and 3. In addition, TVA's proposed resolution of NRC Bulletin 96-03 for BFN Unit 1 takes credit for containment overpressure to maintain adequate ECCS pump Net Positive Suction Head (NPSH). Containment pressure will increase following a pipe break occurring inside containment. Crediting containment overpressure in the analysis of the consequences of the Loss of Coolant Accident (LOCA) does not affect the precursors for the LOCA, nor does it affect the precursors for any other accident or transient analyzed in Chapter 14 of the BFN Updated Final Safety Analysis Report (UFSAR). Therefore, there is no increase in the probability of any accident previously evaluated.

The changes in consequences of hypothetical accidents, which would occur from 102% of the stretch power uprate reactor thermal power compared to those previously evaluated, are in all cases insignificant. The stretch power uprate accident evaluations do not exceed any of their NRC-approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and are shown to meet the plant's currently licensed regulatory criteria. In the area of core design, for example, the fuel operating limits such as Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Safety Limit Minimum Critical Power Ratio (SLMCPR) are still met, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC. Challenges to fuel (ECCS performance) are evaluated, and shown to continue to meet the criteria of 10 CFR 50.46.

Challenges to the containment have been evaluated at the increased power level, and the containment and its associated cooling systems continue to meet the design and licensing criteria. Radiological release events (accidents) have been evaluated at the increased power level, and shown to be less than the limits of 10 CFR 50.67.

The radiological consequences of the design basis accident are not increased by taking credit for the post-LOCA suppression chamber airspace pressure. The containment will continue to function as designed. This proposed change only takes credit for containment pressure that would exist following a LOCA. Crediting this pressure in ensuring adequate ECCS NPSH will not result in an increase in containment leakage assumed in any analysis.

Therefore, the proposed amendment does not result in a significant increase in consequences or a significant increase in the probability or consequences of any accident previously evaluated.

2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: **No**.

Equipment that could be affected by operation at 105% OLTP has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. Operation at 105% OLTP uses developed technology, and applies it within the capabilities of existing plant safety related equipment in accordance with the regulatory criteria, including NRC approved codes, standards and methods. No new power dependent accidents have been identified.

The BFN Unit 1 TS require revision to implement operation at 105% OLTP. All revisions have been assessed, and it has been determined that the proposed change will not introduce a different accident than that previously evaluated.

The proposed use of the post-LOCA suppression chamber airspace pressure in the calculation of NPSH for the ECCS pumps does not introduce any new modes of plant operation or make physical changes to plant systems. Rather, the post-LOCA suppression chamber airspace pressure is a consequence of the conditions that would exist in the containment following a large pipe break inside containment. The proposed amendment does not introduce new equipment which could create a new or different kind of accident. No new external threats, release pathways, or equipment failure modes are created.

Therefore, the change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) **Does the proposed change involve a significant reduction in a margin of safety?**

Response: **No.**

The calculated loads on all affected structures, systems and components will remain within their design allowables for all design basis event categories. No NRC acceptance criterion is exceeded. Because the BFN Unit 1 configuration and reactions to transients and hypothetical accidents does not result in exceeding the presently approved NRC acceptance limits, operation at 105% OLTP does not involve a significant reduction in a margin of safety.

The post-LOCA suppression chamber airspace pressure is a byproduct of the conditions that will exist in the containment after a line break inside containment. Conservative analyses have been performed that demonstrate that sufficient post-accident suppression chamber airspace pressure will be available to meet the NPSH requirements for the low pressure ECCS pumps. By enabling credit of these conditions for the low pressure ECCS pumps, adequate NPSH margin will be ensured, and accordingly, the ECCS pumps will meet their performance requirements. Therefore, the credit for containment overpressure does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

Applicable Regulatory Requirements

10 CFR 50.36 (c)(2)(ii) Criterion 2, requires that TS LCOs include process variables, design features, and operating restrictions that are initial conditions of design basis accident analysis. The TS ensure that BFN Unit 1 system performance parameters are maintained within the values assumed in the safety analyses. The TS have been justified by analyses performed in accordance with methodology approved for BFN Unit 1 and continue to provide a comparable level of protection as those TS previously issued by the NRC.

Section 50.46 of Title 10 of the Code of Federal Regulations (10 CFR 50.46) requires that licensees design their ECCS systems to meet five criteria, one of which is to provide long-term cooling capability of sufficient duration following a successful system initiation so that the core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The BFN Unit 1 ECCS is designed to meet this criterion, assuming the worst single failure. External operating experience demonstrates that excessive buildup of debris from thermal insulation, corrosion products, and other particulates could prevent the ECCS from providing long-term cooling following a LOCA. Regulatory Guide 1.82, Revision 2, "Water Sources for Long -Term Recirculation Cooling Following a Loss-of-Coolant Accident," provides an acceptable method of ensuring compliance with 10 CFR 50.46. The BFN Unit 1 ECCS design is consistent with 10 CFR 50.46 and the guidance contained in Regulatory Guide 1.82, Revision 2.

In conclusion, based on the considerations discussed previously, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the commission's regulations, and (3) the issuance of the amendment will not be detrimental to the common defense and security of the health and safety of the public.

VI. ENVIRONMENTAL CONSIDERATION

The proposed Renewed Operating License and TS changes required for implementation of EPU meet the requirements for an environmental review as set forth in 10 CFR 51.20, "Criteria For And Identification Of Licensing And Regulatory Actions Requiring Environmental Impact Statements." The Environmental Report provided in Enclosure 2 of the June 28, 2004 (ML041840109) EPU submittal concludes that worker radiation exposures will continue to be significantly less than the limits established by federal regulation. The uprate to 105% OLTP being requested by this supplement is bounded by the EPU Environmental Report.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT 1 – EXTENDED POWER UPRATE (EPU)

MARKED-UP RENEWED OPERATING LICENSE AND TS PAGES

AFFECTED PAGE LIST

The following pages have been revised. On the affected pages, the revised portions have been highlighted. A line has been drawn through the deleted text, and a double underline has been provided for new or revised text.

Renewed License

Page 3

Definitions

Page 1.1-6

Technical Specifications

3.1-25

3.3-6

3.3-7

3.3-34

3.4-4

3.4-8

3.4-30

3.4-31

3.5-6

3.5-13

5.0-20

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~3293~~3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 255, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 234 to Facility Operating License DPR-33, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 234. For SRs that existed prior to Amendment 234, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 234.

1.1 Definitions (continued)

OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 13.10, Refueling Test Program; of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3293~~3458 MWt.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
	Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1.	Once within 8 hours after discovery that SPB concentration is > 9.2% by weight AND 12 hours thereafter
SR 3.1.7.5	Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ ~~186~~203 pounds.	31 days
SR 3.1.7.6	Verify the SLC conditions satisfy the following equation: $\frac{(C)(Q)(E)}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days AND Once within 24 hours after water or boron is added to the solution
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ ~~1275~~1325 psig.	18 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ ~~0.58~~0.66 W + ~~62~~66% RTP and ≤ 120% RTP(b)
c. Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) [~~0. 58~~0.66W + ~~62~~66% - ~~0.58~~0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ ~~1055~~1090 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

SURVEILLANCE REQUIREMENTS

---NOTE---
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level - Low Low, Level 2 Function.	24 hours
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level - Low Low, Level 2: ≥ 471.52 inches above vessel zero; and b. Reactor Steam Dome Pressure - High: ≤ ~~1146.5~~1175 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

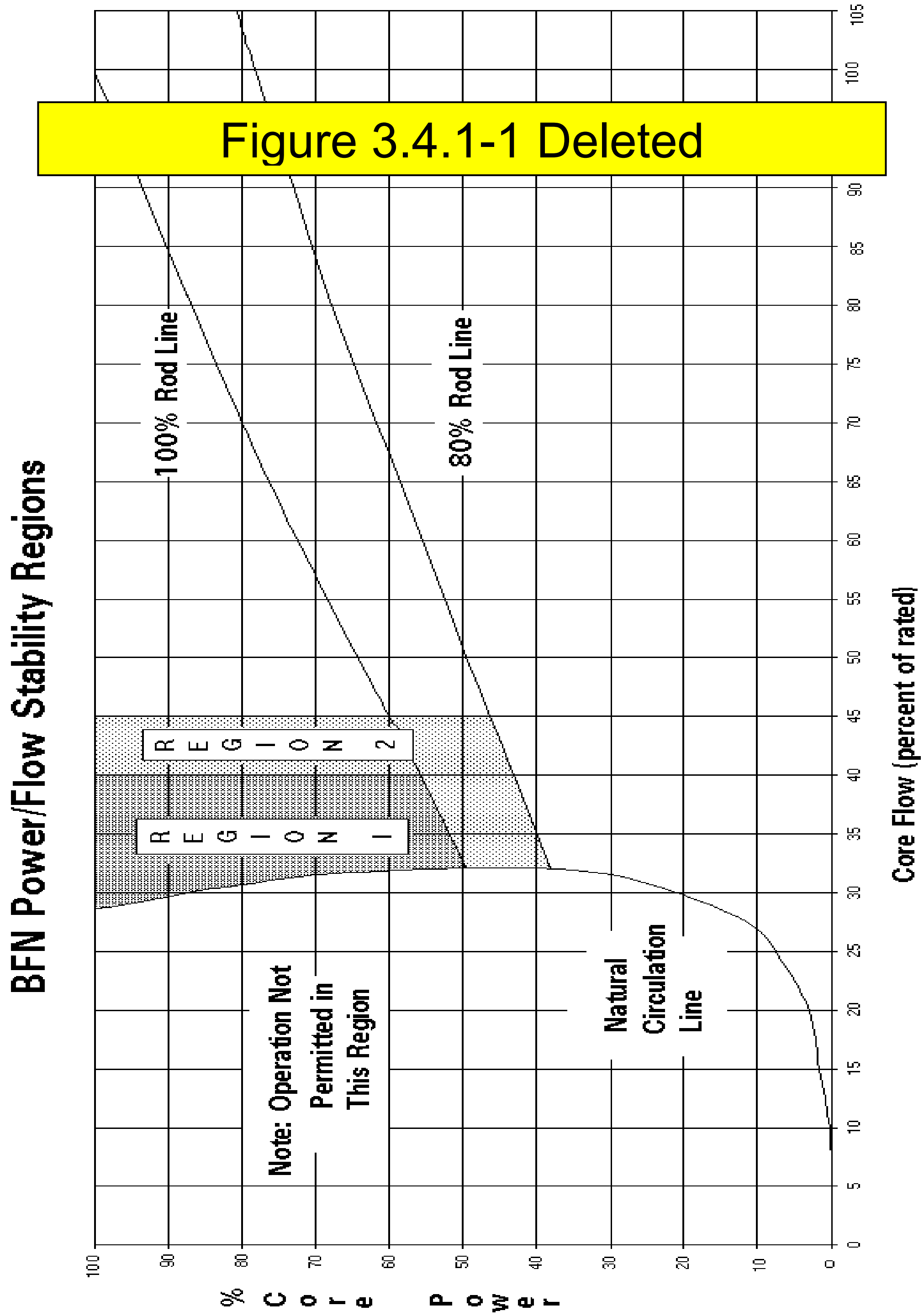

Figure 3.4.1-1
THERMAL POWER VERSUS CORE FLOW STABILITY REGIONS

SURVEILLANCE REQUIREMENTS

<table>
<tr><th colspan="2">SURVEILLANCE</th><th>FREQUENCY</th></tr>
<tr><td>SR 3.4.3.1</td><td>Verify the safety function lift settings of the required 12 S/RVs are within ± 3% of the setpoint as follows:
<table><tr><th>Number of S/RVs</th><th>Setpoint (psig)</th></tr><tr><td>4</td><td>~~1105~~1135</td></tr><tr><td>4</td><td>~~1115~~1145</td></tr><tr><td>5</td><td>~~1125~~1155</td></tr></table>Following testing, lift settings shall be within ± 1%.</td><td>In accordance with the Inservice Testing Program</td></tr>
<tr><td>SR 3.4.3.2</td><td>---------------------------NOTE------------------------
Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.

Verify each required S/RV opens when manually actuated.</td><td>18 months</td></tr>
</table>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ ~~1020~~1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ ~~1020~~1050psig.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.7	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ ~~1010~~1040 and ≥ ~~920~~950 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	18 months
SR 3.5.1.9	-------------------------NOTE------------------------ Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	--------------------------NOTE------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ ~~1010~~1040 psig and ≥ ~~920~~950 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	--------------------------NOTE------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is ~~49.6~~48.5 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT 1 – EXTENDED POWER UPRATE (EPU)

REVISED RENEWED OPERATING LICENSE AND TS PAGES

AFFECTED PAGE LIST

The following pages have been revised. A revision bar has been place in the right hand margin to indicate where changes occur.

Renewed License

Page 3

Definitions

Page 1.1-6

Technical Specifications

3.1-25

3.3-6

3.3-7

3.3-34

3.4-4

3.4-8

3.4-30

3.4-31

3.5-6

3.5-13

5.0-20

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 255, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 234 to Facility Operating License DPR-33, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 234. For SRs that existed prior to Amendment 234, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 234.

1.1 Definitions (continued)

OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 13.10, Refueling Test Program; of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3458 MWt.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
	Verify the concentration and temperature of boron in solution are within the limits of Figure 3.1.7-1.	Once within 8 hours after discovery that SPB concentration is > 9.2% by weight <u>AND</u> 12 hours thereafter
SR 3.1.7.5	Verify the minimum quantity of Boron-10 in the SLC solution tank and available for injection is ≥ 203 pounds.	31 days
SR 3.1.7.6	Verify the SLC conditions satisfy the following equation: $$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(13\text{ wt. }\%)(86\text{ gpm})(19.8\text{ atom}\%)} \geq 1$$ where, C = sodium pentaborate solution concentration (weight percent) Q = pump flow rate (gpm) E = Boron-10 enrichment (atom percent Boron-10)	31 days <u>AND</u> Once within 24 hours after water or boron is added to the solution
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 39 gpm at a discharge pressure ≥ 1325 psig.	18 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 0.66 W + 66% RTP and ≤ 120% RTP(b)
c. Neutron Flux - High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP

(continued)

(c) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(d) [0.66W + 66% - 0.66 - Δ W] RTP when reset for single loop operation per LCO 3.4.1, “Recirculation Loops Operating.”

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[(a)]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[(a)]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

SURVEILLANCE REQUIREMENTS

---NOTE---
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level - Low Low, Level 2 Function.	24 hours
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level - Low Low, Level 2: ≥ 471.52 inches above vessel zero; and b. Reactor Steam Dome Pressure - High: ≤ 1175 psig.	18 months
SR 3.3.4.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

FIGURE 3.4.1-1 Deleted

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift settings of the required 12 S/RVs are within ± 3% of the setpoint as follows: Number of S/RVs — Setpoint (psig) 4 — 1135 4 — 1145 5 — 1155 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program
SR 3.4.3.2	---------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each required S/RV opens when manually actuated.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1050 psig.	12 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.7	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1040 and ≥ 950 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.1.8	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	18 months
SR 3.5.1.9	-------------------------NOTE------------------------ Vessel injection/spray may be excluded. --- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1040 psig and ≥ 950 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48.5 psig. The maximum allowable primary containment leakage rate, L_a, shall be 2% of primary containment air weight per day at P_a.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT 1 – EXTENDED POWER UPRATE (EPU)

CORRELATION BETWEEN SER FOR UNITS 2 AND 3 AMENDMENT FOR 105% OLTP OPERATION AND INFORMATION SUBMITTED FOR BFN UNIT 1 OPERATION AT 120% OLTP

The interim operation of BFN Unit 1 at 105% of original licensed thermal power (OLTP) will operationally be the same as current operations of Units 2 and 3 at the same power level (3458 MWt). Minor equipment differences may exist until the next refueling outages of Units 2 and 3 when plant modifications can be completed. Otherwise, the major operating parameters for all three units should be alike, including thermal power, core and steam flows, heat cycle, and electrical generation. Reactor steam dome pressures should all be the same at rated conditions.

Because of the similarity of Unit 1 to the operating units, TVA prepared the information contained in this enclosure to provide NRC staff reference to previous evaluations conducted in support of the licensing of Units 2 and 3 at 105% OLTP. License amendments for the operation of BFN Units 2 and 3 at 105% OLTP were issued on September 8, 1998. Table E4-1 provides a comparison of NRC’s Safety Evaluation for the 105% OLTP of Units 2 and 3 to the EPU documentation provided for 120% OLTP operation of Unit 1. Where Unit 1 EPU docketed information was updated or superseded, Table E4-1 references the latest TVA submittal, usually as a response to NRC Request for Additional Information questions.

Table E4-2 provides references to EPU Power Uprate Safety Evaluation Report (PUSAR) sections, and other licensing basis changes that support EPU, that are within the scope of the reviews documented in NRC’s September 8, 1998, Safety Evaluation for the 105% power uprate of Units 2 and 3.

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
3.0	Reactor Core and Fuel Performance	05/01/2006	all (no question number)
3.0	Reactor Core and Fuel Performance	09/01/2006	all (no question number)
3.0	Reactor Core and Fuel Performance	03/07/2006	EMCB-A.1
3.0	Reactor Core and Fuel Performance	03/07/2006	EMCB-A.1
3.0	Reactor Core and Fuel Performance	03/07/2006	EMCB-A.2
3.0	Reactor Core and Fuel Performance	03/07/2006	EMCB-A.3
3.0	Reactor Core and Fuel Performance	03/07/2006	EMCB-A.4
3.0	Reactor Core and Fuel Performance	03/07/2006	EMCB-A.5
3.0	Reactor Core and Fuel Performance	12/19/2005	EMEB-B.2
3.0	Reactor Core and Fuel Performance	03/07/2006	IPSB-A.1
3.0	Reactor Core and Fuel Performance	02/23/2005	Request 2
3.0	Reactor Core and Fuel Performance	08/16/2006	SBWB-26
3.0	Reactor Core and Fuel Performance	07/06/2006	SBWB-27
3.0	Reactor Core and Fuel Performance	08/16/2006	SBWB-30
3.0	Reactor Core and Fuel Performance	08/16/2006	SBWB-32
3.0	Reactor Core and Fuel Performance	08/16/2006	SBWB-33
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-35
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-36
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-37
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-38
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-39
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-40
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-41
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-42
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-43
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-44
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-45
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-46

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-47
3.0	Reactor Core and Fuel Performance	08/18/2006	SBWB-48
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.1
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.2
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.5
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.6
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.7
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.9
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.10
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.11
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.12
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.17
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.18
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.20
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.22
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.23
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.24
3.0	Reactor Core and Fuel Performance	03/07/2006	SRXB-A.25
3.2	Reactor Coolant Systems Connected Systems	05/01/2006	all (no question number)
3.2	Reactor Coolant Systems Connected Systems	09/01/2006	all (no question number)
3.2	Reactor Coolant Systems Connected Systems	03/07/2006	EMCB-A.5
3.2	Reactor Coolant Systems Connected Systems	06/06/2005	Request 1
3.2	Reactor Coolant System Connected Systems	02/23/2005	Request 2
3.2	Reactor Coolant Systems Connected Systems	06/06/2005	Request 2
3.2	Reactor Coolant Systems Connected Systems	06/06/2005	Request 3

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
3.2	Reactor Coolant Systems Connected Systems	06/06/2005	Request 4
3.2	Reactor Coolant Systems Connected Systems	06/06/2005	Request 7
3.2	Reactor Coolant Systems Connected Systems	06/06/2005	Request 8
3.2	Reactor Coolant Systems Connected Systems	06/23/2006	SPLB-A.1
3.2	Reactor Coolant Systems Connected Systems	06/23/2006	SPLB-A.2
3.2	Reactor Coolant Systems Connected Systems	06/23/2006	SPLB-A.3
3.2	Reactor Coolant Systems Connected Systems	03/07/2006	SRXB-A.16
3.2	Reactor Coolant Systems Connected Systems	03/07/2006	SRXB-A.21
4.1	Containment System Performance	03/07/2006	ACVB.1
4.1	Containment System Performance	03/07/2006	ACVB.6
4.1	Containment System Performance	03/07/2006	ACVB.7
4.1	Containment System Performance	03/07/2006	ACVB.8
4.1	Containment System Performance	03/07/2006	ACVB.9
4.1	Containment System Performance	03/07/2006	ACVB.14
4.1	Containment System Performance	03/23/2006	ACVB.17
4.1	Containment System Performance	03/23/2006	ACVB.18
4.1	Containment System Performance	03/23/2006	ACVB.26
4.1	Containment System Performance	03/07/2006	ACVB.34
4.1	Containment System Performance	03/07/2006	ACVB.35
4.1	Containment System Performance	07/21/2006	ACVB.38/36
4.1	Containment System Performance	07/21/2006	ACVB.43/41
4.1	Containment System Performance	07/21/2006	ACVB.44/42
4.1	Containment System Performance	07/21/2006	ACVB.46/44
4.1	Containment System Performance	08/18/2006	ACVB-59/57
4.1	Containment System Performance	08/18/2006	ACVB-60/58

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
4.1	Containment System Performance	09/15/2006	ACVB-65
4.1	Containment System Performance	09/15/2006	ACVB-67
4.1	Containment System Performance	09/15/2006	APLA-25
4.1	Containment System Performance	03/23/2006	SPSB-A.11
4.1	Containment System Performance	03/07/2006	SRXB-A.19
4.2	Containment Dynamic Loads	03/07/2006	ACVB.6
4.2	Containment Dynamic Loads	03/07/2006	ACVB.10
4.2	Containment Dynamic Loads	03/07/2006	ACVB.11
4.2	Containment Dynamic Loads	03/07/2006	ACVB.15
4.2	Containment Dynamic Loads	03/07/2006	ACVB.16
4.2	Containment Dynamic Loads	06/06/2005	Request 5
4.3	Containment Isolation	07/21/2006	ACVB.43/41
4.3	Containment Isolation	07/21/2006	ACVB.44/42
4.3	Containment Isolation	07/21/2006	ACVB.45/43
4.3	Containment Isolation	12/19/2005	EMEB-B.3
4.4	Combustible Gas Control in Containment	03/07/2006	ACVB.34
4.4	Combustible Gas Control in Containment	03/07/2006	ACVB.36
4.5	ECCS	03/07/2006	ACVB.12
4.5	ECCS	03/23/2006	ACVB.17
4.5	ECCS	03/23/2006	ACVB.18
4.5	ECCS	03/23/2006	ACVB.26
4.5	ECCS	03/23/2006	SPSB-A.11
4.5	ECCS	03/07/2006	SRXB-A.17
4.6	ECCS Performance	03/07/2006	ACVB.12
4.6	ECCS Performance	07/21/2006	ACVB.46/44
4.6	ECCS Performance	09/15/2006	ACVB.67
4.6	ECCS Performance	02/23/2005	Request 2
4.6	ECCS Performance	07/06/2006	SBWB-27
4.6	ECCS Performance	09/01/2006	SBWB-49

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
4.6	ECCS Performance	02/28/2006	SPSB-A.11
4.6	ECCS Performance	03/07/2006	SRXB-A.14
4.6	ECCS Performance	03/07/2006	SRXB-A.19
4.6	ECCS Performance	03/07/2006	SRXB-A.22
4.6	ECCS Performance	03/07/2006	SRXB-A.25
5.0	Instrumentation	06/06/2005	Request 5
5.0	Instrumentation	03/07/2006	SRXB-A.4
6.2	Water Systems	02/23/2005	Request 5.d
6.2	Water Systems	12/19/2005	SPLB-A.5, Round 2
6.2	Water Systems	12/19/2005	SPLB-A.6, Round 2
6.2	Water Systems	12/19/2005	SPLB-A.7, Round 2
6.2	Water Systems	12/19/2005	SPLB-A.8, Round 2
6.5	Ultimate Heat Sink	03/07/2006	SPLB-A.8, Round 3
6.6	Standby Liquid Control	02/23/2005	Request 2
6.7	Power Dependent HVAC	03/07/2006	ACVB.5
6.8	Fire Protection	07/21/2006	APLA.24/26
6.8	Fire Protection	06/06/2005	Request 6
6.8	Fire Protection	03/07/2006	SPLB-B.1
6.8	Fire Protection	03/07/2006	SPLB-B.2
6.8	Fire Protection	12/19/2005	SPLB-B.4
6.8	Fire Protection	12/19/2005	SPLB-B.5
6.8	Fire Protection	12/19/2005	SPLB-B.6
7.1	Turbine Generator	03/07/2006	SPLB-A.5, Round 3
7.1	Turbine Generator	03/07/2006	SPLB-A.7, Round 3
7.1	Turbine Generator	03/07/2006	SRXB-A.8
7.1	Turbine Generator	03/07/2006	SRXB-A.13
7.2	Miscellaneous Power Conversion Systems	03/07/2006	DORL. 1
8.0	Radiological Issues	12/19/2005	IPSB-B.2
8.1	CREVS	03/07/2006	ACVB.1

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
8.1	CREVS	03/07/2006	ACVB.2
8.2	Liquid Radwaste	03/07/2006	SPLB-A.9
8.3	Gaseous Radwaste	03/07/2006	ACVB.1
8.3	Gaseous Radwaste	03/07/2006	ACVB.3
8.3	Gaseous Radwaste	03/07/2006	ACVB.4
8.3	Gaseous Radwaste	03/07/2006	SPLB-A.10
8.4	Radiation Sources in the Core and Coolant	12/19/2005	IPSB-B.3
8.4	Radiation Sources in the Core and Coolant	12/19/2005	IPSB-B.4
8.4	Radiation Sources in the Core and Coolant	12/19/2005	IPSB-B.6
8.5	Radiation Levels	04/13/2006	IPSB-B.1
8.5	Radiation Levels	12/19/2005	IPSB-B.7
8.5	Radiation Levels	04/13/2006	IPSB-B.8
8.5	Radiation Levels	12/19/2005	IPSB-B.9
8.5	Radiation Levels	12/19/2005	IPSB-B.11
8.5	Radiation Levels	12/19/2005	IPSB-B.12
8.6	Design Basis Accidents (Dose Info)	03/07/2006	ACVB.13
8.6	Design Basis Accidents (Dose Info)	02/23/2005	Request 1.b
8.6	Design Basis Accidents (Dose Info)	02/23/2005	Request 5.c(3)
8.6	Design Basis Accidents (Dose Info)	03/07/2006	SRXB-A.9
8.6	Design Basis Accidents (Dose Info)	03/07/2006	SRXB-A.25
9.0	Reactor Safety Performance Features	05/01/2006	all (no question number)
9.0	Reactor Safety Performance Features	09/01/2006	all (no question number)
9.0	Reactor Safety Performance Features	07/06/2006	SBWB-27

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
9.1	Reactor Transients	03/07/2006	DORL. 1
9.1	Reactor Transients	03/07/2006	IPSB-A.10
9.1	Reactor Transients	02/23/2005	Request 1.c
9.1	Reactor Transients	02/23/2005	Request 2
9.1	Reactor Transients	02/23/2005	Request 3
9.1	Reactor Transients	08/16/2006	SBWB-34
9.1	Reactor Transients	03/07/2006	SRXB-A.3
9.1	Reactor Transients	03/07/2006	SRXB-A.4
9.1	Reactor Transients	03/07/2006	SRXB-A.5
9.1	Reactor Transients	03/07/2006	SRXB-A.8
9.1	Reactor Transients	03/07/2006	SRXB-A.9
9.1	Reactor Transients	03/07/2006	SRXB-A.21
9.1	Reactor Transients	03/07/2006	SRXB-A.25
9.2	Special Events	02/23/2005	Request 5.e
9.2	Special Events	03/07/2006	SRXB-A.15
9.2	Special Events	03/07/2006	SRXB-A.25
10.0	HELB, MELB, Mechanical EQ	07/26/2006	EEMB.45
10.0	HELB, MELB, Mechanical EQ	03/07/2006	EMCB-A.1
10.0	HELB, MELB, Mechanical EQ	12/19/2005	EMEB-B.14
10.0	HELB, MELB, Mech EQ	02/23/2005	Request 5.a
10.0	HELB, MELB, Mechanical EQ	03/07/2006	SPLB-A.6 Round 3
11.0	Electrical Issues	06/15/2006	EEEB.8
11.0	Electrical Issues	06/15/2006	EEEB.9
11.0	Electrical Issues	06/15/2006	EEEB.10
11.0	Electrical Issues	06/15/2006	EEEB.11
11.0	Electrical Issues	07/26/2006	EEEB.12
11.0	Electrical Issues	07/26/2006	EEEB.13
11.0	Electrical Issues	07/26/2006	EEEB.14
11.0	Electrical Issues	12/19/2005	EEIB-B.1.a
11.0	Electrical Issues	12/19/2005	EEIB-B.1.b

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		**120% Uprate RAI Questions**	
Section Number	Title	Letter Date	Question Number
11.0	Electrical Issues	12/19/2005	EEIB-B.1.c
11.0	Electrical Issues	12/19/2005	EEIB-B.1.d
11.0	Electrical Issues	12/19/2005	EEIB-B.1.e
11.0	Electrical Issues	12/19/2005	EEIB-B.2
11.0	Electrical Issues	12/19/2005	EEIB-B.3
11.0	Electrical Issues	12/19/2005	EEIB-B.4
11.0	Electrical Issues	12/19/2005	EEIB-B.5
11.0	Electrical Issues	12/19/2005	EEIB-B.6
11.0	Electrical Issues	12/19/2005	EEIB-B.7
11.0	Electrical Issues	07/26/2006	EEMB.45
11.0	Electrical Issues	02/23/2005	Request 1.b
11.0	Electrical Issues	06/15/2006	SBPB.14
12.0	Mechanical Issues	06/15/2006	CVIB.6
12.0	Mechanical Issues	06/15/2006	CVIB.7
12.0	Mechanical Issues	07/26/2006	EEMB.38
12.0	Mechanical Issues	07/26/2006	EEMB.39
12.0	Mechanical Issues	07/26/2006	EEMB.40
12.0	Mechanical Issues	07/26/2006	EEMB.41
12.0	Mechanical Issues	07/26/2006	EEMB.42
12.0	Mechanical Issues	07/26/2006	EEMB.43
12.0	Mechanical Issues	07/26/2006	EEMB.44
12.0	Mechanical Issues	07/26/2006	EEMB.46
12.0	Mechanical Issues	07/26/2006	EEMB.47
12.0	Mechanical Issues	07/26/2006	EEMB.48
12.0	Mechanical Issues	12/19/2005	EMEB-B.4
12.0	Mechanical Issues	12/19/2005	EMEB-B.5
12.0	Mechanical Issues	12/19/2005	EMEB-B.8
14.0	Human Factors	06/15/2006	IOLB.1
14.0	Human Factors	06/15/2006	IOLB.2
14.0	Human Factors	12/19/2005	IROB-B-1

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
14.0	Human Factors	03/07/2006	IROB-B-1
14.0	Human Factors	12/19/2005	IROB-B-2
14.0	Human Factors	12/19/2005	IROB-B-3
14.0	Human Factors	12/19/2005	IROB-B-4
14.0	Human Factors	08/18/2006	SBWB-45
14.0	Human Factors	08/18/2006	SBWB-46
14.0	Human Factors	12/19/2005	SPSB-A.7
15.0	Test Control	04/25/2005	all (no question number)
15.0	Test Control	07/26/2006	EEEB.15
15.0	Test Control	02/01/2006	EMEB-B.6
15.0	Test Control	03/07/2006	Enclosure 3
15.0	Test Control	03/07/2006	IPSB-A.1
15.0	Test Control	03/07/2006	IPSB-A.2
15.0	Test Control	03/07/2006	IPSB-A.3
15.0	Test Control	03/07/2006	IPSB-A.4
15.0	Test Control	03/07/2006	IPSB-A.5
15.0	Test Control	03/07/2006	IPSB-A.6
15.0	Test Control	03/07/2006	IPSB-A.7
15.0	Test Control	03/07/2006	IPSB-A.8
15.0	Test Control	03/07/2006	IPSB-A.9
15.0	Test Control	03/07/2006	IPSB-A.10
15.0	Test Control	12/19/2005	IPSB-B.10
15.0	Test Control	03/07/2006	SPLB-A.11
15.0	Test Control	03/07/2006	SPLB-A.12
15.0	Test Control	03/07/2006	SPLB-A.13
18.0	Environmental Consideration	03/07/2006	REBB.1
18.0	Environmental Consideration	03/07/2006	REBB.2
18.0	Environmental Consideration	03/07/2006	REBB.3
18.0	Environmental Consideration	03/07/2006	REBB.4

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		**120% Uprate RAI Questions**	
Section Number	Title	Letter Date	Question Number
18.0	Environmental Consideration	03/07/2006	REBB.5
	COP / NPSH *	08/23/2006	ACVB.17
	COP / NPSH *	03/23/2006	ACVB.18
	COP / NPSH *	03/23/2006	ACVB.18
	COP / NPSH *	03/07/2006	ACVB.19
	COP / NPSH *	03/07/2006	ACVB.20
	COP / NPSH *	03/07/2006	ACVB.21
	COP / NPSH *	03/07/2006	ACVB.22
	COP / NPSH *	03/07/2006	ACVB.23
	COP / NPSH *	03/07/2006	ACVB.24
	COP / NPSH *	03/07/2006	ACVB.25
	COP / NPSH *	03/07/2006	ACVB.26
	COP / NPSH *	03/23/2006	ACVB.26
	COP / NPSH *	03/07/2006	ACVB.27
	COP / NPSH *	03/07/2006	ACVB.28
	COP / NPSH *	03/07/2006	ACVB.29
	COP / NPSH *	03/07/2006	ACVB.30
	COP / NPSH *	03/07/2006	ACVB.31
	COP / NPSH *	08/31/2006	ACVB.32 and Calculations MDQ0999970046 and MDQ099920060011
	COP / NPSH *	03/07/2006	ACVB.33
	COP / NPSH *	08/04/2006	ACVB.37/35
	COP / NPSH *	08/04/2006	ACVB.39/37
	COP / NPSH *	08/04/2006	ACVB.40/38
	COP / NPSH *	08/04/2006	ACVB.41/39
	COP / NPSH *	08/04/2006	ACVB.42/40
	COP / NPSH *	07/21/2006	ACVB.43/41
	COP / NPSH *	07/21/2006	ACVB.44/42
	COP / NPSH *	07/21/2006	ACVB.45/43

Table E4-1

Matrix of 105% Uprate Safety Evaluation Section Versus 120% Uprate RAI Questions

105% Power Uprate Safety Evaluation		120% Uprate RAI Questions	
Section Number	Title	Letter Date	Question Number
	COP / NPSH *	07/21/2006	ACVB.46/44
	COP / NPSH *	08/04/2006	ACVB.47/45
	COP / NPSH *	07/21/2006	ACVB.48/46
	COP / NPSH *	08/04/2006	ACVB.49/47
	COP / NPSH *	07/21/2006	ACVB.50/48
	COP / NPSH *	07/21/2006	ACVB.51/49
	COP / NPSH *	07/21/2006	ACVB.52/50
	COP / NPSH *	07/22/2006	ACVB.53/51
	COP / NPSH *	08/04/2006	ACVB.54/52
	COP / NPSH *	07/24/2006	ACVB.55/53
	COP / NPSH *	08/04/2006	ACVB.56/54 (Figure ACVB-56 was replaced by Figure 7.14 of Enclosure 2 of 08/31/2006 letter.)
	COP / NPSH *	07/24/2006	ACVB.57/55
	COP / NPSH *	08/04/2006	ACVB.58/56
	COP / NPSH *	08/18/2006	ACVB-61/59
	COP / NPSH *	08/18/2006	ACVB-62
	COP / NPSH *	08/18/2006	ACVB-63
	COP / NPSH *	08/18/2006	ACVB-64
	COP / NPSH *	07/21/2006	APLA.22/24
	COP / NPSH *	08/04/2006	APLA.23/25
	COP / NPSH *	07/21/2006	APLA.24/26
	COP / NPSH *	09/15/2006	APLA-26
	COP / NPSH *	03/23/2006	SPSB-A.11

* COP / NPSH = Containment Overpressure / Net Positive Suction Head

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
3.0.1	Fuel Design and Operation	• Amendment Request TS-455	• Safety Limit Minimum Critical Power Ratio
3.0.2	Thermal Limits Assessment	• U1 PUSAR Section 2.2 • Amendment Request TS-455	• Thermal Limit Assessment • Safety Limit Minimum Critical Power Ratio
3.0.3	Power/Flow Operating Map	• U1 PUSAR Section 2.3	• Reactivity Characteristics
3.0.4	Stability	• Amendment Request TS-443	• Oscillation Power Range Monitor
3.0.5	Reactivity Control - Control Rod Drives (CRD) and CRD Hydraulic System	• U1 PUSAR Section 2.5	• Reactivity Control
3.2.1	Nuclear System Pressure Relief	• U1 PUSAR Section 3.1	• Nuclear System Pressure Relief

[1] See attached list of references.

[2] NRC Safety Evaluation for the power uprate (105% OLTP) of BFN units 2 and 3, September 8, 1998

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		**Source / Title**	
3.2.2	Code Overpressure Protection	• U1 PUSAR Section 3.2	• Reactor Overpressure Protection Analysis
3.2.3	Reactor Vessel Fracture Toughness	• U1 PUSAR Section 3.3 • Amendment Request TS-428	• Reactor Vessel and Internals • Update of Pressure-Temperature (P-T) Curves
3.2.4	Reactor Recirculation System	• U1 PUSAR Section 3.4 • Amendment Request TS-455 • U1 PUSAR Section 3.5	• Reactor Recirculation System • Safety Limit Minimum Critical Power Ratio • Reactor Coolant Pressure Boundary Piping
3.2.5	Main Steam Isolation Valves (MSIVs)	• U1 PUSAR Section 3.7	• Main Steam Isolation Valves
3.2.6	Reactor Core Isolation Cooling System (RCIC)	• U1 PUSAR Section 3.8	• Reactor Core Isolation Cooling
3.2.7	Residual Heat Removal System	• U1 PUSAR Section 3.9	• Residual Heat Removal System

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
3.2.7.a	Shutdown Cooling Mode	• U1 PUSAR Section 3.9.1	• Shutdown Cooling Mode
3.2.7.b	Suppression Pool Cooling and Containment Spray Modes	• U1 PUSAR Section 3.9.2 • U1 PUSAR Section 3.9.3	• Suppression Pool Cooling Mode • Containment Spray Cooling Mode
3.2.7.c	RHR System - Supplemental Fuel Pool Cooling Mode	• U1 PUSAR Section 3.9.4	• Supplemental Spent Fuel Pool Cooling
3.2.8	Reactor Water Cleanup (RWCU) System	• U1 PUSAR Section 3.10	• Reactor Water Cleanup System
4.1	Containment System Performance	• U1 PUSAR Section 4.1	• Containment System Performance
4.1.1	Containment Pressure and Temperature Response - Bulk Pool Temperature	• U1 PUSAR Section 4.1.1	• Containment Pressure and Temperature Response
4.1.1.a.1	Long-Term Suppression Pool Temperature Response	• U1 PUSAR Section 4.1.1.1	• Long-Term Suppression Pool Temperature Response

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
4.1.1.a.2	Local Suppression Pool Temperature with Main Steam Relief Valve (MSRV) Discharge	• U1 PUSAR Section 4.1.1.1.b	• Local Pool Temperature with MSRV Discharge
4.1.1.b	Containment Gas Temperature Response	• U1 PUSAR Section 4.1.1.2	• Short-Term Gas Temperature Response
4.1.1.c	Short-Term Containment Pressure Response	• U1 PUSAR Section 4.1.1.3	• Short-Term Containment Pressure Response
4.2.1	LOCA Containment Dynamic Loads	• U1 PUSAR Section 4.1.2 • U1 PUSAR Section Tables 4-1 and 5-1 • U1 PUSAR Section 4.1.2.1	• Containment Dynamic Loads
4.2.2	Main Steam Relief Valve (MSRV) Containment Dynamic Loads	• U1 PUSAR Section 4.1.2.2	• Main Steam Relief Valve Loads

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
4.2.3	Subcompartment Pressurization	• U1 PUSAR Section 4.1.2.3	• Subcompartment Pressurization
4.3	Containment Isolation	• U1 PUSAR Section 4.1.3	• Containment Isolation
		• U1 PUSAR Section 4.1.4	• Generic Letter 89-10 Program
		• U1 PUSAR Section 4.1.5	• Generic Letter 89-16
		• U1 PUSAR Section 4.1.6	• Generic Letter 95-07
		• U1 PUSAR Section 4.1.7	• Generic Letter 96-06
4.4	Combustible Gas Control In Containment	• U1 PUSAR Section 4.7	• Post-LOCA Combustible Gas Control
4.5	Emergency Core Cooling Systems (ECCSs)	• U1 PUSAR Section 4.2	• Emergency Core Cooling Systems

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
4.5.1	High-Pressure Core Injection System (HPCI)	• U1 PUSAR Section 4.2.1 • U1 PUSAR Section 4.3	• High Pressure Coolant Injection System • Emergency Core Cooling System Performance
4.5.2	Low-Pressure Core Injection System (LPCI mode of RHR)	• U1 PUSAR Section 4.2.2 • U1 PUSAR Section 4.3	• Low Pressure Coolant Injection • Emergency Core Cooling System Performance
4.5.3	Core Spray (CS) System	• U1 PUSAR Section 4.2.3 • U1 PUSAR Section 4.3	• Core Spray System • Emergency Core Cooling System Performance
4.5.4	Automatic Depressurization Systems (ADSs)	• U1 PUSAR Section 4.2.4	• Automatic Depressurization System
4.6	ECCS Performance Evaluation	• U1 PUSAR Section 4.3	• Emergency Core Cooling System Performance

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
4.7.1	Main Control Room Atmosphere Control System (CRACS)	• U1 PUSAR Section 4.4 • Generic Letter 2003-01 Response	• Main Control Room Atmosphere Control System
4.7.2	Emergency Cooling Water System and Auxiliary Systems	• U1 PUSAR Section 6.4	Water Systems
5.0.1	TS Table 3.3.1.1-1, Reactor Protection System Instrumentation, Function 2.b, Average Power Range Monitors, Flow Biased Simulated Thermal Power - High. Decrease the allowable value from 0.66W+71 percent rated thermal power (RTP) to 0.66W+66 percent RTP.	• U1 PUSAR Section 5.1.2 • Amendment Request TS-430	• Neutron Monitoring System

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
5.0.2	TS Table 3.3.1.1-1, Reactor Protection System Instrumentation, Function 3, Reactor Vessel Steam Dome Pressure - High. Increase the allowable value from 1055 psig to 1090 psig.	• U1 PUSAR Section 5.3.1	• High Pressure Scram
5.0.3	SR 3.3.4.2.3.b, ATWS-RPT Instrumentation, Reactor Steam Dome Pressure - High. Increase the allowable value from 1146.5 psig to 1175 psig.	• U1 PUSAR Section 5.3.2	• High Pressure ATWS Recirculation Pump Trip
5.0.4	SR 3.4.3.1, Main Steam Relief Valve Setpoints. Increase each setpoint by 30 psig.	• U1 PUSAR Section 5.3.3	• Main Steam Relief Valve

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
5.0.5	Unit 2 Only, TS Table 3.3.6.1-1, Function 5.1, RWCU System Isolation, Main Steam Valve Vault Area Temperature. Decrease the allowable temperature from 201°F to 188°F.	• Amendment Request TS-447	• Calibration Interval Extension for HPCI and RCIC Temperature Switches
6.1	SFP Cooling System	• U1 PUSAR Section 6.3	• Fuel Pool
6.2	Water Systems	• U1 PUSAR Section 6.4	• Water Systems
6.2.1	Raw Water Systems	• U1 PUSAR Section 6.4.1	• Service Water Systems
6.2.1.1	Safety-Related Loads	• U1 PUSAR Section 6.4.1.1	• Safety Related Loads
6.2.2	Nonsafety-Related Loads	• U1 PUSAR Section 6.4.3 • U1 PUSAR Section 6.4.4	• Reactor Building Closed Cooling Water • Raw Cooling Water System

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
6.2.3	Main Condenser, Circulating Water, and Normal Heat Sink Systems	• U1 PUSAR Section 6.4.2	• Main Condenser / Circulation Water / Normal Heat Sink Performance
6.3	Reactor Building Closed-Cooling Water (RBCCW) System	• U1 PUSAR Section 6.4.3	• Reactor Building Closed Cooling Water
6.4	Raw Cooling Water System	• U1 PUSAR Section 6.4.4	• Raw Cooling Water System
6.5	Ultimate Heat Sink	• U1 PUSAR Section 6.4.5	• Ultimate Heat Sink
6.6	Standby Liquid Control System (SLCS)	• U1 PUSAR Section 6.5	• Standby Liquid Control System
6.7	Power Dependent Reactor Building and Plant HVAC Systems	• U1 PUSAR Section 6.6	• Power Dependent HVAC
6.8	Fire Protection	• U1 PUSAR Section 6.7	• Fire Protection
6.9	Systems Not Impacted By Power Uprate	• U1 PUSAR Section 6.8	• Systems Not Impacted by EPU

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
7.1	Turbine-Generator	• U1 PUSAR Section 7.1	• Turbine-Generator
7.2	Miscellaneous Power Conversion Systems	• U1 PUSAR Section 7.2 • U1 PUSAR Section 7.3 • U1 PUSAR Section 7.4	• Condenser and Steam Jet Air Ejectors • Turbine Steam Bypass • Feedwater and Condensate Systems
8.0	Radiological Issues	• U1 PUSAR Section 9.2 • U1 PUSAR Section 8.5	• Design Basis Accidents • Radiation Levels
8.1	Control Room Emergency Ventilation System (CREVS)	• U1 PUSAR Section 9.2 • Amendment Request TS-405	• Design Basis Accidents
8.2	Liquid Waste Management	• U1 PUSAR Section 8.1	• Liquid and Solid Waste Management
8.3	Gaseous Waste Management	• U1 PUSAR Section 8.2	• Gaseous Waste Management

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
8.4	Radiation Sources in the Core and the Coolant	• U1 PUSAR Section 8.3	• Radiation Sources in the Reactor Core
8.5	Radiation Levels	• U1 PUSAR Section 8.5	• Radiation Levels
8.6	Design Basis Accidents (DBAs)	• U1 PUSAR Section 8.5.3	• Post Accident
9.1	Reactor Transients	• U1 PUSAR Section 9.1	• Reactor Transients
9.2	Special Events	• U1 PUSAR Section 9.3.1 • U1 PUSAR Section 9.3.2	• Anticipated Transients Without Scram • Station Blackout

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
10.0	High and Moderate Energy Line Breaks and EQ of Mechanical Components	• U1 PUSAR Section 10.3.1.2 • U1 PUSAR Section 10.2 • U1 PUSAR Section 10.3.2 • U1 PUSAR Section 10.3.3	• Outside Containment • Moderate Energy Line Break • Mechanical Equipment with Non-Metallic Components • Mechanical Component Design Qualification
11.1	Main Turbine Generator and its Auxiliary Equipment	• U1 PUSAR Section 7.1	• Turbine Generator
11.2	Electrical Power Systems	• U1 PUSAR Section 6.1.1 • U1 PUSAR Section 9.3.2	• Offsite Power System • Station Blackout
11.3	Safety-Related Electrical Equipment Qualification	• U1 PUSAR Section 10.3.1	• Electrical Equipment

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
12.0	Mechanical Issues	• U1 PUSAR Section 3.3	• Reactor Vessel and Internals
12.1	RPV and Internals	• U1 PUSAR Section 3.3.2 • U1 PUSAR Section 3.3.3 • U1 PUSAR Section 3.3.4	• Reactor Vessel Structural Evaluation • Reactor Internal Pressure Differences • Reactor Internals Structural Evaluation
12.2	Control Rod Drive Mechanism (CRDM)	• U1 PUSAR Section 2.5.3	• Control Rod Drive Integrity Assessment
12.3	Reactor Coolant Piping and Components	• U1 PUSAR Section 3.5	• Reactor Coolant Pressure Boundary Piping
12.4	Equipment Seismic and Dynamic Qualification	• U1 PUSAR Section 3.5	• Reactor Coolant Pressure Boundary Piping

Table E4-2: BFN Unit 1 5% Uprate Supporting Document Matrix

5% Power Uprate Safety Evaluation		Comparable Supporting Source/Documentation[1]	
SE[2] Section Number / Title		Source / Title	
12.5	Safety-Related Safety/Relief Valves and Power-Operated Valves	• U1 PUSAR Section 3.1	• Nuclear System Pressure Relief
		• U1 PUSAR Section 3.2	• Reactor Overpressure Protection Analysis
		• U1 PUSAR Section 4.1.6	• Generic Letter 95-07
		• U1 PUSAR Section 4.1.4	• Generic Letter 89-10
		• U1 PUSAR Section 4.1.7	• Generic Letter 96-06
13.0	Emergency Operating Instructions	• U1 PUSAR Section 10.6	• Operator Training and Human Factors
14.0	Human Factors Issues	• U1 PUSAR Section 10.6	• Operator Training and Human Factors
15.0	Test Control	• U1 PUSAR Section 10.4	• Testing

REFERENCES

1. NRC Letter to TVA, "Issuance of Amendments RE: Power Uprate - Browns Ferry Plant, Units 2 and 3 - (TAC Nos. M99711 and M99712)," dated September 8, 1998
2. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Proposed Technical Specifications (TS) Change TS - 431 - Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 28, 2004
3. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1- Proposed Technical Specifications (TS) Change TS - 431 - Request for License Amendment - Extended Power Uprate (EPU) Operation Probabilistic Safety Assessment (PSA) Update," dated August 23, 2004
4. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Response to NRC Acceptance Review Letter and Request for Additional Information Related to Technical Specifications (TS) - Change No. TS-431 - Request for Extended Power Uprate Operation (TAC No. MC3812)," dated February 23, 2005
5. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 - Response to NRC's Request for Additional Information Related to Technical Specifications (TS) Change No. TS-431– Request for Extended Power Uprate Operation (TAC No. MC3812)," dated April 25, 2005
6. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 – Response to NRC's Request for Additional Information Related to Technical Specifications (TS) Change No. TS - 431 – Request for License Amendment – Extended Power Uprate (EPU) Operation (TAC No. MC3812)," dated June 6, 2005
7. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 - Response to NRC Round 2 Requests for Additional Information Related to Technical Specifications (TS) Change No. TS-431 – Request for Extended Power Uprate Operation (TAC No. MC3812)," dated December 19, 2005
8. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Unit 1 – Response to NRC Request Emeb-B.6 From NRC Round 2 Requests for Additional Information Related to Technical Specifications (TS) Change No. TS-431 - Request for Extended Power Uprate (TAC No. MC3812)," dated February 1, 2006

9. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-431 - Response to Request for Additional Information SPSB-A.11 Regarding Extended Power Uprate - Credit for Net Positive Suction Head (TAC No. MC3812)," dated February 28, 2006

10. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Response to NRC Round 3 Requests for Additional Information Related to Technical Specifications (TS) Change No. TS-431 - Request for Extended Power Uprate Operation (TAC No. MC3812)," dated March 7, 2006

11. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 – Technical Specifications (TS) Change TS-431 - Request for Extended Power Uprate Operation– Response to NRC Requests for Additional Information Regarding Credit for Containment Overpressure (TAC No. MC3812)," dated March 23, 2006

12. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) Change Nos. TS-418 and TS-431 - Request for Extended Power Uprate Operation - Response to NRC Request for Additional Information Of March 7, 2006, Regarding Probabilistic Risk Analyses (TAC Nos. MC3812, MC3743, and MC3744)," dated March 31, 2006

13. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 - Technical Specifications (TS) Change Nos. TS-418 and TS-431 – Extended Power Uprate (EPU) Operation - Revised Responses to NRC Round 2 Requests for Additional Information - (TAC Nos. MC3812, MC3743, and MC3744)," dated April 13, 2006

14. TVA Letter to NRC, "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 5 Request for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated June 15, 2006

15. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 - Technical Specifications (TS) Changes TS-431 and TS-418 – Extended Power Uprate (EPU) - Revised Response to NRC Round 2 Requests for Additional Information Splb-A.1, Splb-A.2, and Splb-A.3 - (TAC Nos. MC3812, MC3743, and MC3744)," dated June 23, 2006

16. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) Changes TS-431 and TS-418 - Extended Power Uprate (EPU) - Replacement Cooling tower (TAC Nos. MC3812, MC3743, and MC3744)," dated June 27, 2006

17. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-431 - Extended Power Uprate (EPU) - Response to NRC Round 6 Request for Additional Information On GE Methods (TAC No. MC3812)," dated July 6, 2006

18. TVA Letter to NRC, "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 6 Request for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated July 21, 2006

19. TVA Letter to NRC, "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 7 Requests for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated July 26, 2006

20. TVA Letter to NRC, "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 6 Request for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated August 4, 2006

21. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-431 - Extended Power Uprate (EPU) - Supplemental Response to NRC Round 6 Request for Additional Information (RAI) SBWB-26 and SBSW-30 and Partial Response to Round 8 On Fuel Analysis Methods (TAC No. MC3812)," dated August 16, 2006

22. TVA Letter to NRC, "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 8 Request for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated August 18, 2006

23. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 - Technical Specifications (TS) Changes TS-431 and TS-418 – Extended Power Uprate (EPU) – Replacement Documentation (TAC Nos. MC3812, MC3743, and MC3744)," dated August 31, 2006

24. TVA Letter to NRC, "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 9 Request for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated September 1, 2006

25. TVA Letter to NRC, "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 9 Requests for Additional Information (TAC Nos. MC3812, MC3743, and MC3744)," dated September 15, 2006

26. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - License Amendment - Alternative Source Term," dated 07/31/2002 [TS-405]

27. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) Unit 1 - Technical Specifications (TS) Change 430 - Power Range Neutron Monitor Upgrade With Implementation of Average Power Ranger Monitor and Rod Block Monitor Technical Specification Improvements and Maximum Extended Load Line Limit Analyses," dated November 10, 2003

28. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) Unit 1 - Technical Specifications (TS) Change TS 428 - Update of Pressure - Temperature (P-T) Curves," dated December 6, 2004

29. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 Response to NRC Generic Letter (GL) 2003-01 - Control Room Habitability," dated December 8, 2003

30. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) Change TS-447 - Extension of Channel Calibration Surveillance Requirement Performance Frequency and Allowable Value Revision," dated August 16, 2004

31. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) Unit 1 - Technical Specifications (TS) Change TS-443 - Oscillation Power Range Monitor (OPRM)," dated January 6, 2006

32. TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-455 - Safety Limit Minimum Critical Power Ratio (SLMCPR) - Cycle 7 Operation," dated May 1, 2006

33. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Technical Specifications (TS) Change TS-455 - Safety Limit Minimum Critical Power Ratio (SLMCPR) - Cycle 7 Operation - Response to Request for Additional Information (RAI) (TAC No. MD1721)," dated September 1, 2006

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

BROWNS FERRY NUCLEAR PLANT (BFN) – UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT – EXTENDED POWER UPRATE (EPU)

CHANGE CONSIDERATIONS OF INTEREST TO THE NRC

Based on discussions with the NRC staff, the following topics were identified to be of specific interest:

1. **NRC Item of Interest**

   Provide a no significant hazards consideration similar to that used for the Unit 2 and 3 Power Uprate application.

   TVA Response

   Section V of Enclosure 1 of this submittal addresses the no significant hazards consideration.

2. **NRC Item of Interest**

   Provide list of other TS required prior to issuance of Safety Evaluation Report for PU.

   TVA Response

   This information was provided in the cover letter for this submittal.

3. NRC Item of Interest

   Provide information concerning the simulator configuration for BFN Unit 1 operation at 105% OLTP.

   TVA Response

   TVA has two control room simulators at the BFN site. One simulator is designated as the Unit 1 simulator and has been updated to coincide with operation at 105% OLTP. This simulator reflects the TS-related values requested herein for 105% OLTP operation. This simulator will reflect the other modifications required for 120% OLTP operation (e.g., larger feedwater pumps). Thus, this simulator will reflect the actual BFN Unit 1 plant conditions at restart. Operating crews will receive simulator training prior to operation of the unit at 105% OLTP and again at EPU conditions. Simulator and classroom training will be completed during the last training phase prior to restart and will include normal operating procedure actions required to achieve power uprate, power ascension testing, and physical plant changes as modeled in the simulator. TVA will install changes on the simulator prior to Unit 1 restart to support training of all operating crews. Setpoint changes corresponding to 105% OLTP will also be reflected on the simulator prior to completion of training.

Acceptance testing of the simulator will be conducted to benchmark its performance and will be implemented in accordance with ANSI/ANS 3.5. The performance of the simulator will be validated against the expected 105% OLTP response and then against operating data collected during power uprate implementation and startup testing. Based on the results of the validation, TVA will make any necessary adjustments to the simulator model.

4. **NRC Item of Interest**

Address Generic Letter 2003-01 for the Control Room Emergency Ventilation System (CREVS).

TVA Response

TVA responded to GL 2003-01 via a letter to the NRC dated December 8, 2003 (ML033430322). In summary, the BFN Units 1, 2, and 3 design basis and licensing basis are in compliance with the applicable regulatory requirements. The plant is constructed and maintained in accordance with its design, and the testing performed in accordance with the BFN Technical Specifications (TS) and their bases is adequate to demonstrate this compliance and material condition. No changes to this response are required for BFN Unit 1 operation at 105% OLTP.

5. **NRC Item of Interest**

Address PUSAR discussion on peak containment internal pressure for 120%.

TVA Response

The evaluation for item 14 of Table E1-1 of Enclosure 1 of this submittal satisfies this request.

6. **NRC Item of Interest**

RWCU, HPCI, and RCIC temperature element TS numbers are based on pressure not power. Define changes driven by pressure change / power uprate.

TVA Response

As stated in the cover letter for this submittal, BFN Units 1, 2, and 3 TS Change 447 - Extension of Channel Calibration Surveillance Requirement Performance Frequency and Allowable Value Revision, is one of the changes that TVA requested to be issued prior to or concurrent with the BFN Unit 1 PU amendment request.

The Unit 1 PUSAR included with the request for Unit 1 EPU contains the following:

"5.2.3 Leak Detection System

The instrument setpoints associated with primary system leak detection have been evaluated with respect to the slightly higher operating steam flow, steam pressure and FW temperature for EPU. Each of the systems, where leak detection potentially could be affected by the EPU, is addressed below.

MS Valve Vault and Tunnel Temperature Based Leak Detection:

EPU increases the RWCU, FW and MS pressures and temperatures, which potentially affects the MSL tunnel and MSVV LDSs. The pipe break mass and energy calculations have been performed to establish Analytical Limits (ALs) that support the current TS values. Thus the temperature based leak detection is not adversely affected and the ALs in the TS do not require a change.

RCIC System Temperature Based Leak Detection:

EPU increases the RCIC steam pressure and temperature. The pipe break mass and energy calculations have been performed to establish ALs that support the current TS values. Thus the RCIC temperature based leak detection is not adversely affected and the ALs in the TS do not require a change.

HPCI System Temperature Based Leak Detection:

EPU increases the HPCI steam pressure and temperature. The pipe break mass and energy calculations have been performed to establish ALs that support the current TS values. Thus the HPCI temperature based leak detection is not adversely affected and the ALs in the TS do not require a change."

The temperature switch setpoints are based on high energy line break (HELB) analyses. The HELB analyses are controlled by reactor pressure rather than power level; thus, the setpoints are the same for operation at either 105% OLTP or 120% OLTP.

7. **NRC Item of Interest**

Discuss basis for Pressure-Temperature curves for the reactor pressure vessel (RPV).

TVA Response

PUSAR Section 3.3.1, "Reactor Vessel Fracture Toughness," states "… The TS P-T curves will be revised considering the increases in shifts affecting the beltline portion of the curves. These curves will be provided in a separate licensing submittal. …"

BFN Unit 1 TS Change 428 – Update of Pressure-Temperature (P-T) Curves was submitted to NRC by TVA letter dated December 6, 2004. Per this submittal, "… These fluence values conservatively assume operation over the entire analyzed period at an extended power uprate condition of 3952 MWt. …"

In a letter dated July 26, 2006, the NRC issued BFN Unit 1 TS amendment 256 in response to this request.

Operation at 3458 MWt rather than 3952 MWt will slow down neutron-related effects, and thus result in the existing curves being conservative.

8. **NRC Item of Interest**

Discuss affect on the material surveillance program for the RPV.

TVA Response

TVA has committed to implement the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) for Unit 1 upon NRC acceptance of BWRVIP-116. As stated in TVA's March 7, 2006, response to RAI question EMCB-A.2, upon NRC approval and updating of BWRVIP documentation, TVA plans to request a license amendment for Unit 1 to implement the ISP. Transition to the ISP is not affected by the reactor power level.

9. **NRC Item of Interest**

Address question on (15) Test Control (starts on page 52) of the Safety Evaluation Report for PU for Units 2 and 3.

TVA Response

The details of the testing to be performed are provided in the cover letter for this submittal. They are summarized below.

TVA will conduct a startup test program for Unit 1 that includes:

- The BFN Extended Power Uprate Startup Test Program includes testing from less than 90% OLTP to 120% OLTP.
- Power ascension testing planned for BFN Unit 1.
- TVA will perform the applicable portions of the Restart Test Program planned to occur up to and including operation at 105% OLTP. The remainder of the program above 105% OLTP will be conducted following NRC approval of operation at 120% OLTP.

As part of the two-step approach to EPU, TVA will perform two large transient tests:

- A large transient test that simulates the rejection of generator load will be performed at 105% OLTP.
- An MSIV Closure with valve position scram large transient test will be performed at 115% to 120% OLTP.

10. **NRC Item of Interest**

Address Environmental Qualification (EQ).

TVA Response

In TVA's June 15, 2006, reply to RAI question EEEB.10, additional information was provided regarding the environmental qualification (EQ) of electrical equipment located inside containment. Because Unit 1 has been in an extended shutdown, the EQ program will be re-established to meet the current EQ program applicable to Units 2 and 3, which is designed for EPU conditions. The program descriptions for EPU are applicable for lower power conditions, including those present at 105% OLTP. Therefore, the EQ program for 105% OLTP is no different than the program for full EPU conditions. As discussed above, the pipe break detection temperature setpoints are based on reactor pressure rather than reactor power. The high main steam line flow setpoint will be calibrated based on rated steam flow at 105% OLTP.

Item 2.H. of the BFN Unit 1 Renewed Operating License states "The licensee must complete the thirteen (13) Unit 1 restart commitments that are discussed in Appendix F of the license renewal application, dated December 31, 2003, as supplemented by letters dated January 31, 2005, March 2, and April 21, 2006. Completion of these activities must be met prior to power operation of Unit 1."

Appendix F.4 is "Implement Environmental Qualification Program.

11. **NRC Item of Interest**

Address Station Blackout (SBO).

TVA Response

The evaluations performed for station blackout (SBO) of Unit 1 at full EPU conditions bounds an SBO scenario assuming 105% OLTP. PUSAR section 9.3.2 describes the evaluation conducted and the conformance to 10 CFR 50.63 for EPU conditions. With a lower initial power level and less decay heat than assumed for EPU, the evaluation conservatively encompasses the conditions for 105% OLTP.

TVA's December 19, 2005, response to EPU Round 2 RAI question EEIB-B.6 is unchanged for BFN Unit 1 reactor power operation at 105% OLTP. For 105% OLTP conditions, there is no impact on the coping duration category or alternate AC power availability.

12. **NRC Item of Interest**

Address Anticipated Transient Without a Scram (ATWS).

TVA Response

ATWS is described in PUSAR section 9.3.1 and in the responses to various RAI questions. As noted in the PUSAR, there are no substantive differences in plant response, operator actions, or boron absorption requirements between 105% OLTP and full EPU. Because EPU does not change the control rod line on the power-flow operating domain, the response to an ATWS event is very similar for 105% OLTP and EPU conditions.

13. **NRC Item of Interest**

Provide a chart of process flow variables.

TVA Response

Requested information follows:

BROWNS FERRY UNIT 1 OPERATING PARAMETERS

Parameter	Current* Licensed Value	105% OLTP Value
Thermal Power (MWt)	3293	3458
Vessel Steam Flow (Mlb/hr)**	13.37	14.15***
Full Power Core Flow Range		
• (Mlb/hr)	76.9 to 107.6	83.0 to 107.6
• % Rated	75 to 105	81 to 105
Maximum Nominal Dome Pressure (psia)	1020	1050
Maximum Nominal Dome Temperature (°F)	547	551
Pressure at upstream side of turbine stop valve (psia)	988	988***
Full Power Feedwater		
• Flow (Mlb/hr)	13.32	14.1
• Temperature (°F)	376.6	381.7***
Core Inlet Enthalpy (Btu/lb)****	521.6	524.7***

* Based on current reactor heat balance

** At normal Feedwater heating

*** Approximate

**** At 100% core flow condition

14. NRC Item of Interest

Please provide a Table of the following fuel thermal limits: OLMCPR, SLMCPR, MAPLHGR, and LHGR.

TVA Response

Requested information follows:

BFN UNIT 1 FUEL THERMAL LIMITS

PARAMETER	LIMIT (1)
Operating Limit Minimum Critical Power Ratio (OLMCPR)	Provided in Section 11 of SRLR (2)
Safety Limit Minimum Critical Power Ratio (SLMCPR)	• 1.09 (two loop operation) • 1.11 (single loop operation)
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)	Provided in Tables 16.3-1 through 16.3 13 of SRLR (2)
Linear Heat Generation Rate (LHGR)	• GE13: 14.4 kw/ft (3) • GE14: 13.4 kw/ft (3)

(1) The values are based on the design and operation of a full EPU (120% OLTP) operating cycle. Core analyses for 105% OLTP may result in different limits.

(2) Provided in TVA to NRC letter, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Supplemental Reload Licensing Report - Cycle 7 Operation," dated May 15, 2006.

(3) initial value - decreases with exposure

15. **NRC Item of Interest**

Provide commitment that the following will be closed prior to BFN Unit 1 Restart:

a. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance";

b. Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves";

c. Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors"; and

d. Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions."

TVA Response

a. The following information was included on page E1-70 of TVA to NRC letter, "Browns Ferry Nuclear Plant (BFN) - Status of Unit 1 Restart Issues, Revision 7," Dated August 22, 2006:

"… Based on test requirements and system configurations, it would be necessary to perform differential pressure testing on some motor operated valves during the power ascension test program. Consequently, TVA committed to complete the required testing within 30 days following the completion of the power ascension test program. …"

b. The following commitment / status were noted on page E1-80 of TVA to NRC letter, "Browns Ferry Nuclear Plant (BFN) - Status of Unit 1 Restart Issues, Revision 7," Dated August 22, 2006:

"Unit 1 will be evaluated for the requirements of Generic Letter 95-07 prior to restart."

"The safety related power operated gate valves in Unit 1 have been reviewed for potential susceptibility to the pressure locking and thermal binding phenomenon. There is one High Pressure Coolant Injection valve in Unit 1 which is susceptible to thermal binding. Prior to restart, this valve will be replaced with a double disc valve of similar design as Units 2 and 3. Two Core Spray minimum flow valves in Unit 1 will be replaced with double disc valves prior to Unit 1 restart. In addition, five safety related power operated gate valves will be modified prior to Unit 1 restart to preclude the potential for pressure locking. The reactor side disc face of these five valves will be modified by drilling a hole in the disc face into the cavity between the disc faces to avoid pressure locking. …"

"The valves discussed above will be modified or replaced prior to restart."

c. TVA has completed this item. The following information was included on page E1-21 of TVA to NRC letter, "Browns Ferry Nuclear Plant (BFN) - Status of Unit 1 Restart Issues, Revision 7," Dated August 22, 2006:

   "TVA has installed new, high capacity passive strainers on Unit 1, which are of the same design as on Units 2 and 3."

d. The following information was included on page E1-84 of TVA to NRC letter, "Browns Ferry Nuclear Plant (BFN) - Status of Unit 1 Restart Issues, Revision 7," Dated August 22, 2006:

   "TVA will address Generic Letter 96-06 prior to restart."

16. <u>NRC Item of Interest</u>

Provide assurance that the Alternate Decay Heat Removal System will be operable prior to BFN Unit 1 Restart.

<u>TVA Response</u>

Item 2.H. of the BFN Unit 1 Renewed Operating License states "The licensee must complete the thirteen (13) Unit 1 restart commitments that are discussed in Appendix F of the license renewal application, dated December 31, 2003, as supplemented by letters dated January 31, 2005, March 2, and April 21, 2006. Completion of these activities must be met prior to power operation of Unit 1."

Appendix F.11 states "Modify Auxiliary Decay Heat Removal System to serve Unit 1."

17. **NRC Item of Interest**

Provide a Table of containment accident pressures for operation of BFN Unit 1 at 105% OLTP. A table like Table 4-1, page 4-14, of the PUSAR is requested.

TVA Response

The following Table 4-1 from the Unit 1 PUSAR provides peak containment parameters for EPU conditions which bound 105% OLTP operations:

Table 4-1

BFN Unit 1 Containment Performance Analysis Results

Parameter	OLTP(1) (Historical)	EPU (Current Methods)	Limit
Peak Drywell Pressure (psig)	49.6	48.5(2)	56
Peak Drywell Temperature (°F)(3)	294	295.2(2)	340/281
Peak Bulk Pool Temperature (°F)	170	187.3(4, 5)	281
Peak Wetwell Pressure (psig)	27	30.5	56

(1) Unit 1 UFSAR Section 14.11.3 values

(2) LAMB mass and energy release data used as input to M3CPT

(3) The acceptance limit for drywell airspace temperature is 340°F, while the shell design value is 281°F. The listed peak values are for airspace temperature.

(4) Uses ANS/ANSI 5.1 (+2σ uncertainty) decay heat model.

(5) Service water temperature of 95°F

18. **<u>NRC Item of Interest</u>**

Address the schedule for completing the evaluation requested by RAI EEMB-160. Then as part of the RAI response, answer the question.

<u>TVA Response</u>

TVA noted in its July 26, 2006, response to Round 7 RAIs that Unit 1 piping systems and supports would be evaluated and modified prior to restart. This work is scheduled for completion in February 2007.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 1

BROWNS FERRY NUCLEAR PLANT (BFN) - UNIT 1 - TECHNICAL SPECIFICATIONS (TS) CHANGE TS-431, SUPPLEMENT 1 - EXTENDED POWER UPRATE (EPU)

NEW REGULATORY COMMITMENTS

As part of the two-step approach to EPU, TVA will perform two large transient tests:

1. A large transient test that simulates the rejection of generator load will be completed within 30 days of reaching 105% OLTP.

2. An MSIV Closure with valve position scram large transient test will be performed within 30 days of reaching 115% to 120% OLTP.