Contains proprietary information Per 10 CFR 2.390

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

TVA-BFN-TS-418
June 6, 2005

10 CFR 50.90

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) – UNITS 2 AND 3 - RESPONSE TO NRC's REQUEST FOR ADDITIONAL INFORMATION RELATED TO TECHNICAL SPECIFICATIONS (TS) CHANGE NO. TS-418 – REQUEST FOR LICENSE AMENDMENT – EXTENDED POWER UPRATE (EPU) OPERATION (TAC NOS MC3743 and MC3744)

This letter contains the additional information requested by the NRC Staff in its December 29, 2004 letter (Reference 1). This reply is in support of TVA's license amendment request TS-418 submitted on June 25, 2004 (Reference 2). TS-418 requested a license amendment and associated TS changes to support an increase in the reactor thermal power level to 3952 MWt, an approximate 15 percent increase in thermal power from the current licensed power level.

Enclosure 1 to this letter provides TVA's response to the questions transmitted by Reference 1.

Some of the information in Enclosure 1 is proprietary to Framatome Advanced Nuclear Power (ANP). Framatome ANP requests that the proprietary information in the enclosure be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4), 10 CFR 2.390(a)(4) and 10 CFR 2.390(b)(1). An affidavit supporting this request is included in Enclosure 1. A non-proprietary version of this response is contained in Enclosure 2.

As detailed in the response to NRC Request 7 contained in Enclosures 1 and 2 of this submittal, the Framatome ANP Appendix R analysis for the case involving

APO1

spurious High Pressure Coolant Injection (HPCI) System initiation has been revised. Previously, the Framatome ANP analysis indicated that HPCI operation be secured within five minutes during an Appendix R event. That analysis was revised to more accurately model the effect of HPCI injection on vessel depressurization and subsequent vessel level rise. The results of the revised analysis indicate that operators have six minutes to secure HPCI operation during an Appendix R event, which is consistent with the analysis performed for GE fuel.

TVA is providing similar information regarding the Unit 1 EPU application in a separate submittal. There are no new regulatory commitments associated with this submittal. If you have any questions concerning this letter, please telephone me at (256) 729-2636.

Pursuant to 28 U.S.G. § 1796 (1994), I declare under penalty of perjury that the forgoing is true and correct.

Executed on this 6th day of June, 2005.

Sincerely,

William D. Crouch
Acting Manager of Licensing
and Industry Affairs

Enclosures:

1. Reply to Request for Additional Information for BFN Units 2 and 3 Extended Power Uprate Application (Proprietary Version).

2. Reply to Request for Additional Information for BFN Units 2 and 3 Extended Power Uprate Application (Non-Proprietary Version).

References:

1. NRC letter to TVA "Browns Ferry Nuclear Plant, Units 2 and 3 - Request for Additional Information Regarding Extended Power Uprate, (TAC Nos. MC3743 and MC3744) (TS-418)," dated December 29, 2004.

2. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 - Proposed Technical Specifications (TS) Change TS - 418 - Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 25, 2004.

(Via NRC Electronic Distribution)

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ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNITS 2 AND 3
REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR
EXTENDED POWER UPRATE APPLICATION
(Non-Proprietary Version)

See Attached:

- Reply To Request For Additional Information For BFN Units 2 and 3 Extended Power Uprate Application (Non-Proprietary Version)

Reply to Request for Additional Information for BFN Units 2 And 3 Extended Power Uprate Application

By letter dated June 25, 2004 (Reference 1), TVA submitted for NRC review, an application pursuant to 10 CFR 50.90 requesting amendments to the Units 2 and 3 operating licenses that increase the maximum power level to 3952 MWt. As part of the Staff's review of TVA's application, they have identified questions concerning the application. By letter dated December 29, 2004 (Reference 2) the NRC transmitted the questions to TVA. The following provides TVA's response to the transmitted questions.

Some of the information in Enclosure 1 is proprietary to Framatome Advanced Nuclear Power (ANP). Framatome ANP requests that the proprietary information in the enclosure be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4), 10 CFR 2.390(a)(4) and 10 CFR 2.390(b)(1). An affidavit supporting this request is included in Enclosure 1. A non-proprietary version of this response is contained in this Enclosure.

The information that is proprietary to Framatome ANP is also marked in the same manner as in Enclosure 5 of the license amendment application. That is, a bracket is at the beginning and end of each proprietary statement.

NRC Request 1

This submittal represents the first extended power uprate (EPU) application with a 100 percent Framatome ATRIUM-10 core on Unit 1. General Electric (GE) submitted EPU Licensing Topical Report (ELTR)-1 and ELTR-2, well in advance and different evaluation models for transient, accident, anticipated transient without scram, stability, etc., and approved them for EPU application. In Framatome Uprated [*sic*] Safety Analysis Report EMF-2982 (FUSAR), it is indicated that the evaluation models given in Table 1-3 of FUSAR are valid for EPU application. Provide an evaluation explaining the assumptions, limitations, restrictions, etc., in the models and discuss why the applications of the models are valid for EPU.

TVA Reply 1

TVA's current plans do not include use of Framatome ANP fuel in the Unit 1 core. Hence, the following discussion is only applicable to Units 2 and 3.

Framatome ANP reload licensing analyses are performed to ensure that all fuel design and operating limits are satisfied for the limiting assembly in the core. The first step in determining the applicability of current licensing methods to EPU conditions was a review of current BWR topical reports to identify SER restrictions on the Framatome ANP BWR analytical methodology. This review identified that there are no SER restrictions on power level for the Framatome ANP topical reports. The review also indicated that there are no SER restrictions on the parameters most impacted by the increased power level; steam flow, feedwater flow, jet pump M-ratio, and core average void fraction. The second step consisted

of an evaluation of the differences between current core and reactor conditions and those experienced under EPU conditions to determine any challenges to the validity of the models.

When the reactor power is uprated, the increase in core thermal power is compensated to a large extent by a decrease in the limiting assembly radial power factor. This decrease in the limiting assembly radial power factor is necessary since the thermal operating limits (MCPR, MAPLHGR and LHGR) are fairly insensitive to the increase in core thermal power. From this fundamental constraint, the following observations may be made about the EPU operating conditions:

- The reduction in the hot assembly radial peaking factor leads to a more uniform radial power distribution and consequently a more uniform core flow distribution. The net result of this is less flow starvation of the hottest assemblies, i.e., a more uniform flow distribution across the core.
- With the flatter radial power distribution, more assemblies and fuel rods are near thermal operating limits.
- From a reactor heat balance perspective, there will be higher steam flow and feedwater flow rates.
- With an increase in average assembly power in the reactor the core pressure drop will increase slightly resulting in a decrease in the jet pump M-ratio for a given core flow rate.
- The core average void fraction will increase.

Based on these fundamental characteristics of power uprate, each of the major analysis domains (thermal-hydraulics, core neutronics, transient analysis, Loss of Coolant Accident (LOCA) analysis, stability analysis, and special events) can be assessed to determine any challenges to EPU application.

Thermal-hydraulics

Framatome ANP assembly thermal-hydraulic methods are qualified and validated against full-scale tests in the KATHY test facility in Karlstein, Germany. The KATHY tests are used to characterize the assembly two-phase pressure drop and critical heat flux (CHF) performance. This allows the hydraulic models to be verified for Framatome ANP fuel designs over a wide range of hydraulic conditions prototypic of reactor conditions. The standard matrix of test conditions for KATHY is compared to reactor conditions in Figure 1. This figure illustrates that the test conditions bound both uprated and non-uprated assembly conditions. In addition, the key physical phenomena (e.g., heat flux, fluid quality and assembly flows) for uprated conditions are within the scope of current reactor experience. This similarity of assembly conditions is further enforced in Framatome ANP analysis methodologies by the imposition of SPCB CHF correlation limits and therefore, both current designs and uprated designs must remain within the same parameter

space. Since the bundle operating conditions for EPU are within the envelope of hydraulic test data used for model qualification and operating experience, the hydraulic models used to compute the core flow distribution and local void content remain valid.

Core Neutronics

The Framatome ANP neutronic methodologies are characterized by technically rigorous treatment of phenomena and are very well benchmarked (>100 cycles of operation plus gamma scan data for ATRIUM-10). Some of the recent operating experience is tabulated in Tables 1 and 2. These tables present the reactor operating conditions and in particular the average and hot assembly powers for both US and European applications. As can be seen from this information, the average and peak bundle powers in this experience base exceed those associated with the power uprate for BFN. The increased steam flow from power uprate comes from increased power in normally lower power assemblies in the core running at higher power levels. High powered assemblies in uprated cores will be subject to the same LHGR, MAPLHGR, MCPR, and cold shutdown margin limits and restrictions as high powered assemblies in non-uprated cores. This similarity of operating conditions between current and uprate conditions assures that the neutronic methods used to compute the nodal reactivity and power distributions remain valid for uprate conditions. Consequently, the neutronic characteristics computed by the steady-state simulator and used in safety analysis remain valid.

Transient Analysis

The phenomena of primary interest for limiting transients in BWRs are void fraction/quality relationships, determination of CHF, pressure drop, reactivity feedbacks and heat transfer correlations. One fundamental validation of the core hydraulic solution is separate effects testing against Karlstein transient CHF measurements. The transient benchmark to time of dryout for prototypic load reject without bypass (LRNB) and Pump Trip transients encompass the transient integration of the continuity equations (including the void-quality closure relation), heat transfer and determination of CHF. Typical benchmarks to Karlstein (Figure 2) illustrate that the transient hydraulic solution and application of SPCB result in conservative predictions of the time of dryout.

Outside of the core, the system simulation relies primarily on solutions of the basic conservation equations and equations of state. While there are changes to the feedwater flow rate and jet pump M-ratio associated with power uprate, the most significant change is the steam flow rate and the associated impact on the steamline dynamics for pressurization events. The models associated with predicting the pressure wave, however, are general and have no limitation within the range of variation associated with power uprate.

The reactivity feedbacks are validated by a variety of means including initial qualification of advanced fuel design lattice calculations to Monte Carlo results as required by SER restrictions (Reference 3), steady-state monitoring of reactor

operation (power distributions and eigenvalue) and the Peach Bottom 2 turbine trip benchmarks that exhibited a minimum of 5% conservatism in the calculation of integral power.

From these qualifications and the observation that the nodal hydraulic conditions are expected to be within the current operating experience, the transient methods remain valid.

LOCA Analysis

The impact of power uprate on LOCA analysis is primarily associated with the increase in decay heat levels in the core. Decay heat is well understood and does not require code modifications to address.

Small Break LOCA (SBLOCA) analysis is more sensitive to higher decay heat levels. This is the case because of the amount of liquid which remains in the reactor after the blowdown period for SBLOCA. A large amount of the original reactor inventory is lost through the break during a large break LOCA. SBLOCA cases do not lose as much inventory. The amount of liquid which is not lost during the blowdown period is affected by the amount of decay heat in the system.

LOCA results are determined primarily from the hot assembly initial stored energy and primary system liquid inventories. The stored energy is not expected to change significantly and inventory differences which might result due to power uprate do not change the capability of the codes to model LOCA.

This is proven by the fact that a break spectrum is required to perform a LOCA analysis and therefore, LOCA analysis tools are required to manage differences in reactor inventories at the end of blowdown caused by differences in break sizes. LOCA analysis methods remain valid for EPU conditions.

Stability Analysis

The flatter radial power profile induced by the power uprate will tend to decrease the first azimuthal eigenvalue separation and result in slightly higher regional decay ratios. These effects are computed by STAIF as it directly computes the channel, global and regional decay ratio and does not rely on a correlation to protect the regional mode.

STAIF has been benchmarked against full assembly tests to validate the channel hydraulics from a decay ratio of approximately 0.4 to limit cycles. These benchmarks include prototypic ATRIUM-10 assemblies. From a reactor perspective, STAIF is benchmarked to both global and regional reactor data as late as 1998, and therefore includes current reactor cycle and fuel design elements. This strong benchmarking qualification and the direct computation of the regional mode assure that the impact of the flatter core design for power uprate will be reflected in the stability analysis.

Special Events

Framatome ANP performed an anticipated transient without scram (ATWS) analysis to demonstrate compliance with the peak pressurization criteria which occurs very early in the transient. The early system response during an ATWS event is essentially the same as a transient event and the same code is used to calculate the system response. The impact of EPU operation on the ATWS peak pressure analysis is the same as in the transient analysis system response discussed earlier.

The Appendix R analysis is performed using the approved LOCA analysis codes. Similar to LOCA events, the impact of EPU on Appendix R analysis is primarily an increase in decay heat in the core. Decay heat is well understood and does not require any code modifications for EPU. Use of the Appendix K heat transfer correlations and logic is conservative for Appendix R calculations.

Table 1
CASMO-4/MICROBURN-B2 Operating Experience

Reactor	Reactor Size, #FA	Power, MWt (%Uprated)	Ave. Bundle Power, MWt/FA	Peak Bundle Power, MWt/FA	Cycles Calc. w/ MB2	Fuel/ Cycle Licensing*	Comments
GER-1	592	2575 (0.0)	4.4	7.2	8	X	
GER-2	592	2575 (0.0)	4.4	7.4	13	X	
GER-3	532	2292 (0.0)	4.3	7.3	11	(X)	
GER-4	840	3690 (0.0)	4.4	7.5	17	X	
FIN-1	500	2500 (15.7)	5.0	8.0	11	X	For 3 cycles oper.
SWE-1	444	1800 (5.9)	4.1	7.3	11	X	
SWE-2	676	2928 (8.0)	4.3	7.4	8	(X)	
SWE-3/ SWE-4	700	3300 (9.3)	4.7	8.0	8/3	(X)/(X)	
GER-5, 6	784	3840 (0.0)	4.9	8.1	24	(X)	
SP-1	624	3237 (11.9)	5.2	7.8	3	(X)	
SWZ-1	648	3600 (14.7)	5.6	8.6	9	(X)	For 1 cycle oper.
SWE-5	648	2500 (10.1)	3.9	6.9	10	(X)	
US-1	624	3091 (6.7)	5.0	7.7	6	X	
US-2	800	3898 (1.7)	4.9	7.7	6	X	
US-3	764	3489 (5.0)	4.6	7.2	3	X	
				Total	>150		
Browns Ferry 2/3	764	3952 (20.0)	5.2	7.3		none	Equilibrium cycle study

* (x)=currently fuel licensing only (Europe).

Table 2
CASMO-4/MICROBURN-B2 Other Planned Uprate Support

Reactor	Reactor Size, #FA	Power, MWt (%Uprated)	Ave. Bundle Power, MWt/FA	Peak Bundle Power, MWt/FA	Fuel/ Cycle Licensing*	Comments
SWE-2	676	3253 (20.0)	4.8	7.8	(X)	
GER-5, 6	784	4000 (4.2)	5.1	8.1	(X)	License applied for
SWE-3/ SWE-4	700	3896 (29.0)	5.6	8.6	(X)/(X)	
Browns Ferry 2/3	764	3952 (20.0)	5.2	7.3	none	Equilibrium cycle study

* (X)=currently fuel licensing only (Europe).

Figure 1
Comparison of Karlstein Two-Phase Pressure Drop Test Matrix and Typical Reactor Conditions.

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Figure 2
Typical Hydraulic Benchmarks to Karlstein Transient Simulations

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NRC Request 2

Provide a discussion explaining why the reactor coolant pressure boundary (RCPB) piping materials are not affected by the power uprate.

TVA Reply 2

Evaluation of the effect of changes due to EPU operation in system flows, temperature, and pressure for the RCPB and balance of plant piping were discussed in Enclosure 4, Sections 3.5 and 3.11, of the license amendment application, respectively. The impact of operation at EPU conditions on system materials was previously addressed generically in NEDC-32523P-A, "Licensing Topical Report Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2) (Reference 4). Consistent with the discussion in Section 3.6.1 of ELTR2, TVA has taken actions to identify, monitor, and mitigate inter-granular stress corrosion cracking (IGSCC) and flow-accelerated corrosion (FAC) in the BFN Units 1, 2, and 3 RCPB.

For IGSCC to occur, three conditions must exist. IGSCC requires the existence of a susceptible material, the presence of residual stress in the weld, and the

presence of an aggressive environment; IGSCC will not occur if any of these conditions are not present. Operation at EPU conditions will result in somewhat higher temperature and flow for some systems comprising portions of the RCPB, but these changes do not influence the causative factors required for IGSCC to occur. Operation at higher power level will result in a slightly higher oxygen generation rate due to radiolysis of water; however, coolant chemistry will continue to be strictly controlled and maintained within specified limits. Therefore, as concluded in Section 3.6.1 of ELTR2, operation at EPU is expected to have a negligible impact on the occurrence of IGSCC. Irrespective of this conclusion, and as discussed in further detail in the responses to Requests 3, 4, and 5 below, TVA has taken comprehensive measures to mitigate and IGSCC. These measures included replacement of piping with IGSCC-resistant material, application of weld stress improvement measures, and implementation of Hydrogen Water Chemistry (HWC). These measures address the IGSCC causative factors and will protect the RCPB against IGSCC.

As discussed in Enclosure 4, Section 3.11 of the license amendment application TVA has evaluated the impact of operation at EPU conditions on FAC-susceptible piping within the RCPB. Consistent with GE's evaluation documented in Section 3.6.1 of ELTR2, TVA's evaluation concluded that the increases in temperature and flow will not contribute significantly to increased wear due to FAC.

The results of TVA's evaluation for BFN were consistent with GE's evaluation described in ELTR2. In its September 14, 1998, Safety Evaluation (Reference 5), the NRC concurred with GE's evaluation, provided that licensees reexamine their erosion/corrosion inspection programs. As part of implementation of EPU, the BFN FAC program will be updated to incorporate changes in operating conditions due to EPU, and susceptible piping will continue to be monitored as required by that program.

Based on the evaluations performed, TVA has concluded that operation at EPU conditions will have a negligible impact on RCPB materials.

NRC Request 3

Identify the materials of construction for the Reactor Recirculation System piping and discuss the effect of the requested EPU on the material. If other than type "A" (per NUREG-0313) material exists, discuss augmented inspection programs and the adequacy of augmented inspection programs, in light of the EPU.

TVA Reply 3

The Unit 2 Reactor Recirculation System (RRS) consists of suction and discharge piping fabricated with Type 304 stainless steel in accordance with ASTM A358. The bottom portion of the ten system risers is also ASTM A358 Type 304 stainless steel. The top portion of each riser including the riser elbow and recirculation inlet safe end is ASME SA376 Type 316 NG. The recirculation inlet safe ends are an improved crevice-free design. To mitigate weld residual stresses in the Unit 2 RRS,

Mechanical Stress Improvement Process (MSIP) or Induction Heat Stress Improvement (IHSI) was applied to the accessible welds.

The Unit 3 RRS consists of suction piping fabricated with Type 304 stainless steel in accordance with ASTM Type A358. All piping downstream of the 28 inch recirculation discharge piping, including the cross, ring header, the risers, and the recirculation inlet safe ends, are ASME SA376 Type 316 NG stainless steel. The recirculation inlet safe ends are an improved crevice-free design. To mitigate weld residual stresses in the Unit 3 RRS, MSIP or IHSI was applied to the accessible welds. The Unit 3 Type 316 NG stainless piping utilizes an improved design which eliminated several piping welds.

The use of IGSCC resistant replacement materials, application of stress improvement, and improved designs to reduce welds and crevices mitigate the possibility of future IGSCC. To provide further mitigation, TVA implemented HWC on Units 2 and 3.

The RRS welds have been categorized and inspected in accordance with NUREG-0313, Rev. 2, as modified by TVA's Risk-Informed Inservice Inspection (ISI) Program based on the Boiling Water Reactor Vessel and Internals Project (BWRVIP) report BWRVIP-75, "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules."

The nuclear industry has established that the IGSCC in stainless steel piping welds is the result of weld residual stress, an oxidizing environment, and a susceptible material. As described above, TVA has employed the use of IGSCC resistant replacement material, applied weld stress improvement, and reduced the oxidizing environment with HWC. Implementation of EPU on the Browns Ferry units will not affect the causative factors for IGSCC and, as such, the current established inspection and mitigation programs are adequate to support implementation of EPU.

NRC Request 4

Section XI of the American Society of Mechanical Engineers (ASME) Code allows flaws to be left in service after a proper evaluation of the flaws is performed in accordance with the ASME, Section XI rules. Indicate whether such flaws exist in the Reactor Recirculation System piping and evaluate the effect of the EPU on the flaws.

TVA Reply 4

IGSCC-induced flaws have been identified in the BFN Units 2 and 3 RRS. These flaws have been evaluated in accordance with ASME Section XI requirements. NRC-accepted evaluation, repair or mitigation measures as defined in NUREG-0313, Rev. 2 and Generic Letter (GL) 88-01 "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," have been applied to these flaws.

It has been established that for IGSCC to initiate and propagate, weld residual stress, an oxidizing environment, and a susceptible material are required. Since implementation of EPU for BFN does not adversely impact any of these factors, the potential of IGSCC initiation or growth will not increase due to EPU. TVA has employed weld overlay repairs, resistant replacement material, weld stress improvement, and HWC to reduce the oxidizing environment on the BFN Units 2 and 3 RRS piping. The current IGSCC inspection and mitigation programs are adequate to support implementation of EPU. The piping components containing these flaws have been categorized as described in NRC GL 88-01 and will continue to be inspected in accordance with the BWRVIP-75 and risk-informed ISI program.

NRC Request 5

Discuss flaw mitigation steps that have been taken for the RCPB piping and discuss changes, if any, that will be made to the mitigation process as a result of the EPU.

TVA Reply 5

The nuclear industry has established that initiation and propagation of IGSCC in stainless steel piping welds is the result of weld residual stress, an oxidizing environment, and a susceptible material. To mitigate the initiation or growth of IGSCC, TVA has employed IGSCC-resistant replacement material, weld stress improvement, and HWC to reduce the oxidizing environment in BFN Units 2 and 3. Operation at higher power level will result in a slightly higher oxygen generation rate due to radiolysis of water; however, coolant chemistry will continue to be strictly controlled and maintained within specified limits. Since EPU operation does not adversely affect any of the factors required to initiate or propagate IGSCC, no changes to IGSCC mitigation measures are needed or planned for EPU operation.

To mitigate the potential for IGSCC initiation or propagation in BFN Units 2 and 3, the following piping was previously replaced:

- The Units 2 and 3 RRS inlet safe ends were replaced with an improved design employing resistant material and a crevice-free configuration. The majority of the Unit 2 RRS riser piping was replaced with corrosion-resistant material in conjunction with replacement of the Unit 2 inlet safe ends; thus, reducing the number of welds on this piping.

- The Unit 3 RRS piping was replaced downstream of the 28 inch pump discharge piping including the risers and ring-header. The replaced piping was of an improved design employing resistant materials and fewer welds.

- A portion of the Unit 2 Reactor Water Cleanup (RWCU) System piping inside and outside the drywell was replaced with corrosion-resistant material. Unit 3

RWCU System piping inside and outside containment was replaced with corrosion-resistant material.

- The Unit 2 and 3 Core Spray System piping inside the drywell was replaced to the inboard containment isolation valve.
- The Residual Heat Removal RHR System piping on Unit 3 was replaced between the manual isolation and the tie-in to the RRS connection for the 24 inch piping except for the two injection valves.
- The Unit 2 jet pump instrumentation nozzles safe ends and seal assemblies were replaced with a new design using corrosion-resistant material.

Where practical in the above replacements, improved designs were employed to reduce the number of welds in IGSCC-susceptible piping. In several cases, full structural weld overlays have been used in both Units 2 and 3 as a mitigative measure and, where possible, Inductive Heat Stress Improvement (ISHI) or Mechanical Stress Improvement Process (MSIP) has been used.

The NRC has previously reviewed TVA's programs to mitigate IGSCC in Units 2 and 3 and concluded in NRC safety evaluations dated December 21, 1989 (Reference 6) and December 3, 1993 (Reference 7), respectively, that these programs were in accordance with the guidance in NRC Generic Letter 88-01.

Implementation of EPU on the Browns Ferry units will not adversely affect the causal factors needed for IGSCC to initiate and propagate; therefore, the current established inspection and mitigation programs are adequate to support implementation of EPU. The welds in both units will continue to be inspected in accordance with NUREG-0313, Rev. 2 requirements as modified by TVA's Risk-Informed Inservice Inspection Program.

NRC Request 6

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that setpoint Allowable Values (AVs) established by means of Instrumentation, Systems, and Automation Society, document ISA 67.04, Part 2, Method 3 (Method 3), do not provide adequate assurance that a plant will operate in accordance with the assumptions upon which the plant safety analyses have been based. These concerns are summarized in the June 17, 2004, letter from Mr. Ledyard B. Marsh to Mr. Alex Marion, Nuclear Energy Institute, available on the public website under ADAMS Accession Number ML041690604. In this submittal, several setpoint AVs have been established using Method 3. Tennessee Valley Authority should describe the approach intended to ensure at least 95 percent probability with at least 95 percent confidence that the associated Technical Specification (TS) action will be initiated with the process variable no less conservative than the initiation value assumed in the plant safety analyses. The approach presented should be detailed and should explicitly address how the approach provides adequate assurance that the safety analysis assumptions will not be violated.

TVA Reply 6

TVA is working with the Nuclear Energy Institute (NEI) Setpoint Methods Task Force to reach resolution of the issues regarding the use of the Instrumentation, Systems, and Automation Society (ISA) Recommended Practice, ISA RP67.04, Part II, Method 3. Once satisfactory resolution has been reached between TVA, NEI, and the NRC, TVA will expeditiously prepare and submit an amendment request, if necessary, to implement the generic resolution of this issue.

NRC Request 7

Provide a detailed discussion to address the impact of the EPU on the fire protection program or post fire safe-shutdown analysis evaluation. GE report "GE ELTR NEDC- 33047P, Rev. 2," in Enclosure 4 of the submittal, appears to be the only discussion of fire protection program, fire suppression and detection systems in the submittal.

TVA Reply 7

The Browns Ferry Nuclear Plant Fire Protection Report (BFN FPR), in accordance with requirements in 10 CFR 50.48, discusses the Browns Ferry Fire Protection Program which includes the following components:

- Fire protection features, including suppression and detection systems, fire barriers and fire dampers, emergency lighting, etc.,
- Fire Hazard Analysis,
- Appendix R Safe Shutdown Analysis,
- Fire emergency procedures including safe shutdown instructions and pre-fire plans,
- Fire protection organization,
- Training,
- Periodic inspection and testing of fire protection systems.

Browns Ferry Units 2 and 3 have NFPA compliant fire suppression and detection systems and Appendix R required fire barrier assemblies including doors, penetrations and dampers. The simultaneous operation of Browns Ferry Units 1, 2, and 3 at EPU conditions will not affect the design or operation of the units' fire detection systems, fire suppression systems or Appendix R fire barrier assemblies installed to satisfy NRC fire protection requirements. The plant is compartmentalized and protected in accordance with Appendix R requirements such that a fire in one area will not affect the equipment in another area or, alternate shutdown paths capable of controlling each of the units are available. The increase in power associated with EPU will not affect this compartmentalization approach. Changes in physical plant configuration and

combustible materials as a result of planned modifications to implement EPU will be evaluated as part of the fire hazards analysis in accordance with the requirements of the BFN FPR.

The BFN FPR currently demonstrates Units 2 and 3 compliance with 10 CFR 50.48 and 10 CFR 50 Appendix R requirements to achieve and maintain safe shutdown following a fire by achieving the following: (1) one train of systems necessary to achieve and maintain hot shut down be maintained free of fire damage, and (2) that the (a) systems necessary to achieve and maintain cold shutdown can be repaired within 72 hours if redundant systems are being used, or (b) the system can be repaired, and cold shut down can be achieved, within 72 hours if alternative or dedicated shutdown capability is being used. As part of the Units 2 and 3 EPU implementation, the BFN FPR will be revised to demonstrate continued compliance with 10 CFR 50.48 and 10 CFR 50 Appendix R requirements.

Thermal-hydraulic analyses of important plant process parameters following a fire assuming EPU conditions were performed for both GE and Framatome ANP fuel and the results in Enclosures 4 and 5 of the license amendment application (Reference 1) indicate the limits for the reactor process variables are not exceeded following a fire event.

The limiting Appendix R fire event from the current Browns Ferry Units 2 and 3 analyses was reanalyzed at EPU conditions. For GE fuel, the fuel heatup analysis was performed using the SAFER/GESTR-LOCA analysis model. The containment analysis was performed using the SHEX model. Justification for using SAFER/GESTR-LOCA and SHEX models for EPU calculations is presented in Section 4 of Enclosure 4 of the EPU licensing application. These are the same analysis methodologies that were used for the current Unit 2 and 3 Appendix R fire event analysis. For Framatome ANP fuel, the Appendix R analysis was performed using the Framatome ANP loss of coolant accident-emergency core cooling system (LOCA-ECCS) Appendix K analysis codes for Framatome fuel. These evaluations determined the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event, for both GE and Framatome ANP fuel types.

Table 3 provides the key inputs for the analyses for both GE and Framatome ANP fuel types.

Table 3

Browns Ferry Appendix R Fire Event Evaluation Key Inputs			
Item	Parameter	Units	EPU Value
25	Fuel Type	NA	GE13 GE14 ATRIUM-10
26	Initial Core Thermal Power	MWt	3952
27	Initial Core Flow	Mlbm/hr	102.5
28	Initial Dome Pressure	psia	1055(1)
29	Initial Indicated Water Level, Above Vessel Zero (AVZ)	inch	550
30	Loss of Off Site Power (LOOP), Reactor Scram	sec	0
31	Main Steam Isolation Valve (MSIV) Closure Initiation	sec	0
32	MSIV Closure Time	sec	4
33	Feedwater Flow Ramps to Zero after Scram	sec	5
34	Decay Heat Model	NA	1979 ANS 5.1
35	Low pressure core injection (LPCI) Flow Rate at 20 psig	gpm	9,400
36	Maximum Vessel Pressure at Which Pump Can Inject Flow	psig	319.5
37	LPCI Injection Valve Pressure Permissive	psig	385
38	Main steam relief valve (MSRV) Setpoint	psig	1140/ 1150/ 1160(2)
39	MSRV Capacity/Valve at Reference Pressure of 1125 psig	Mlbm/hr	0.8
40	Initial Suppression Pool Temperature	°F	95
41	Initial Containment Pressure	psia	15.9
42	Initial Suppression Pool Water Volume	ft^3	121,500
43	One RHR Pump Flow in Alternate Shutdown Cooling mode	gpm	6,000
44	One RHR Heat Exchanger K-Factor	Btu/sec-°F	223
45	RHRSW Temperature	°F	95
46	HPCI Rated Flow	gpm	5,000
47	HPCI Response Time	sec	21
48	HPCI Water Temperature	°F	100

(1) The 1055 psia dome pressure is conservatively used to bound the operating dome pressure of 1050 psia at EPU conditions.
(2) Bounding for the operating MSRV nominal setpoints of 1135/1145/1155 psig

Appendix R Analysis for GE Fuel

The postulated Appendix R fire event using the minimum Safe Shutdown System (SSDS) was analyzed for GE fuel for the three cases described below:

Case 1: No spurious operation of plant equipment occurs and the operator initiates three main steam relief valves (MSRVs) 25 minutes into the event.

Case 2: One MSRV opens immediately due to a spurious opening signal generated as a result of the fire. The MSRV is reclosed 10 minutes into the event by operator action. The operator initiates three MSRVs 20 minutes into the event.

Case 3: One MSRV opens immediately as in Case 2, but remains open throughout the event. The operator initiates three MSRVs 20 minutes into the event.

The above are the same cases as those described in the BFN FPR. These cases were evaluated for EPU with some reduction in conservatism in the analytical assessment as compared to the methods used currently for Units 2 and 3 for pre-EPU conditions.

For pre-EPU analyses for Browns Ferry Units 2 and 3, for all cases it was conservatively assumed that the LPCI injection does not occur until reactor pressure is ≤ 200 psig, instead of the standard injection point of 319.5 psig, (pump shutoff head). Assuming the injection does not occur until ≤ 200 psig, delays LPCI injection into the vessel. For the EPU assessment, the LPCI injection valve is assumed to be opened by operator action when the reactor vessel pressure reaches 385 psig. LPCI flow to the vessel begins at 319.5 psig which is the maximum pressure at which the LPCI pumps can inject into the vessel. This adjustment to the analysis does not affect any operator action or plant configuration changes because the current procedures direct the operations staff to open the LPCI injection valve when RPV pressure is ≤ 450 psig and the pump characteristics have not been changed by the EPU so that injection will occur at 319.5 psig. The recirculation line discharge valve is assumed to always remain open, which reduces the LPCI flow to the core.

The results of the analyses are contained in Table 6-5 of Enclosure 4 of the EPU application and replicated in Table 4 below for convenience.

Table 4

Browns Ferry Appendix R Fire Event Evaluation Results Units 2 & 3 GE Fuel			
Parameter	CLTP	EPU	App. R Criteria
Cladding Heatup (PCT), °F	1485	1428	≤ 1500
Primary System Pressure, psig	1150	1150	≤ 1375
Primary Containment Pressure, psig	18.6	13.6	≤ 56
Suppression Pool Bulk Temperature, °F	212	227	≤ 281(2) ≤ 227(3)
Net Positive Suction Head (1)	Yes	Yes	Adequate for system using suppression pool water source

1. Net positive suction head (NPSH) demonstrated adequate, see Section 4.2.5 of Enclosure 4 of the license amendment application.
2. Containment structure design limit.
3. Torus attached piping limit.

Based on the above analysis results for GE fuel, each of the analyzed parameters is less than the associated 10 CFR 50 Appendix R acceptance criteria and, thus, the integrity of the fuel, reactor vessel and primary containment structure will be maintained.

The bounding case for PCT is Case 1. For this case, the time available to the operator to open three MSRVs is reduced from 30 minutes to 25 minutes at the EPU conditions. This reduction in the time available does not have any effect because the current procedures require this action to be completed within 20 minutes. Although the analysis assumes the time available to perform this operator action is reduced by five minutes for Case 1, five minutes of margin remains compared to the present analysis. For CLTP and EPU, the PCTs are calculated using conservative LPCI performance characteristics (e.g., minimum flow rate as functions of vessel pressure).

Spurious operation of the High Pressure Core Injection (HPCI) System was also evaluated. The purpose of the evaluation was to determine the time the water level would reach the elevation of the main steam lines if operation of the HPCI system is assumed to start at the beginning of an Appendix R event. The analysis results show that the time the reactor water level would reach the main steam line elevation remains greater than six minutes.

Appendix R Analysis for Framatome ANP Fuel

As discussed in Enclosure 5 of the license amendment application (Reference 1), a plant specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions and Framatome ANP fuel. As discussed above, the Appendix R

analysis was performed using the Framatome ANP LOCA-ECCS Appendix K analysis codes for Framatome ANP fuel.

The Framatome ANP analyses inputs are as described above for the GE fuel.

The postulated Appendix R fire event using the minimum SSDS was analyzed for the three cases described below:

Case 1: No spurious operation of plant equipment occurs and the operator initiates three MSRVs 25 minutes into the event.

Case 2: One MSRV opens immediately due to a spurious signal generated as a result of the fire. The MSRV is closed 10 minutes into the event by operator action. The operator initiates three MSRVs 20 minutes into the event.

Case 3: One MSRV opens immediately and remains open. The operator initiates three MSRVs 20 minutes into the event.

The analyses assume one LPCI pump is available to mitigate the Appendix R event. LPCI flow is modeled to begin at 319.5 psig as discussed above for GE fuel. The recirculation line isolation valve is assumed to always remain open, which reduces the LPCI flow to the core.

The results of the analyses are contained in Table 5 below:

Table 5

Browns Ferry Appendix R Fire Event Evaluation Results Units 2 & 3 Framatome ANP Fuel		
Parameter	EPU	App. R Criteria
Cladding Heatup (PCT), °F	1238	≤ 1500
Primary System Pressure, psig	1224	≤ 1375
Primary Containment Pressure, psig	13.6[4]	≤ 56
Suppression Pool Bulk Temperature, °F	227[4]	≤ 281[2] ≤ 227[3]
Net Positive Suction Head [1]	Yes	Adequate for system using suppression pool water source

1. NPSH demonstrated adequate, see Section 4.2.5 of Enclosure 5 of the license amendment application.
2. Containment structure design limit.
3. Torus attached piping limit.
4. Results are independent of fuel type, these results based upon GE Containment Analysis.

Based on the above analysis results for Framatome ANP fuel, each of the analyzed parameters is less than the associated 10 CFR 50 Appendix R acceptance criteria and thus, the integrity of the fuel, reactor vessel, and primary containment structure will be maintained.

The bounding PCT case is Case 1. For this case, the time available to the operator to open three MSRVs is reduced from 30 minutes to 25 minutes at the EPU conditions. This reduction in the time available does not have any effect because the current procedures require this action to be completed within 20 minutes. Although the analysis assumes the time available to perform this operator action is reduced by five minutes for Case 1, five minutes of margin remains compared to the present analysis. For CLTP and EPU, the PCTs are calculated using conservative LPCI performance characteristics (e.g., minimum flow rate as functions of vessel pressure).

Spurious operation of the HPCI System was also evaluated. The purpose of the evaluation was to determine the time the water level would reach the elevation of the main steam lines if operation of the HPCI system is assumed to start at the beginning of an Appendix R event. In Section 6.7.1 of Enclosure 5 to the license amendment application, TVA reported that the results of the Framatome ANP analysis indicated that the time the reactor water level would reach the main steam line elevation was greater than five minutes. Consequently, TVA indicated that a change to the response procedure was necessary to ensure that HPCI system isolation occurs within five minutes.

Subsequent to the initial submittal, Framatome ANP revised the analysis of the case involving spurious operation of the HPCI System to more accurately model the effect of HPCI injection on reactor vessel depressurization. In the initial analysis, HPCI injection was assumed to occur above the reactor vessel downcomer region, resulting in greater steam condensation, faster depressurization, and greater level swell than expected for this event. While this accurately models HPCI impact during a LOCA, it over-predicts reactor vessel level rise for this Appendix R event. At the start of the assumed spurious HPCI initiation, the reactor vessel downcomer level is 4.3 feet above the HPCI injection location and will remain above the HPCI elevation for the entire event. The Appendix R analysis for this event was revised to more accurately model HPCI impact on vessel level rise. The revised analysis indicates that the reactor water level would reach the main steam line elevation in greater than six minutes. Therefore, the current response procedures which require securing HPCI within six minutes during an Appendix R event remain valid and no changes or additional operator training are required.

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Additional Evaluation

The flow rates of RCIC and CRD Systems are approximately 600 gpm and 200 gpm, respectively. While these flowrates are adequate to overcome the inventory loss, they are insufficient to raise water level due to more decay heat within the first 30 minutes into the event at EPU. Therefore, operation of the RCIC and CRD Systems will not cause water intrusion into the MSLs.

During an Appendix R fire event, the feedwater controller may spuriously operate resulting in an increase in the feedwater flow. This could happen only if the event occurs with offsite power available and the operators can remain in the control room. If offsite power is not available, the MSIVs would close automatically by their fail-safe design as well as reactor feedwater pump, the condensate and condensate booster pumps would be lost. If control room evacuation is required, the operators would manually isolate the MSIVs prior to leaving the control room. This would prevent the feedwater pumps from overfilling the vessel. Consequently, both offsite power and the control room must be available during the fire event in order for spurious operation of feedwater system to occur. Under these conditions, if both offsite power and control room are available during a fire event, the operator would have full knowledge of the reactor conditions and could trip the feedwater pumps from the control room when the reactor water level approaches the MSL. Therefore, spurious operation of the feedwater system will not lead to water intrusion into the MSL.

Conclusion

The results of the Appendix R evaluation for EPU demonstrate that the fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained and that sufficient time is available for the operators to perform the necessary actions. The exemption for the momentary core uncovery during depressurization as described in the BFN FPR remains necessary for EPU. EPU does not affect any other exemptions described in the BFN FPR. There are no changes necessary to the systems and equipment required for safe shutdown. At EPU conditions, one train of systems remains available to achieve and maintain safe shutdown conditions from either the main control room or the remote shutdown panel. The operator actions required to mitigate the consequences of a fire are not affected by EPU. Sufficient time is available for the operators to perform the necessary actions and any necessary changes to procedures will be accomplished concurrent with EPU implementation. Therefore, EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of an Appendix R fire event and satisfies the requirements of Appendix R with respect to achieving and maintaining safe shutdown in the event of a fire.

The introduction of EPU does not impact the currently existing Fire Protection organization or training program. The fire protection systems will continue to be inspected and tested to the same criteria as currently defined in plant procedures.

Thus, TVA has concluded that EPU does not adversely impact the Fire Protection Program or the post fire safe shutdown analysis evaluation.

NRC Request 8

Discuss how the change in the fluence by the EPU will affect the surveillance capsule withdrawal schedule (i.e., discuss whether there are any effects on the Boiling Water Reactor Vessel and Internals Project Integrated Surveillance Program, as applicable to Units 2 and 3) because of this power uprate.

TVA Reply 8

BFN Units 2 and 3 are participating in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP). The ISP is described in topical report BWRVIP-86-A, "BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan." That topical report, reviewed and approved by the NRC, identifies the participating BWR fleet vessel materials, surveillance capsules and contents and locations of those capsules, representative materials for each participating BWR reactor vessel, and the withdrawal schedule for each capsule. It is not expected that the increase in fluence resulting from implementation and operation at EPU conditions will impact the surveillance withdrawal schedule identified in BWRVIP-86-A.

TVA removed the first BFN Unit 2 reactor vessel material surveillance capsule in 1994 and submitted the associated surveillance material test report to the NRC by letter dated October 18, 1995 (Reference 8). The results of that testing confirmed that the measured shifts in the 30 ft-lb nil-ductility transition temperature (RT_{NDT}) and the measured decreases in Upper Shelf Energy (USE) were within the Regulatory Guide 1.99, Revision 2 predictions.

The second and third BFN Unit 2 reactor vessel material surveillance capsules were previously planned for removal and testing in 2001 and 2007 respectively under its plant-specific surveillance program; the schedule of which was initially adopted by the ISP. However, as discussed in Section 4.2 and Tables 4-3 and 4-4 of BWRVIP-86-A, the schedule for removal and testing of the second BFN Unit 2 capsule was deferred until 2011 and the third capsule, at the time of approval of BWRVIP-86-A, was deferred indefinitely for future use for license renewal.

These deferrals were made for two reasons. First, they were deferred to facilitate testing and evaluation of nine BWR Owners' Group (BWROG) Supplemental Surveillance Program (SSP) capsules that had been fabricated and installed in host reactors and were scheduled for withdrawal in the near term. Secondly, the deferrals were made in response to an NRC Staff request to delay testing in order to obtain better consistency between the capsule fluences and the target reactor vessel 1/4T end-of-life fluences. This resulted in deferring withdrawal of the second BFN Unit 2 until 2011 (three years before the expiration of the current BFN Unit 2 operating license). Because the lead factors for the surveillance capsules (the ratio of flux at the surveillance capsule to the peak flux at the inside vessel

surface) is not changed significantly, the basis used by the BWRVIP for scheduling withdrawal of the second BFN Unit 2 capsule in 2011 is not changed. Therefore, operation at EPU conditions is not expected to result in a need to change the existing withdrawal schedule. However, as stated in BWRVIP-86-A, the BWRVIP has committed to periodically evaluate the testing matrix based on information such as updated fluence analyses and submit any planned changes to the NRC for its approval.

In July, 2003, the BWRVIP published BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal." That report, still being reviewed by the NRC, provides the bases and proposed reactor vessel material surveillance capsule withdrawal schedule to support extended operation following license renewal. BWRVIP-116 established a target fluence of 40 Effective Full Power Years (EFPY). Based on that target, the BWRVIP proposes withdrawal of the third BFN Unit 2 surveillance capsule in 2026.

Under the ISP, the BFN Unit 2 reactor vessel materials are representative materials for BFN Unit 3. The BFN Unit 3 capsules are not now planned for removal, but remain in the vessel as standby capsules under the ISP, and are available for removal and analysis if needed by the ISP. Therefore, implementation of EPU has no impact on the ISP as regards withdrawal of BFN Unit 3 surveillance capsules.

NRC Request 9

Discuss the effects of the EPU on the Upper Shelf Energy of the beltline components and the welds of the Units 2 and 3 reactor pressure vessels.

TVA Reply 9

As stated in TVA's August 2, 1993, response to an NRC request for additional information concerning NRC Generic Letter 92-01, Revision 1 (Reference 9), TVA adopted the BWROG Equivalent Margin Analysis as its licensing basis for demonstrating that the BFN Upper Shelf Energy throughout the life of the plant meet the requirements of 10 CFR 50 Appendix G.

The impact of EPU operation on the Equivalent Margin Analyses for the BFN Units 2 and 3 limiting reactor vessel beltline materials is provided in Tables 6 through 9 that follow. These evaluations demonstrate adequate upper shelf energy margins for EPU operation.

Flux analysis performed by Framatome ANP using the ATRIUM-10 fuel determined that the flux analysis performed for Units 2 and 3 by GE for GE-14 fuel conservatively bound the results for ATRIUM-10 fuel. Therefore, the equivalent margins analysis applicability forms provided below are valid and conservative for Atrium-10 fuel.

Table 6

Browns Ferry Unit 2 Reactor Vessel Plate Upper Shelf Energy Equivalent Margin Analysis for 34 EFPY at EPU Conditions

Plate EMA 34 EFPY – Plant Applicability Verification Form

Surveillance Plate USE (Heat A0981-1):

%Cu	=	0.14	
1st Capsule Fluence	=	1.52 E+17 n/cm^2	
2nd Capsule Fluence	=	N/A	
1st Capsule Measured % Decrease	=	4	(Charpy Curves)
2nd Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule R.G. 1.99 Predicted % Decrease	=	9	(R.G. 1.99, Figure 2)
2nd Capsule R.G. 1.99 Predicted % Decrease	=	N/A	(R.G. 1.99, Figure 2)

Limiting Beltline Plate USE (Heat C2463-1):

%Cu	=	0.17	
34 EFPY 1/4T Fluence	=	1.0 E+18 n/cm^2	
R.G. 1.99 Predicted % Decrease	=	15.5	(R.G. 1.99, Figure 2)
Adjusted % Decrease	=	N/A	(R.G. 1.99, Position 2.2)

15.5% ≤ 21%, so vessel plates are bounded by equivalent margin analysis

Table 7

Browns Ferry Unit 2 Reactor Vessel Weld Upper Shelf Energy Equivalent Margin Analysis for 34 EFPY at EPU Conditions

Weld EMA 34 EFPY--Plant Applicability Verification Form

Surveillance Weld USE (Heat D55733):

%Cu	=	0.20	
1st Capsule Fluence	=	1.52 E+17 n/cm^2	
2nd Capsule Fluence	=	N/A	
1st Capsule Measured % Decrease	=	-3	(Charpy Curves)
2nd Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule R.G. 1.99 Predicted % Decrease	=	13	(R.G. 1.99, Figure 2)
2nd Capsule R.G. 1.99 Predicted % Decrease	=	N/A	(R.G. 1.99, Figure 2)

Limiting Beltline Weld USE (Heat ESW):

%Cu	=	0.24	
34 EFPY 1/4T Fluence	=	1.0 E+18 n/cm^2	
R.G. 1.99 Predicted % Decrease	=	22.5	(R.G. 1.99, Figure 2)
Adjusted % Decrease	=	N/A	(R.G. 1.99, Position 2.2)

22.5% ≤ 34%, so vessel welds are bounded by equivalent margin analysis

Table 8

Browns Ferry Unit 3 Reactor Vessel Plate Upper Shelf Energy Equivalent Margin Analysis for 34 EFPY at EPU Conditions

Plate EMA 34 EFPY – Plant Applicability Verification Form

Surveillance Plate USE (Heat C3188-2):

%Cu	=	0.10	
1st Capsule Fluence	=	N/A	
2nd Capsule Fluence	=	N/A	
1st Capsule Measured % Decrease	=	N/A	(Charpy Curves)
2nd Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule R.G. 1.99 Predicted % Decrease	=	N/A	(R.G. 1.99, Figure 2)
2nd Capsule R.G. 1.99 Predicted % Decrease	=	N/A	(R.G. 1.99, Figure 2)

Limiting Beltline Plate USE (Heat C3222-2):

%Cu	=	0.15	
34 EFPY 1/4T Fluence	=	1.0 E+18 n/cm^2	
R.G. 1.99 Predicted % Decrease	=	14	(R.G. 1.99, Figure 2)
Adjusted % Decrease	=	N/A	(R.G. 1.99, Position 2.2)

14% ≤ 21%, so vessel plates are bounded by equivalent margin analysis

Table 9

Browns Ferry Unit 3 Reactor Vessel Weld Upper Shelf Energy Equivalent Margin Analysis for 34 EFPY at EPU Conditions

Weld EMA 34 EFPY--Plant Applicability Verification Form

Surveillance Weld USE (Heat ESW):

%Cu	=	0.11	
1st Capsule Fluence	=	N/A	
2nd Capsule Fluence	=	N/A	
1st Capsule Measured % Decrease	=	N/A	(Charpy Curves)
2nd Capsule Measured % Decrease	=	N/A	(Charpy Curves)
1st Capsule R.G. 1.99 Predicted % Decrease	=	N/A	(R.G. 1.99, Figure 2)
2nd Capsule R.G. 1.99 Predicted % Decrease	=	N/A	(R.G. 1.99, Figure 2)

Limiting Beltline Weld USE (Heat ESW):

%Cu	=	0.24	
34 EFPY 1/4T Fluence	=	1.0 E+18 n/cm^2	
R.G. 1.99 Predicted % Decrease	=	22.5	(R.G. 1.99, Figure 2)
Adjusted % Decrease	=	N/A	(R.G. 1.99, Position 2.2)

22.5% ≤ 34%, so vessel welds are bounded by equivalent margin analysis

NRC Request 10

Provide a discussion on any potential emergency action level changes that have been identified as a result of the proposed power uprate.

TVA Reply 10

The only currently determined EPU effect on emergency action levels at Browns Ferry is the change in threshold values of primary containment radiation used for the determination of event classification. Browns Ferry radiological analyses have been revised to account for the effects of EPU. The analyses consider the specific locations of Browns Ferry drywell radiation monitors and accident isotopic releases to the containment atmosphere in accordance with applicable regulatory requirements. Effects on the resulting drywell radiation monitor values will be placed in emergency procedure revisions for emergency action level changes concurrent with implementation of EPU. Emergency event response actions are not affected. Changes in core design are routinely evaluated as part of the reload process for impact on emergency action entry conditions and procedures revised as necessary.

NRC Request 11

Provide a list specifically identifying all design bases changes, excluding TS changes, in the submittal requiring prior NRC approval.

TVA Reply 11

The Browns Ferry EPU license amendment request is based upon the NRC approved generic format and content for EPU licensing reports as described in ELTR1. As established by ELTR1, analyses and evaluations have been performed to justify increasing the licensed thermal power. Inherent in this process is integration of plant design bases changes for the systems, structures, and components that are affected. These changes are provided in the BFN EPU license amendment request including the enclosures thereto.

When licensees determine that changes to the plant involve a Technical Specification change, associated design bases changes are not individually reviewed to determine if prior NRC approval is required. The changes to the plant are packaged as a whole (TS changes and design basis changes) and submitted for NRC approval in accordance with regulations. As with the Browns Ferry EPU license amendment request, design bases changes are not individually reviewed (consistent with 10 CFR 50.59) to determine if prior NRC approval is required.

As provided by the NRC in RS-001, the review standard has established standardized review guidance and acceptance criteria for the staff's reviews of EPU applications in order to enhance the consistency, quality, and completeness of reviews. RS-001 serves as a tool for the staff's use when processing EPU applications in that it provides detailed references to various regulatory documents containing information related to the specific areas of review. Reviews for prior

NRC approval of individual changes associated with the EPU license amendment request are not proposed by either ELTR1 or RS-001.

Neither the GE BFN Units 2 and 3 PUSAR provided in Enclosure 4 of the license amendment application, nor the Framatome ANP Browns Ferry Units 2 and 3 FUSAR provided in Enclosure 5 of the license amendment application were annotated to identify the individual design basis changes that require prior NRC approval. However, to assist in the regulatory review of the Browns Ferry EPU license amendment request, TVA reviewed the application to identify design/licensing bases changes, that if made independent of the EPU 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. These changes are identified in Table 10 below. The below listed changes are considered to be specific to Browns Ferry requirements and may not have been part of the NRC review of prior licensee EPU requests. These changes have not been reviewed (consistent with 10 CFR 50.59) to determine if prior NRC approval is required.

Table 10

EPU Design/Licensing Bases Changes	
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 hardened wetwell vent
PUSAR Section 4.2.5	Change in ECCS NPSH margin/containment overpressure credit
PUSAR Section 4.3	Change in limiting PCT event
PUSAR/FUSAR Section 4.7	Change in nitrogen consumption rate
PUSAR/FUSAR Section 6.7.1	Appendix R analyses - Reduction in time to open 3 MSRVs
PUSAR Section 7.2	Reduction in retention time of condensate in the condenser hotwell

NRC Request 12

In Enclosure 4, Section 7.4 of the submittal, a flow margin of 5 percent is established for the feedwater/condensate system. Discuss the basis for this

criterion and how it compares with the pre-EPU margin. Discuss whether this is a change to the licensing basis, and how the flow margin and feedwater pump runout assumptions will be confirmed during startup testing.

TVA Reply 12

The basis for the 5% feedwater/condensate flow margin is the Browns Ferry transient analyses, which indicates the system need only have the transient capacity necessary to provide at least 105% of the EPU power feedwater/condensate flow at the current reactor dome pressure of 1050 psia. This flow assures that the plant remains available during water level affected transients that may require more than rated feedwater/condensate flow to avoid a low reactor water level scram (e.g., large recirculation flow changes, pressure regulator failures, etc.) and avoid unnecessary challenges to plant safety systems. The EPU feedwater/condensate flow margin remains above 5%, and is consistent with the current pre-EPU margin; therefore, this is not a change to the licensing basis.

Hydraulic calculations based on EPU conditions determined the flow margin and runout values following the modifications planned to the Condensate, Condensate Booster, and Feedwater Pumps. System testing to verify the overall runout condition is not practical; however, planned post modification testing will confirm pump performance on an individual pump basis by a comparison of the designed flow versus actual flow. Feedwater/condensate flow margin and feedwater pump runout will be confirmed on a system basis by comparison of data from startup testing to the calculated values.

NRC Request 13

Provide a description of the major differences in the operation; procedures; system configuration; and flow, pressure, and level setpoints between Units 2 and 3.

TVA Reply 13

Browns Ferry Units 2 and 3 currently operate in functional congruency. However, as is normal for a multi-unit nuclear power plant, operation and maintenance activities require periodic individual unit modifications. Browns Ferry design and operating procedures require that the appropriate technical evaluations be performed for modifications to ensure that design requirements and system and unit functionality are retained along with unitized specific identification and information on replacement or repaired components. Typically, unit differences normally only stand for one operating cycle

EPU results in a higher main steam flow rate achieved by increasing the reactor power along slightly modified rod and core flow control lines. EPU implementation requires revising a limited number of operating parameters, adjusting some setpoints, and recalibration of instruments.

TVA plans to implement EPU on an operating unit basis in consecutive operating cycles. Plant procedures will be revised and tests will be performed for the unitized implementation of EPU. Therefore, individual units will operate with some differences in system configurations, procedures, and setpoints until completion of EPU implementation on all units. Operating procedures will reflect the differences between the EPU and Non-EPU units. Implementation of EPU on all of the Browns Ferry units will remove operational differences and, thus, return functional congruency to all of the Browns Ferry units.

References:

1. TVA letter, T. E. Abney to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 2 and 3 - Proposed Technical Specifications (TS) Change TS – 418 - Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 25, 2004.

2. NRC letter to TVA "Browns Ferry Nuclear Plant, Units 2 and 3 Request for Additional Information Regarding Extended Power Uprate, (TAC Nos. MC3743 and MC3744) (TS-418)," dated December 29, 2004.

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

4. NEDC-32523P-A, "Licensing Topical Report Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," dated February 1999.

5. Letter from T. H. Essig (NRC) to J. F. Quirk (GE), "Staff Safety Evaluation of General Electric Boiling Water Reactor (BWR) Extended Power Uprate Generic Analyses," dated September 14, 1998.

6. Letter from S. C. Black (NRC) to O. D. Kingsley (TVA) Evaluation of Tennessee Valley Authority's Supplemental Responses to Generic Letter 88-01, "NRC Position on IGSCC in BWR Systematic Shutdown Stainless Steel Piping" (TAC 61049)," dated December 21, 1989.

7. Letter from J. F. Williams (NRC) to M. O. Medford (TVA), "Browns Ferry Nuclear Plant Unit 3 - Safety Evaluation of Supplemental Response to Generic Letter 88-01 (TAC NO. M85296)," dated December 3, 1993.

8. TVA letter, P. Salas to NRC, "Browns Ferry Nuclear Plant (BFN) Unit 2 – Submittal of Eight Effective Full Power Years (EFPY) Reactor Vessel Material Surveillance Specimen Test Results and Determination of Applicability of NEDO-32205-A – Revision 1, Topical Report on Upper Shelf Energy Equivalent Margin Analysis," dated October 18, 1995.

9. TVA letter, P. Salas to NRC, "Browns Ferry Nuclear Plant (BFN) - Response to Request for Additional Information, Generic Letter 92-01, Revision 1," dated August 2, 1993.

10. NRC letter to TVA, "Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Re: Implementation of the Boiling-Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program to Address the Requirements of Appendix H to 10 CFR Part 50 (TAC Nos. MB6677 and MB6678)," dated January 28, 2003.