June 6, 2005

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 – 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)

This letter contains the additional information requested by the NRC Staff in its December 30, 2004 letter (Reference 1). This reply is in support of TVA's license amendment request TS-431 submitted on June 28, 2004 (Reference 2). TS-431 requested a license amendment and associated TS changes to support an increase in the reactor thermal power level to 3952 MWt, an approximate 20 percent increase from the original licensed thermal power level.

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

TVA is providing similar information regarding the Units 2 and 3 EPU applications 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,

Original signed by:

William D. Crouch
Acting Manager of Licensing and
Industry Affairs

References:

1. NRC letter to TVA "Browns Ferry Nuclear Plant, Unit 1- Request for Additional Information for Extended Power Uprate, (TAC No. MC3812) (TS-431)," dated December 30, 2004.

2. TVA letter, T. E. Abney 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.

Enclosure:

Reply to Request for Additional Information for BFN Unit 1 Extended Power Uprate Application.

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT
UNIT 1
REPLY TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE APPLICATION

See Attached:

- **Reply to Request for Additional Information for BFN Unit 1 Extended Power Uprate Application**

Reply to Request for Additional Information for BFN Unit 1
Extended Power Uprate Application

By letter dated June 28, 2004 (Reference 1), TVA submitted for NRC review, an application pursuant to 10 CFR 50.90 requesting an amendment to the Unit 1 operating license that increases 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 30, 2004 (Reference 2) the NRC transmitted the questions to TVA. The following provides TVA's response to the transmitted questions.

NRC Request 1

Explain why the reactor coolant pressure boundary (RCPB) piping materials are not affected by the power uprate.

TVA Reply 1

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 3). 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 will implement actions to monitor and mitigate flow-accelerated corrosion (FAC) in the BFN Unit 1 RCPB prior to startup.

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 pressure, 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 a 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 2, 3, and 4 below, TVA has taken, or is taking, comprehensive measures to mitigate IGSCC. These measures include replacement of piping with IGSCC-resistant material, application of weld stress improvement measures, and the planned 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 pressure, 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 4), 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 2

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

TVA Reply 2

The entire Unit 1 Reactor Recirculation System (RRS) piping has been replaced with corrosion-resistant material. This includes the pump suction and discharge piping, the ring header, the riser piping, and the inlet and outlet safe ends. The replacement piping and safe end material is Type 316 NG stainless material, which is resistant to IGSCC. Additionally, the use of EPRI welding techniques (such as machine welding where practical and reduced energy input) and the application of a Mechanical Stress Improvement Process (MSIP) will be utilized to reduce the potential for IGSCC. The replacement piping utilized an improved design which eliminated several piping welds. The safe ends were replaced with an improved crevice-free design. As a result of these efforts, all the Unit 1 RRS welds are Category "A" welds in accordance with NUREG-0313, Rev. 2 classifications.

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 plans to implement HWC in Unit 1.

The BFN Unit 1 RRS welds have been categorized and will be inspected prior to restart in accordance with NUREG-0313, Rev. 2. Consistent with BFN Units 2 and 3, TVA is developing a Risk-Informed Inservice Inspection (ISI) Program for BFN Unit 1 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." Following restart of BFN Unit 1, weld categorization and inspection will be in accordance with the planned Unit 1 Risk-Informed ISI Program.

The nuclear industry has established that initiation and growth of IGSCC in stainless steel piping welds results from the combination 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 will reduce the oxidizing environment by implementing HWC in BFN Unit 1 prior to restart. Operation at a 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. Implementation of EPU will not adversely 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 3

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 3

TVA has completely replaced the RRS piping on BFN Unit 1 and there are no known flaws.

NRC Request 4

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 4

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 plans to implement HWC on BFN Unit 1 to reduce the oxidizing environment. Operation at a 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 Unit 1, the following piping has been replaced:

- The RRS inlet and outlet safe ends were replaced. The replacement safe ends utilized an improved design employing Type 316 NG stainless steel, a corrosion-resistant material, and a crevice-free configuration.

- The RRS piping was replaced. This includes the 28 inch pump suction and discharge piping, the 12 inch risers and 22 inch ring-header. The replacement piping is Type 316 NG stainless which is less susceptible to IGSCC. Improved construction methods and bent pipe result in fewer welds. The ring header design eliminates the ring-header crosstie valves.

- The Core Spray (CS) System and Residual Heat Removal (RHR) System piping inside the containment were replaced. The replacement piping is Type 316 NG stainless for RHR and ASME SA-333 Gr 6 high toughness grade carbon steel for the CS System, both of which are less susceptible to IGSCC.

- Reactor Water Cleanup System piping operating above 200°F is being replaced both inside and outside containment with Type 316 NG stainless steel, which is resistant to IGSCC.

- The jet pump instrumentation nozzle safe ends and seal assemblies were replaced with an improved design, fabricated from IGSCC-resistant Type 316 NG materials.

Additionally, TVA used EPRI welding techniques (such as machine welding where practical, reduced energy input, etc.) and will implement a Mechanical Stress Improvement Process (MSIP) to the RCPB welds to further provide for flaw mitigation. The planned installation of a Hydrogen Water Chemistry system will further reduce flaw initiation on IGSCC-susceptible stainless steel materials. Further details were provided in supplemental responses to NRC Generic Letter 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping,” submitted to the NRC by letter dated July 21, 2004 (Reference 5) and supplemented by letter dated April 25, 2005 (Reference 6).

Implementation of EPU 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 BFN Unit 1 will be inspected in accordance with NUREG-0313, Rev. 2 requirements. Following restart of BFN Unit 1, weld categorization and inspection will be in accordance with the planned Unit 1 Risk-Informed ISI Program.

NRC Request 5

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that setpoint Allowable Values (AV) 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 that at least 95 percent probability with at least 95 percent confidence that the associated 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 5

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 6

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

TVA Reply 6

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 Hazards 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 Unit 1 will have NFPA compliant fire suppression and detection systems and Appendix R required fire barrier assemblies including doors, penetrations and dampers installed as part of the Unit 1 restart program. 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 Unit 1 Restart, the BFN FPR will be revised to demonstrate compliance with 10 CFR 50.48 and 10 CFR 50 Appendix R requirements with three units operating under EPU conditions.

Thermal-hydraulic analyses of important plant process parameters following a fire assuming EPU conditions were performed and the results provided in Enclosure 4 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 analysis was reanalyzed for Browns Ferry Unit 1. 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 license amendment application. These are the same analysis methodologies that were used for the Units 2 and 3 EPU Appendix R fire event analysis. This evaluation determined the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event.

Table 1 provides the key inputs for the analyses, based on GE fuel.

Table 1

Browns Ferry Appendix R Fire Event Evaluation Key Inputs			
Item	**Parameter**	**Units**	**EPU Value**
1	Fuel Type	NA	GE13 GE14
2	Initial Core Thermal Power	MWt	3952
3	Initial Core Flow	Mlbm/hr	102.5
4	Initial Dome Pressure	psia	1055(1)
5	Initial Indicated Water Level, Above Vessel Zero (AVZ)	inch	550
6	Loss of Off Site Power (LOOP), Reactor Scram	sec	0
7	Main Steam Isolation Valve (MSIV) Closure Initiation	sec	0
8	MSIV Closure Time	sec	4
9	Feedwater Flow Ramps to Zero after Scram	sec	5
10	Decay Heat Model	NA	1979 ANS 5.1
11	LPCI Flow Rate at 20 psig	gpm	9,400
12	Maximum Vessel Pressure at Which Pump Can Inject Flow	psig	319.5
13	LPCI Injection Valve Pressure Permissive	psig	385
14	MSRV Setpoint	psig	1140/ 1150/ 1160(2)
15	MSRV Capacity/Valve at Reference Pressure of 1125 psig	Mlbm/hr	0.8
16	Initial Suppression Pool Temperature	^{0}F	95
17	Initial Containment Pressure	psia	15.9
18	Initial Suppression Pool Water Volume	ft^3	121,500
19	One RHR Pump Flow in Alternate Shutdown Cooling mode	gpm	6,000
20	One RHR Heat Exchanger K-Factor	Btu/sec-°F	223
21	RHRSW Temperature	°F	95
22	HPCI Rated Flow	gpm	5,000
23	HPCI Response Time	sec	21
24	HPCI Water Temperature	^{o}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

The postulated Appendix R fire event using the minimum safe shutdown systems was analyzed 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, except as described below. 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 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 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 below in Table 2 for convenience.

Table 2

Browns Ferry Appendix R Fire Event Evaluation Results Unit 1 GE Fuel			
Parameter	**105% OLTP**(1)	**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
Suspression Pool Bulk Temperature, °F	212	227	≤ 281(3) ≤ 227(4)
NPSH (2)	Yes	Yes	Adequate for system using suppression pool water source

1. Values based on Browns Ferry Units 2 and 3 at 105% OLTP.
2. NPSH demonstrated adequate, see Section 4.2.5 of Enclosure 4 of the license amendment application.
3. Containment structure design limit.
4. Torus attached piping limit.

Based on the above analysis results, 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 25 minutes at EPU conditions. The Browns Ferry Units 2 and 3 pre-EPU analysis determined that three MSRVs were required to be opened within 30 minutes. This reduction in the time available does not have any adverse effect on the plant operators because the procedures will require this action to be completed within 20 minutes. For OLTP and EPU, the PCTs are calculated using conservative LPCI performance characteristics (e.g., minimum flow rate as functions of vessel pressure).

In addition, spurious operation of HPCI was reviewed in accordance with the BFN FPR. The HPCI System was assumed to initiate at the onset of the Appendix R event and flow at its normal flow rate. The time at which the reactor vessel water level would reach the MSLs is greater than six minutes. Therefore, the procedures will require HPCI System isolation prior to six minutes during an Appendix R event.

The flow rates of the 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.

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 BFN Fire Protection Program or the post fire safe shutdown analysis evaluation.

NRC Request 7

Discuss how the change in the fluence by 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 Unit 1, because of this power uprate).

TVA Reply 7

BFN Unit 1 is not currently part of the BWRVIP Integrated Surveillance Program (ISP). Prior to restart from its current extended outage and implementation of EPU, TVA plans to submit and obtain a license amendment to allow BFN Unit 1 participation in the ISP. By letter dated January 28, 2003 (Reference 7), the NRC approved amendments to the BFN Units 2 and 3 operating licenses, enabling participation in the BWRVIP ISP as the means for demonstrating compliance with 10 CFR 50, Appendix H.

The BWRVIP recently transmitted to the NRC BWRVIP-135, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations." That document added BFN Unit 1 to the ISP population and identified the limiting reactor vessel beltline weld materials and representative materials under the ISP. Subject to NRC approval, BWRVIP-135 identified BFN Unit 2 as the representative material for the BFN Unit 1 limiting plate material and the Boiling Water Reactors Owners' Group (BWROG) Supplemental Surveillance Program (SSP) Capsules A, B, D, E, G, and I as containing representative material for the BFN Unit 1 limiting reactor vessel beltline weld.

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 BWROG 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 capsule 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 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.

As scheduled in BWRVIP-86-A, the SSP capsules containing representative material for BFN Unit 1 have already been withdrawn from their host reactors. Therefore, subject to NRC approval of BFN Unit 1 participation in the BWRVIP ISP, operation of BFN Unit 1 at EPU conditions will not impact the withdrawal schedule for the BFN Unit 1 representative weld material capsules under the ISP.

NRC Request 8

Discuss the effects of the EPU on the Upper Shelf Energy of the beltline components and the welds of the Unit 1 reactor pressure vessel.

TVA Reply 8

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 Margins 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 Margins Analyses for the BFN Unit 1 limiting reactor vessel beltline materials is provided in Tables 3 and 4 that follow. These evaluations demonstrate adequate upper shelf energy margins for EPU operation.

Table 3

Browns Ferry Unit 1 Reactor Vessel Plate Upper Shelf Energy Equivalent Margin Analysis for 32 EFPY at EPU Conditions

Plate EMA 32 EFPY – Plant Applicability Verification Form

Surveillance Plate USE – Not Available:

%Cu	=	N/A	
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 B5864-1):

%Cu	=	0.15	
32 EFPY 1/4T Fluence	=	7.91 E+17 n/cm^2	
R.G. 1.99 Predicted % Decrease	=	13.5	(R.G. 1.99, Figure 2)
Adjusted % Decrease	=	N/A	(R.G. 1.99, Position 2.2)

13.5% ≤ 21%, so vessel plates are
bounded by equivalent margin analysis

Table 4

Browns Ferry Unit 1 Reactor Vessel Weld Upper Shelf Energy Equivalent Margin Analysis for 32 EFPY at EPU Conditions

Weld EMA 32 EFPY – Plant Applicability Verification Form

Surveillance Weld USE – Not Available:

%Cu	=	N/A	
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 406L44):

%Cu	=	0.27	
32 EFPY 1/4T Fluence	=	7.91 E+17 n/cm^2	
R.G. 1.99 Predicted % Decrease	=	23.5	(R.G. 1.99, Figure 2)
Adjusted % Decrease	=	N/A	(R.G. 1.99, Position 2.2)

23.5% ≤ 34%, so vessel welds are
bounded by equivalent margin analysis

NRC Request 9

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

TVA Reply 9

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 the 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 10

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

TVA Reply 10

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.

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. 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 5 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 5

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 Section 4.7	Change in nitrogen consumption rate
PUSAR 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 11

In Enclosure 4, Section 7.4, 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 11

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 proposed 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 12

Provide a description of the major differences in the Unit 1 operation; procedures; system configuration; and flow, pressure, and level setpoints as compared to those of Units 2 and 3.

TVA Reply 12

As discussed in Section 1.2.4 of Enclosure 4 of the license amendment application, part of Browns Ferry Unit 1 recovery processes is to update Unit 1 configuration to be operationally the same as Units 2 and 3. Consequently, there will not be any major differences in the Unit 1 operation as compared to Units 2 and 3. Unit 1 recovery activities are structured and implemented to ensure that Unit 1 will functionally operate and respond to postulated events similar to Units 2 and 3.

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. TVA's philosophy for Unit 1 recovery requires the same consistent configuration control approach be applied.

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 operation increases reactor vessel dome pressure (< 3%) to help provide sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine. This increase is identical to the pressure increase accomplished previously on Units 2 and 3 concurrent with the implementation of Power Uprate (105% OLTP). 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.

NRC Request 13

In Enclosure 4, Section 4.2.5 addresses protective coatings. Discuss the effect of extended shutdown on qualified coatings, the measures taken, and the inspection results.

TVA Reply 13

During the prolonged shutdown of Unit 1, the torus remained filled and chemistry was sampled in accordance with the chemistry program procedure. The drywell remained open to the reactor building, and the Units 1, 2, and 3 common reactor building ventilation remained in service.

During the Unit 1 shutdown, some systems were placed in lay-up status while others, such as the Fuel Pool Cooling System, for example, remained in service. Systems maintained in an operating status were operated and maintained in accordance with their specific system operating procedures. Periodic walkdowns were conducted and equipment was maintained to prevent system leakage. These measures provided a level of protection for the containment structures and their associated coatings against exposure to adverse environments during the Unit 1 shutdown period.

As part of Unit 1 restart activities, TVA has performed visual inspections of the torus in accordance with the BFN Containment Coatings Program and ASME Section XI examination requirements. These inspections included, to the extent practicable, a complete visual inspection of the coatings in the torus vapor space and immersion area. These inspections identified only minor degradation. Areas identified as degraded were repaired in accordance with plant procedures. TVA completely re-coated the torus immersion area to one foot above the normal high water level.

TVA has completed preliminary inspection of the Unit 1 drywell coatings in accordance with the BFN Containment Coatings program but has not yet completed the inspections required to comply with ASME Section XI. The inspections completed to date have identified only minor degradation. Random pull tests performed in the drywell on the liner plate coating material indicated the acceptance criteria of 200 psi could be met in all cases. A final inspection of the drywell coatings will be performed prior to startup.

Based on the results of the inspections, tests, repairs and recoat, the containment coating has not been adversely affected during the lay-up period.

References:

1. TVA letter, T. E. Abney 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.

2. NRC letter to TVA, "Browns Ferry Nuclear Plant, Units 1- Request for Additional Information for Extended Power Uprate, (TAC NO. MC3812) (TS-431)," dated December 30, 2004.

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

4. 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 (TAC M95087)," dated September 14, 1998.

5. TVA letter, T. E. Abney (TVA) to NRC, “Browns Ferry Nuclear Plant (BFN) Unit 1 – Supplemental Response to Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping,” dated July 21, 2004.

6. TVA letter, T. E. Abney (TVA) to NRC, “Browns Ferry Nuclear Plant (BFN) Unit 1 – Response to Request For Additional Information for Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping," dated April 25, 2005.

7. 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.

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.