December 22, 2005

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE (TS-431) (TAC NO. MC3812)

Dear Mr. Singer:

By letter dated June 28, 2004, as supplemented by letters dated August 23, February 23, April 25, 2004, and June 6, 2005, the Tennessee Valley Authority (the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) an amendment request for Browns Ferry Nuclear Plant, Unit 1. The proposed amendment would change the operating license to increase the maximum authorized power level from 3293 to 3952 megawatts thermal. These changes represent an increase of approximately 20 percent above the current maximum authorized power level for Unit 1. The proposed amendment would also change the Unit 1 licensing bases and associated Technical Specifications to credit 3 pounds per square inch gauge (psig) for containment overpressure following a loss-of-coolant accident and increase the reactor steam dome pressure by 30 psig. The NRC staff finds that a response to the enclosed Request for Additional Information is needed before we can complete the review.

This request was discussed with your staff on December 16, 2005, and it was agreed that a response would be provided within 75 days of the issuance of this letter. The NRC staff will be in contact at a later date to schedule the pending valve inspection. If you have any questions, please contact Ms. Eva Brown at (301) 415-2315.

Sincerely,

/RA/
Margaret H. Chernoff, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures:
1. Non-proprietary Request for Additional Information
2. Proprietary Request for Additional Information

cc w/enclosures: See next page

December 22, 2005

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE (TS-431) (TAC NO. MC3812)

Dear Mr. Singer:

By letter dated June 28, 2004, as supplemented by letters dated August 23, February 23, April 25, 2004, and June 6, 2005, the Tennessee Valley Authority (the licensee), submitted to the U.S. Nuclear Regulatory Commission (NRC) an amendment request for Browns Ferry Nuclear Plant, Unit 1. The proposed amendment would change the operating license to increase the maximum authorized power level from 3293 to 3952 megawatts thermal. These changes represent an increase of approximately 20 percent above the current maximum authorized power level for Unit 1. The proposed amendment would also change the Unit 1 licensing bases and associated Technical Specifications to credit 3 pounds per square inch gauge (psig) for containment overpressure following a loss-of-coolant accident and increase the reactor steam dome pressure by 30 psig. The NRC staff finds that a response to the enclosed Request for Additional Information is needed before we can complete the review.

This request was discussed with your staff on December 16, 2005, and it was agreed that a response would be provided within 75 days of the issuance of this letter. The NRC staff will be in contact at a later date to schedule the pending valve inspection. If you have any questions, please contact Ms. Eva Brown at (301) 415-2315.

Sincerely,
/RA/
Margaret H. Chernoff, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures:
1. Non-proprietary Request for Additional Information
2. Proprietary Request for Additional Information

cc w/enclosures: See next page

DISTRIBUTION: See next page

Package No.: ML053570231
ADAMS Accession No. ML053560120 TS: ML053570181 NRR-106

OFFICE	LPL2-2/PM	LPL2-2/PM	LPL2-2/LA	LPL2-2/BC
NAME	EBrown	MChernoff	BClayton	MMarshall
DATE	12/21/05	12/21/05	12/20/05	12/22/05

OFFICIAL RECORD COPY

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 — REQUEST FOR ADDITIONAL INFORMATION REGARDING EXTENDED POWER UPRATE (TAC NO. MC3812)

<u>Distribution</u>

PUBLIC
LPLII-2 R/F
RidsOgcRp
RidsAcrsAcnwMailCenter
RidsNrrPMEBrown
RidsNrrPMMChernoff
BClayton (hard copy)
RidsNrrDorlLpld
RidsNrrDorl
TAlexion
TChan
DTrimble
MKotzalas
LLund
RKaras
DFischer
MYoder
GGeorgiev
RPelton
KMartin
EThrom
HWalker
MHart
FAkstulewicz
MMasnik
HNash
GThomas
MRazzaque
THuang
ZAbdullahi
KManoly
SJones
CWu
JTatum
DReddy
AHowe
DThatcher
SWeerakkody
HLi
RPettis
RGallucci
BSheron
RJenkins
SKlementowiz
MRubin
NTrehan
CHinson
RPederson
MStutzke
SLaur
MMitchell
MKhanna
EImbro
JNakoski
AAttard
RidsRgn2MailCenter(SCahill)

REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

DORL

1. Sections 7.4 and 9.1.3 of the Power Uprate Safety Analysis Report (PUSAR) addresses the feedwater (FW) system and its response to transients. In related correspondence and meetings, it has been indicated that the FW pumps, heaters, and control system are being upgraded, but restrictions are being placed on the system to ensure “similar” operation across all units. Describe the measures put in place to ensure “similar” operation, and address how these changes do not result in more than a minimal increase in the frequency, likelihood, or consequences of occurrence of an accident, nor create or increase the likelihood of occurrence of a malfunction of components important to safety. This discussion should focus on common cause initiators and events resulting in unsatisfactory coolant inventory or temperature.

CVIB (EMCB-A)

1. a) According to the license renewal submittal for Browns Ferry Nuclear Plant (BFN), the applicant stated that the effective-power years (EFPYs) of operation at the end of license (EOL) extended term is 54 EFPYs for Unit 1. Based on previous plant experience, the staff requests that the licensee. Provide the projected neutron fluence (E > 1.0 MeV) for the reactor vessel beltline materials, including the extended power uprate (EPU) conditions, for the current licensing basis term and the extended period of operation for Unit 1.

   b) Provide the following information related to Unit 1, for the past operating history: (a) megawatt thermal power (MWt), (b) calendar years of operation, and (c) capacity factor. In addition, provide the information requested in items (a) through (c) with respect to the future projected operating conditions (i.e., with consideration of the extended period of operation and the EPU conditions). Also, indicate the projected number of EFPYs with consideration of the extended period of operation and the EPU conditions.

2. Tennessee Valley Authority (TVA) has committed to implement the Boiling-Water Reactor (BWR) Vessel and Internals Project (BWRVIP) Report, BWR Integrated Surveillance Program (ISP) (BWRVIP-116) report, for monitoring neutron embrittlement of the BFN Units, reactor pressure vessel beltline materials and welds, for the extended period of operation. Implementation of the EPU for the BFN units will change the neutron fluence values, which will affect the projected neutron fluence of the ISP

capsules. The BWRVIP-116 report states that implementation of the ISP during the extended period of operation provides additional data from host reactor capsules to meet surveillance monitoring needs of the BWR fleet for license renewal. The following table, which was extracted from the BWRVIP-116 report and the BWRVIP's responses to the staff's request for additional information on BWRVIP-116 and submitted by letter dated January 11, 2005, provides information on the current status of the ISP at BFN during the extended period of operation.

Representative Material	ISP Capsule EFPY	Estimated Fluence of the ISP Capsule (n/cm ^{2})	EOL 1/4 T Fluence of Target (n/cm ^{2})	Estimated EOLE 1/4 T Fluence of Target (n/cm ^{2})	ISP Capsule Fluence as a % of EOLE 1/4 T Fluence
Weld 406L44	N/A*	2.89 X 10^{18}	3.96 X 10^{17}	1.35 X 10^{18}	214.1%
Plate - Heat # A0981-1	40	1.37 X 10^{18}	4.89 X 10^{17}	1.66 X 10^{18}	82.5%

* This representative weld material is in the supplemental surveillance program (SSP) test capsules only; "ISP capsule EFPY" is not applicable. The highest SSP capsule fluence attained is used as the ISP capsule fluence.

According to the license renewal submittal, the extended term is for 54 EFPYs. Based on the projected EOL extended fluence, discuss how the surveillance program is affected by the EPU and provide a basis for this position. The applicant should provide information regarding the effect of the EPU on the BFN Unit 1 ISP capsule withdrawal schedule, the estimated fluence of the ISP capsules, EOL 1/4 thickness (T) target fluence values, estimated EOL extended 1/4 T target fluence values and supplemental capsule fluences as a percentage of EOL extended 1/4 T fluence, and the supplemental test capsules for BFN Unit 1.

3. According to the time-limited aging analysis contained in Section 4.7.7, Stress Relaxation of Core Plate Hold-Down Bolts, in the license renewal application (LRA), the hold-down bolts were addressed for the extended period of operation. EPU conditions will enhance the neutron fluence at the core plate region, which will consequently affect the preloading condition in the core plate hold-down bolts, due to stress relaxation. Provide a basis of how the EPU conditions will affect the integrity of the core plate hold-down bolts through the EOL extended term.

4. Austenitic stainless steel reactor vessel internal components are susceptible to irradiation-assisted stress corrosion cracking (IASCC) when exposed to higher neutron fluence due to EPU conditions. The four components listed below are susceptible to IASCC. Discuss the inspection programs in place, to address the aging affects due to IASCC, in the (a) top guide, (b) core shroud, (c) core plate, and (d) in-core instrumentation guide dry tubes and guide tubes, under EPU conditions. Specifically, discuss what BWRVIP inspections are required for each component and if there are any additional requirements beyond the BWRVIP requirements for those components. Additionally, address the impact of the EPU conditions on the inspection programs documented in the BWRVIP reports and any additional requirements for the EPU conditions.

5. According to Aging Management Program B.2.1.8, Boiling Water Reactor Feedwater Nozzle Program, in the LRA for BFN, the FW nozzles are inspected in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure

Vessel Code, Section XI, Subsection IWB (B.2.1.4), and the recommendations of the General Electric (GE) NE-523-A71-0594 Report, Alternate BWR Feedwater Nozzle Inspection Requirements. Describe the analysis that is performed on the FW nozzles, with respect to the GE topical report and how the EPU conditions will affect those analyses.

EQVA (IPSB-A)

1. Table 1 of the April 25, 2005, submittal provides a comparison of the proposed EPU testing program to the original startup testing described in Updated Final Safety Analysis Report (UFSAR) 13.5.2.3. Table 1, STP [Startup Test Procedure]10, describes the intermediate range monitor (IRM) Calibration/Performance test. During the initial test, the IRM-average power range monitor (APRM) overlap was checked and the IRM gains adjusted, as necessary, to improve the IRM system overlap between the source range monitors and IRMs. This adjustment was performed after the APRM heatup calibration and after the first heat balance calibration of the APRMs. Under the 'Testing Planned for EPU' column, Table 1 states that STP-10 is an EPU startup test. However, under the 'EPU Test Conditions' column, Table 1 states that STP-10 is not a startup test, but will be done during the first controlled shutdown following APRM calibration for EPU.

   Clarify whether IRM Calibration/Performance is a startup test and explain whether the test is proposed to be performed during the first controlled shutdown following APRM calibration versus after the APRM heatup calibration (per the initial test). Provide justification why changing when this test is performed is acceptable and meets the intent of the original test.

2. Page E-3 of the April 25, 2005, submittal states that Table 2 demonstrates that the applicable tests in Attachments 1 and 2 (of NUREG-800, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants LWR [Light-Water Reactor] Edition, Section 14.2.1) are addressed by the testing planned for BFN EPU implementation. However, Table 2 only provides a comparison of the steady state and transient tests from the initial BFN startup tests to those described in SRP 14.2.1. Some of the initial tests referenced in Table 2 are not proposed to be performed for EPU implementation (e.g., STP-17, 25, and 27). Further clarification is needed to explain how Table 2 demonstrates that the applicable tests of SRP 14.2.1 are addressed by the proposed EPU testing.

3. Table 2 of the April 25, 2005, submittal lists three SRP 14.2.1 tests (shield and penetration cooling systems, engineered safety feature (ESF) auxiliary and environmental systems, and calibrate systems used to determine reactor thermal power), which were not part of the BFN initial startup tests, but are listed as a standard procedure. Are these tests performed regardless of the EPU implementation?

4. Page E-8 of the April 25, 2005, submittal states that the postmodification testing (of the electro-hydraulic control (EHC) system) can be conducted by inserting simulated signals such as low EHC pressure and stop valve position and that this process has been used for the current operating configuration. Clarify whether the postmodification simulated signals will be performed at EPU conditions or at the current operating configuration.

5. Page E-3 of the April 25, 2005, submittal states that, "BFN UFSAR Section 13.5.2.2 presents a general description of the initial startup testing that was performed for Unit 1 and Section 13.5.2.3 for Units 2 and 3. These UFSAR sections provide the objectives and acceptance criteria for the initial startup tests. The objectives and acceptance criteria as modified to reflect operation at 120% reactor thermal power will be used for planned EPU tests."

   However, UFSAR Section 13.5 states that "[t]his section presents a general description of the startup test that was planned for Browns Ferry and has been retained in the FSAR as a historical reference only. This test description is not in conformance with Regulatory Guide 1.68 and should not be used as a model for future test programs."

   Based on the UFSAR Section 13.5 information, (1) explain why page E-3 and Table 1 of the submittal relies on the UFSAR Section 13.5 information as a basis for their specific acceptance criteria for the EPU testing program, when the UFSAR specifically states that it should not be used; (2) provide copies of the Units 2 and 3 Summary Report of Startup Tests dated May 23, 1975 and May 9, 1977, respectively; and (3) confirm whether or not the proposed EPU test plan is in conformance with Regulatory Guide (RG) 1.68. If the EPU test plan is not in conformance with RG 1.68, provide a justification.

6. Table 1 of the April 25, 2005, submittal describes the original STP 11, LPRM ([Local Power Range Monitor] Calibration), and states that the "method and approach used to perform LPRM calibration is not affected by EPU." Therefore, Table 1 indicates that the LPRM calibration will be performed in accordance with standard plant procedures at less than 90 percent of original licensed thermal power (OLTP). However, Tables 13.5-5 and 13.5-6 of the UFSAR states that STP 11 was performed at 95 to 100-percent power at the 100-percent flow control line. Explain why it is not necessary to perform STP 11 at 95 to 100-percent power at the 100-percent flow control line during power testing at EPU conditions.

7. Table 1 of the April 25, 2005, submittal describes the original STP 24, Bypass Valves, and states that the "no modifications to the turbine control valves or the turbine bypass valves are required for operation at the EPU conditions. Confirmation testing will be performed during power operation." Therefore, Table 1 indicates that the bypass valve testing will be performed in accordance with standard plant procedures at less than 90 percent of OLTP. However, Tables 13.5-5 and 13.5-6 of the UFSAR state that STP 24 was performed at 95 to 100-percent power at the 100-percent flow control line. In addition, both Table 1 and UFSAR Section 13.5.2.3 state that one of the purposes of STP 24 is to "demonstrate that the bypass valve can be tested for proper functioning at rated power without causing a scram." Explain why it is not necessary to perform STP 24 at 95 to 100-percent power at the 100-percent flow control line during power testing at EPU conditions.

8. Table 1 of the April 25, 2005, submittal describes the original STP 33, Main Turbine Stop Valve (TSV) Surveillance Test, and states that "individual main turbine stop valves must be closed periodically during plant operation as required for plant surveillance testing. As described in EPU safety analysis report Section 3.5.2, the TSV bounding closing time was utilized in EPU analysis." Table 1 indicates that the main TSV testing

will be performed in accordance with standard plant procedures at less than 90 percent of OLTP. However, Tables 13.5-5 and 13.5-6 of the UFSAR state that STP 33 was performed at 95 to 100-percent power at the 100-percent flow control line. In addition, both Table 1 and UFSAR Section 13.5.2.3 state that the purpose of STP 33 is to "demonstrate acceptable procedures for daily turbine stop valve surveillance test at a power level as high as possible without producing a scram."

Section 3.5.2 of the Enclosure 4 of the June 28, 2004, submittal, NEDC-33101P, DRF 0000-0010-9439, Browns Ferry Unit 1 Safety Analysis Report for Extended Power Uprate, or the PUSAR evaluates the main steam piping system and associated branch piping compliance with U.S.A. Standards, USAS-B31.1.10 Code stress criteria due to the 20-percent increase in flow due to EPU. This evaluation does not appear to meet the purpose of the original test. Explain why it is not necessary to perform STP 33 at 95 to 100-percent power at the 100 percent flow control line during power testing at EPU conditions.

9. Provide a table that describes the BFN Unit 1 EPU power ascension test plan. The table should provide the test/modification and the power level that the test/modification will be performed. An example of the information requested and the level of detail can be found in Attachment 3 of the Entergy Request for Additional Information response dated January 29, 2004, for the Waterford Unit 3 EPU (ADAMS Accession Number ML040340728).

10. Section B and Table 3 of the April 25, 2005, submittal briefly describe the impact of individual modifications on dynamic plant response. Provide a description of the process/methodology used in considering how, in the aggregate, the planned EPU modifications could affect expected system interactions, transient behavior of systems important to safety, functional system requirements in response to abnormal operating occurrences (AOOs), and other factors which could affect the dynamic response of the plant.

IOLB (IROB-B)

1. Describe any changes to the operator training program resulting from the proposed EPU.

REBB

1. Provide a copy of documentation sent to the U.S. Fish and Wildlife Service (USFWS) in response to the TVA's endangered species conference call with USFWS on October 27, 2005.

2. Page 4-2 of Enclosure 2, Browns Ferry Extended Power Uprate Environmental Report, of the submittal dated June 28, 2004, mentions the potential relocation of transmission line towers. Identify which transmission line towers might be relocated and where. Discuss whether surveys would be conducted to identify cultural and historical resources and protected species. Discuss whether this relocation would alter the vegetative management in the area of the relocated towers. Discuss whether new ground would be cleared for the relocation and whether such a relocation affects compliance with the

National Electric Safety Code. Describe any environmental impacts that might occur as a result of relocating transmission line towers.

3. Address whether there are any potential affects from increased noise on fauna due to the additional cooling tower operation mentioned on page 7-2 of Enclosure 2 to the submittal dated June 28, 2004.

4. Page 7-2 of Enclosure 2 of the submittal dated June 28, 2004, indicates that the report compiled consistent with the Environmental Protection Agency's Phase II rule for Section 316(b) of the Clean Water Act would be issued in the fall of 2005. Provide the Phase II 316(b) report that was estimated to be issued in the fall of 2005.

5. Identify any changes to the Enclosure 2 of the submittal dated June 28, 2004, including modifications and additional information, since completion of the Environmental Report in 2004. Beyond any identified changes, confirm the validity of the 2004 Environmental Report as it will be used in the U.S. Nuclear Regulatory Commission's (NRC's) environmental analysis.

SBPB (SPLB-A)

1. Refer to Section 7.4 of the PUSAR and Enclosure 7, Browns Ferry Extended Power Uprate Listing of Planned Modifications, and provide additional information for the following:

   a) Describe the impact that EPU will have on the plant response to the loss of a condensate pump and/or condensate booster pump, including a discussion of the reactor FW pump (RFP) response, design features that prevent a loss of all RFPs, how margins to RFP trip are affected, and any design or operational changes that are necessary to achieve acceptable performance.

   b) Describe the impact that EPU will have on the plant response to the loss of an RFP, including design features that prevent a loss of all RFPs, how margins to RFP trips are affected, and any design or operational changes that are necessary to achieve acceptable performance.

   c) Describe transient testing that will be performed to assure acceptable performance with respect to a) and b) above, or provide proper justification(s) for why testing is not warranted.

2. Implementation of the proposed EPU for BFN Unit 1 requires increased volumetric flow rates, which result in higher flow velocities and flow volumes in the existing piping systems for the uprate conditions. Provide the calculated flow velocities that will result due to the proposed EPU conditions, and compare them to the design criteria and industry guidelines for plant systems, such as main steam and associated systems, condensate and FW system, and other plant systems that are affected. Also, discuss in detail any dynamic loading and water hammer affects that the EPU will have on system functional and design capabilities.
3. Referring to SRP 3.4.1, describe the impact of EPU on flooding as a consequence of postulated tank and vessel failures.

4. Refer to Matrix 5, Section 2.5.1.2.1 of the letter dated February 23, 2005, which states that “[t]he BFN internally generated missiles evaluations are not impacted by BFN EPU.” In light of higher FW flows, possibly higher FW system pressures, and transient response following the proposed power uprate, discuss the basis for this conclusion.

5. Discuss the current turbine control and overspeed protection features/systems, and by referring to Section 7.1 of the PUSAR explain:

   a) The impact that EPU modifications will have on the existing turbine overspeed protection features and requirements, and how protection from turbine overspeed will continue to be assured, including turbine overshoot considerations,

   b) The changes that are required for the turbine overspeed protection trip setpoints, and,

   c) The impact that EPU will have on the capability to protect equipment important to safety from the effects of turbine missiles.

6. Referring to Section 10.1 and 10.2 and Table 10-1 of the PUSAR, explain why safety-related systems, structures, and components (SSCs) will not be affected due to postulated high energy line breaks and medium energy line breaks at the proposed EPU conditions. In Section 10.1.3, it is stated that the mass and energy releases for double-ended breaks and critical cracks in FW lines were re-analyzed at EPU conditions, but no conclusions were drawn regarding the protection of SSCs important to safety from these postulated breaks. Also, in Table 10-1, changes in mass releases are noted for FW line breaks in the steam tunnel or main steam valve vault and for reactor water cleanup breaks in the Reactor Building, but there is no discussion of the consequences and why they are acceptable. Please provide justification regarding acceptability of these releases or whether they are bound by the current licensing basis regarding the protection of SSCs from these postulated breaks.

7. Provide a discussion of the Turbine Gland Sealing System (TGSS), and confirm that the capability of the TGSS to contain activated nitrogen and to limit exposure to radiation will not be impacted by the proposed power uprate.

8. a) In Section 6.4.5 of the PUSAR, it is stated that:

   > the service water (UHS [ultimate heat sink]) temperature assumed in the DBA [design-basis accident] analyses was increased from 92 EF to 95 EF. Therefore, the TS [Technical Specifications] for UHS limits are changed to reflect these new analyses.

   Discuss all UHS licensing basis considerations and justify the proposed TS change with respect to these considerations, including confirmation that the analyses and assumptions that were used to justify the proposed change are the same as those used to justify the original UHS temperature limit. Also, confirm that data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than those assumed in the analyses that were performed.

b) Also, the last paragraph in Section 6.4.5 states that:

> The UFSAR includes a discussion relative to heatup of the downstream portion of the pool that would exist following the loss of the downstream dam on the Tennessee River. The river thermal rise post-shutdown would increase due to the increase in decay heat associated with EPU conditions but would not significantly affect this event.

Specifically, describe what the significance is with respect to water inventory and limiting water temperature considerations.

c) Explain how instrument uncertainties are accounted for when confirming that the TS limit is not exceeded.

9. Describe the impact that EPU will have on the capability of the liquid waste management system to limit offsite release of radioactive materials and to satisfy as low as reasonably achievable (ALARA) principles in accordance with the provisions of Title 10 to the *Code of Federal Regulations* (10 CFR) Section 20.1302; 10 CFR Part 50, Appendix I, Sections II.A and II.D; draft General Design Criteria (GDC)-70; and other licensing-basis criteria that apply.

10. Describe the impact that EPU will have on the capability of the gaseous waste management system to limit offsite release of radioactive materials and to satisfy ALARA principles in accordance with the provisions of 10 CFR 20.1302; 10 CFR Part 50, Appendix I, Sections II.A and II.D; draft GDC-70; and other licensing-basis criteria that apply.

11. Refer to Table 1, Comparison of BFN Initial Testing and Planned EPU Testing, in the letter dated April 25, 2005, and provide additional information for the following STP items:

a) STP 23 - Feedwater System:

Confirm that the FW system tests that are being conducted will include testing at the 100-percent EPU power level for the purposes described in the STP 23, which are:

i) to adjust the FW control system settings for all power and FW pump modes,

ii) to demonstrate stable reactor response to subcooling changes, and

iii) to demonstrate the capability of the FW system response in that one of the three operating FW pumps tripped and the automatic flow runback circuit acted to drop power to within the capacity of the remaining pumps, thereby preventing a reactor low water level scram.

b) STP 24 - Bypass Valves:

The original test description states that, "[o]ne of the turbine bypass valves was tripped open and closed. The pressure transient was measured and evaluated to aid in making final adjustments to the pressure regulator." Describe how the confirmatory test will be conducted to demonstrate:

i) the capability of the pressure regulator to minimize the reactor pressure disturbance while the plant is operating at 100-percent EPU power during an abrupt change in reactor steam flow, and

ii) that a bypass valve is capable of being tested for proper functioning at rated power without causing a high flux scram.

c) STP 25 - Main Steam Isolation Valves (MSIVs):

The original test description indicates that fast full closure testing of each MSIV was performed at hot standby and at selected power levels to determine the maximum power conditions at which individual valve full closure testing could be performed without causing a reactor scram, and that functional checks (10-percent closure) of each MSIV were performed at selected power levels above the maximum power condition for individual MSIV full closure. According to Table 1, these tests will not be repeated for EPU implementation. Explain how the maximum power conditions for performing individual MSIV full closure tests and functional tests during EPU operation will be determined such that MSIV testing during EPU operation will not result in a reactor trip.

d) STP 33 - Main Turbine Stop Valve Surveillance Test:

As described in STP 33, the purpose of this testing was to determine the highest reactor power level for performing daily TSV surveillance tests without causing a reactor scram. Describe how this power level will be determined for EPU operation.

12. By letter dated April 25, 2005 (TVA-BFN-TS-431), TVA provided additional information that was requested by NRC staff concerning power ascension testing planned for Unit 1 EPU conditions. In Table 1 of the letter, the licensee described the original startup tests and compared them to those planned for EPU. The tests that are planned for the Unit 1 balance-of-plant (BOP) systems are the same as those planned for the Units 2 and 3 uprates, which include elimination of transient tests such as main steam isolation (STP 25) and turbine trip/generator load rejection (STP 27). The justification that was provided for eliminating these tests for Unit 1 is the same as the justification that was provided for Units 2 and 3, even though:

C Unit 1 has not operated for more than 20 years, since its shutdown in 1985, C extensive modifications are necessary in order to fully repair and upgrade Unit 1 to make it similar in design and capability to Units 2 and 3,

C the EPU for Unit 1 involves a pressure increase of 30 pounds per square inch from the current licensed thermal power (CLTP) to EPU conditions, which (unlike the EPU for Units 2 and 3) is not a constant pressure power uprate, and

C no specific testing is proposed to demonstrate that the Unit 1 operating and transient performance characteristics are the same as for Units 2 and 3.

Given these considerations, the NRC has determined that the licensee's justification to eliminate power ascension tests for Unit 1 based on the operating experience of Units 2 and 3 is not acceptable, unless transient testing sufficient to demonstrate that the Unit 1 operating and transient response is essentially the same as the response for Units 2 and 3 has been completed.

Therefore, TVA is requested to describe in detail steady state and transient testing that will be completed in order to confirm that the design and operation of Unit 1 is essentially the same as Units 2 and 3 in all respects.

13. Units 2 and 3 have not been approved for EPU operation and have not operated at the uprated power level. Therefore, there currently is no operational data available to confirm that the EPU analyses for Units 2 and 3 are sufficiently accurate for demonstrating acceptable plant performance at the EPU power level without performing confirmatory transient testing. Because Unit 1 credits the analyses that were completed for Units 2 and 3 as justification for not performing transient testing for EPU operation, additional information is needed to explain in detail how the BOP transient response to postulated events and anticipated operational occurrences for Units 2 and 3 were evaluated and determined, consistent with the original power ascension test program. This information should include discussions addressing:

a) The BOP transient response criteria that are important for assuring reactor safety and for minimizing challenges to plant safety systems;

b) The nature, capability, applicability, accuracy, and sensitivity of the analytical modeling and methods that were used for assessing BOP transient response, including limitations restrictions, sensitivities and uncertainties associated with extrapolating the use of these methods to encompass EPU conditions;

c) Measures that have been taken to confirm and assure that the analytical models and methods accurately represent the BOP transient response and a description of how well predicted performance compares with actual performance, including to what extent analytical models and methods have been updated and corrected to reflect the actual response of BFN Units 2 and 3 behavior following plant transients that have occurred, the extent that BOP features are actually modeled and an explanation for why this is sufficient, and consideration of plant modifications and setpoint adjustments that have been made subsequent to plant transients that have occurred such that the effects of these changes are not represented by the existing plant response data;

d) The impact of plant modifications, setpoint adjustments and parameter changes that are planned on the validity, accuracy, sensitivity, and uncertainty of the analytical methods being used;

e) A comparison of the analytical results (as adjusted to account for uncertainties in the analytical modeling and analyses) to the acceptance criteria that have been established for BOP transient performance; and

f) Measures that are included in the power ascension test program for Unit 1 that will confirm the validity, accuracy, and sensitivity of the analytical results.

ACVB

1. Enclosure 3, Extended Power Uprate RS-001 Revised Areas of Review Matrix, of the letter dated February 23, 2005, Matrix 7, Section 2.7.2, addressed the ESF Atmosphere Cleanup.

   a) Address whether the high efficiency particulate air and carbon adsorber filters have sufficient capacity to mitigate DBAs with respect to contaminant retention, efficiency, and no impairment of function with the increased EPU source term.

   b) Clarify the extent to which the standby gas treatment system (SGTS) is shared among the three units and the impact of EPU on achieving a negative draw down pressure in the secondary containment.

   c) Identify the maximum SGTS inlet temperature under EPU operating conditions and its relationship to any design inlet temperature limitations.

   d) Clarify if the SGTS serves as the ventilation for spent fuel areas under DBA conditions and identify any impact resulting from EPU conditions such as fuel with higher burn up in the spent fuel pool (SFP).

2. Enclosure 3, Extended Power Uprate RS-001 Revised Areas of Review Matrix, of the letter dated February 23, 2005, Matrix 7, Section 2.7.3, addressed the Control Room (CR) Area Ventilation System. Describe what was considered in determining that there was no EPU effect. This discussion should include identification of the major cooling loads both inside the CR and outside the CR such as switchgear and motor control centers, that are cooled by this system and the potential to carry increased heat to the CR.

3. Enclosure 3, Extended Power Uprate RS-001 Revised Areas of Review Matrix, of the letter dated February 23, 2005, Matrix 7, Section 2.7.4, addressed the SFP Ventilation System.

a) Discuss whether the SFP area is normally ventilated through the reactor building ventilation system or some other system and provide information on the impact of EPU on that system.

b) Certain information was noted in Section 6.6 of the PUSAR for increases in area temperatures of the reactor building. Address whether there are other effects relative to higher burnup fuel in the SFP that need to be addressed.

c) Discuss whether there any effects due to EPU on the ventilation system that could result from loss of SFP cooling. For ventilation under accident conditions, reference can be made to the SGTS if appropriate.

4. Enclosure 3, Extended Power Uprate RS-001 Revised Areas of Review Matrix, of the letter dated February 23, 2005, Matrix 7, Section 2.7.5, addressed the Auxiliary and Radwaste Area Ventilation System. A note to Section 2.7.5 indicates that there was no EPU effect. Provide a discussion addressing what was considered in determining that there was no EPU effect. This discussion should include identification of major cooling loads in this area and increased cooling requirements due to higher temperature components that could result in higher room temperatures.

5. Enclosure 3, Extended Power Uprate RS-001 Revised Areas of Review Matrix, of the letter dated February 23, 2005, Matrix 7, Section 2.7.6 addresses ESF Ventilation Systems. A note to Section 2.7.6 indicates that there are not changes to the ESF ventilation as a result of EPU. Provide a description of what was considered in determining that there was no EPU effect. This discussion should include identification of major cooling loads in this area and increased cooling requirements due to higher temperature components that could result in higher room temperatures, impact on filter efficiencies and loading, and impact on flow rates, if any.

6. Section 4.1.1 of the PUSAR addresses containment pressure and temperature response. Verify that all input parameters to the containment peak pressure and temperature, minimum pressure, environmental and subcompartment analyses remain the same as those in the UFSAR except for those affected by the power uprate. For example: containment volume, heat sink descriptions, heat exchanger performance, equipment flow rates and flow temperatures, initial relative humidity, ultimate heat sink temperature etc. . . . Justify any changes made for the power uprate analyses.

7. Provide graphs of wetwell and drywell temperature and pressure for the large break loss-of-coolant accident (LOCA), anticipated transient without scram (ATWS), Station Blackout (SBO) and limiting Appendix R fire events.

8. In Section 4.1.1.1(a) of the PUSAR on bulk pool temperature it is noted that the heat exchanger k-factor (K) remains unchanged. Discuss why this is considered conservative and describe the program to ensure that this K value is not exceeded.

9. Table 4-1 of the PUSAR addresses the containment performance results. Explain Note 3 in Table 4-1, and explain the differences between the original and the M3CPT and LAMB methods.

10. Section 4.1.2.1 of the PUSAR discusses the LOCA loads. Explain why vent thrust loads are less at EPU conditions than those calculated during the Mark I Containment Long Term Program.

11. Section 4.1.2.3 of the PUSAR discusses subcompartment pressurization. Discuss why the 4-inch jet pump instrumentation line is the limiting break for subcompartment pressurization.

12. Verify that, upon a postulated loss of containment accident pressure and the assumed loss of the affected unit's ECCS pumps, the residual heat removal (RHR) suppression pool cooling function, the low pressure coolant injection function and the core spray function can be maintained by inter-ties with another unit. Describe how the operator accomplishes this. Address whether procedures exist for any unit to crosstie with either of the other units.

13. Describe how the secondary containment drawdown time is calculated. Describe the model of secondary containment and the standby gas treatment system. Address how external temperatures are factored into the model in accordance with Information Notice (IN) 88-76, Recent Discovery of a Phenomenon Not Previously Considered in the Design of the Secondary Containment Pressure Control, dated September 19, 1988. Provide a curve of pressure vs. time and describe how the EPU affects this calculation.

14. Section 4.1.1.1(b)of the PUSAR addresses local pool temperature with main steam relief valve (MSRV) discharge Explain the conclusion that the local pool temperature and steam ingestion criteria remain valid for EPU conditions.

15. Section 4.1.2.2 of the PUSAR discusses that the load definition for subsequent MSRV actuations in not affected by EPU. Provide the associated analysis (Reference 10).

16. Section 4.1.2.3 of the PUSAR addresses subcompartment pressurization. Provide a description or reference the assumptions and models used for the subcompartment analyses. Explain why the EPU pressure difference is greater than the CLTP pressure difference for the annulus pressure load.

17. Section 4.2.5 of the PUSAR addresses ECCS net positive suction head (NPSH). This section states that 157 ft^2 of unqualified paint was assumed in the calculation of ECCS strainer head loss. Discuss when this determination was made and why it is still valid. Include a discussion demonstrating that it bounds the actual unqualified paint for both units. Address how this unqualified paint is distributed between the ECCS suction strainers. Verify that there have been no changes to the ECCS suction strainer calculations, including debris generation, transport and head loss. Additionally discuss what temperature is assumed for the suppression pool water in the head loss calculations.

18. Provide a figure showing the minimum wetwell pressure and the pressures required to provide adequate available NPSH for the RHR and core spray pumps as a function of time after accident initiation. Discuss the minimum pressure difference between the pressure required to provide adequate available NPSH and the calculated minimum wetwell accident pressure.

19. Discuss the impact of crediting containment accident pressure for NPSH on operator response to the LOCA, ATWS, Appendix R fire and SBO events. Describe what changes to the emergency operating procedures (EOPs) are necessary and any operator actions necessary to ensure preservation of the necessary level of containment accident pressure for these four events. If none, please explain.

20. Discuss whether Unit 1 is subject to any single failures (e.g., vacuum breakers) that could result in the loss of adequate containment accident pressure.

21. Describe how containment leakage is modeled in the design basis NPSH calculations. Include L_a, MSIV leakage, air lock leakage and equipment hatch leakage.

22. Provide the design basis and realistic values used in the determination of ECCS pump available NPSH. The discussion should include realistic values from the EPU probabilistic risk analyses. These values should include:

    Service water temperature
    Initial containment temperature
    Initial containment pressure
    Initial drywell and wetwell humidity
    Initial suppression pool temperature
    Drywell and wetwell airspace volume
    Suppression pool volume

23. Discuss what indications would be available to the operator during a LOCA which could indicate abnormal ECCS pump performance, especially cavitation due to inadequate NPSH. Discuss what actions an operator would take in response to indications of inadequate ECCS pump NPSH.

24. Discuss whether reactor vessel isolation events have been considered as possibly more limiting than long term suppression pool heat up following a LOCA for ECCS pump available NPSH. Specifically, address the condition when the reactor vessel is isolated with high-pressure coolant injection unavailable and automatic depressurization system (ADS) is activated to proceed to safe shutdown. Assume that suppression pool cooling is not initiated until after ADS actuation.

25. Address whether the recommendations of GE Service Information Letter (SIL) 636 Revision 1 (related to the determination of decay heat) used for the containment calculations and the ECCS pump NPSH calculations.

26. Demonstrate with a "realistic" or best-estimate calculation of available net positive suction head for the RHR or core spray pumps, whichever is most limiting, whether credit for containment accident pressure is needed. All input and assumptions should be, to the extent possible, nominal values.

27. List the conservatisms included in the calculation of available NPSH and containment overpressure.

28. The Hydraulic Institute recommends margin above the required NPSH suppress cavitation. Discuss the need for crediting overpressure for the BFN pumps and how this margin is accounted for in your NPSH calculations. This discussion should include relevant quantitative information.

29. Provide the results of an analysis of the stuck open reactor vessel relief valve that demonstrate that adequate NPSH is available for successful operation of the ECCS pumps.

30. The NRC staff has reviewed Section 4.2.5 of the PUSAR and TVA's responses to NRC Bulletin 96-03, Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors. Describe how the BFN ECCS suction strainer design is consistent with the NRC staff SE report on the BWR Owners Group Utility Resolution Guidance (GE report NEDO 32686) dated August 20, 1998.

31. Describe the correlation used to determine the head loss due to debris for the ECCS suction strainers. Verify that this correlation is valid for the materials, temperatures and flow rates that will be encountered in the BFN suppression pool during a postulated accident or transient after the EPU.

32. Provide for staff review the NPSH calculations (including the containment calculations) for the Unit 1 core spray and RHR pumps at EPU conditions.

33. Section 4.7 of the PUSAR addresses post-LOCA combustible gas control. Indicate the time (days) for containment to reach the repressurization limit of half the design pressure (28 pounds per square inch gauge) due to nitrogen addition to the containment at EPU and verify it is within design basis.

34. Section 4.7 of the PUSAR indicates that the required start time for the containment atmosphere dilution system decreases from 42 hours to 32 hours, as a result of EPU. Clarify why the same time is indicated for BFN Units 2 and 3, which have a smaller increase of 15 percent from current licensed thermal power.

35. Table 4-1 of the PUSAR provides the peak drywell and wetwell pressure and temperature results at EPU using current methods. Provide these values at the original power level using current methods used for EPU for comparison of the effect of power level increase. Discuss the reasons for difference in these values with the original analyses (OLTP).

36. Section 4.7 of the PUSAR indicates that following EPU operation, the required 7-day volume of nitrogen increases from 155,000 scf to 197,000 scf, which exceeds the available 191,000 scf supply required by the TSs. The TS Bases of 7-days nitrogen storage requirement is conservative, because additional liquid nitrogen can be delivered within 1-day travel distance from two liquid nitrogen distribution facilities. The TS Bases will be revised to a 4-days nitrogen storage requirement to accommodate EPU. Please verify that the required nitrogen can be delivered to the site within 4- days time besides any natural phenomena occurring.

SBWB (SRXB-A)

1. GE recommended a surveillance program for monitoring Channel-Control Blade Interference in a letter dated July 14, 2005 [ADAMS Accession No. ML052000328]. Discuss whether the surveillance program recommended by GE will be implemented on Unit 1 before start-up. If the program is not implemented, describe what actions will be taken to ensure that the assumptions used in the safety analyses are maintained.

2. On June 24, 2005, GE submitted a 10 CFR Part 21 (Part 21) 60-day interim report notification with regard to critical power determination for GE-14 and GE-12 Fuel with Zircaloy spacers [ADAMS Accession No. ML051790237]. Discuss TVA's evaluation of the impact of the Part 21 on the critical power ratio (CPR) determination and the impact on R-factor. Explain in detail how this issue was resolved for Unit 1.

3. On March 29, 2005, GE submitted a Part 21 report involving the potential to exceed the low pressure technical specification limit [ADAMS Accession No. ML050950428]. The defect is a calculation of an AOO, which predicts that pressure regulator failure maximum demand (open) transient will be terminated by a high water level trip as a result of level swell in the reactor. Discuss TVA's evaluation of the safety concern reported by GE and explain in detail how this issue was resolved for Unit 1.

4. On August 16, 2004, GE submitted a Part 21 60-day interim notification with regard to narrow range reactor water level [ADAMS Accession No. ML042720293]. An evaluation by GE has determined that water level instruments may indicate high by as much as 8 inches should the reactor water level drop below the dryer seal skirt. Discuss TVA's evaluation of the safety concern reported by GE and explain in detail how this issue was resolved for Unit 1.

5. On August 24, 2004, GE submitted a Part 21 reportable condition and a 60-day interim notification with regard to non-conservative safety limit minimum CPR (SLMCPR). During performance of SLMCPR calculations for an extended operating domain condition, GE discovered an apparent flow impact where a lower flow condition at rated power had a more limiting SLMCPR than the rated flow condition. Discuss TVA's evaluation of the safety concern reported by GE and explain in detail how this issue was resolved for Unit 1.

6. Based on the various plant modifications and different fuel types the operating conditions and plant features of Units 1, 2, and 3 may be similar but not identical. Confirm whether the Unit 1 equilibrium core analysis was specifically performed for Unit 1 or only a single analysis was performed for all three units. If only one analysis was performed for all three units, discuss in detail the differences between the units if any and explain in detail why the conclusions given in the topical report are valid for Unit 1. Also, identify all differences in plant design and operating conditions between the units.

7. Section 1.3.2, Reactor Performance Improvement Features, of the PUSAR states that "[s]ome of the Unit 2/3 reactor performance improvement features have also been licensed for Unit 1." Also in Table 1-2, a list of performance improvement features are listed. It seems that this list is applicable for Units 2 and 3 rather than for Unit 1. Clarify whether the list is valid for Unit 1. If not, specify the features for Unit 1.

8. Table 1-2 of the PUSAR indicates that reactor dome pressure is increased by 30 pounds per square inch atmosphere (psia), but there is no change in the turbine stop valve (TSV) inlet pressure of 988 psia assumed in the analysis. Confirm that this value is correct. The turbine stop valve inlet pressure assumed for Units 2 and 3 is 926 psia, address why the TSV inlet pressure is different for Unit 1.

9. Section 1.2.2 Computer Codes, Table 1.3 of the PUSAR lists all the nuclear steam system codes used for the EPU request. This section indicates that the Unit 1 application of these codes complies with the limitations, restrictions, and conditions specified in the applicable NRC SE report that approved each code, with exceptions as noted in Table 1-3.

   During the review of Vermont Yankee for EPU, the staff identified several issues related to application of GE methods used for EPU evaluations. In response to the staff concerns, Entergy agreed to take penalties on certain parameters. Review the fuel vendor's analytical methods and code systems (neutronic, LOCA, transient, and accidents etc.) used to perform the safety analyses supporting the Unit 1 EPU application and provide the following information:

   (a) Confirm that the steady state and transient neutronic and thermal-hydraulic analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges.

   (b) Confirm that for the EPU conditions, the calculational and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions.

   (c) Confirm that the assessment database and the assessed uncertainty of models used in all licensing codes that interface with or are used to simulate the response of Unit 1 during steady state, transient or accident conditions remain valid and applicable for the EPU conditions.

10. In Section 2.1, Fuel Design and Operation, of the PUSAR it is stated that "[t]he average bundle power for EPU is 5.17 megawatts (MW)/ bundle." Specify the peak bundle power before and after the EPU.

11. The thermal hydraulic instability exclusion regions are not shown in Figure-1 of Section 2.3.1, Power/Flow Operating Map. Provide the figure showing the instability exclusion regions.

    In the proposed technical specification changes, TVA proposed to delete the Thermal Power versus Core Flow Stability Regions Figure 3.4.1-1. Address why this figure is being deleted, including a discussion addressing how the 10 CFR 50.36 requirements for TS will still be met with the deletion of this figure. Also, discuss where this figure will be relocated to and the corresponding document's administrative controls.

12. In Section 2.4, Stability, of the PUSAR, it is stated that thermal-hydraulic instabilities will be addressed in a separate submittal. Discuss the proposed submittal and provide a date when this information will be submitted for NRC staff review and approval.

13. On page 12 of the NRC Safety Evaluation for Licensing Topical Report, NEDC-32523P-A, Generic Evaluations of General Electric Boiling Water Reactor Power Uprate, or ELTR2, it is stated that "[i]n order to reduce the possibility of turbine overspeed trips, plant specific submittals must address the modifications described in GE SIL No. 480 and GE SIL No. 377 (or equivalent modifications)." Address whether the modifications recommended by SIL No. 377 have been completed on Unit 1.

14. With regards to Section 4.3 ECCS Performance, of the PUSAR:

    (a) Describe the first EPU core, and provide the batch size of GE-13 and GE-14.

    (b) Previous staff experience has seen that a power uprate would only have a small effect, typically less than 20 degrees Fahrenheit (°F), on peak cladding temperature (PCT). However, Table 4-5 indicate that the PCT for GE-13 fuel decreased by 30 °F and for GE-14 fuel, the PCT increased by 70 °F. Describe in detail why the PCT for GE-13 fuel is decreasing and the PCT for GE-4 fuel is increasing significantly (more than 50 °F, which is considered as significant).

15. Section 9.3.1, ATWS, Table 9-4 shows that the peak calculated suppression pool temperature is decreased from 214.6 °F to 214.1 °F. Discuss why the suppression pool temperature is decreasing as a result of EPU.

    Discuss whether the operator actions specified in the Unit 1 emergency operating procedures are consistent with the generic emergency plan guidelines/severe accident guidelines insofar as they apply to the operator actions for ATWS. Also, address whether the EOP operator actions are correctly assumed in the ATWS analyses. Specify the time delay used in the ATWS analysis for starting the standby liquid control pump.

16. Review Standard RS-001, BWR Template SE for Sections 2.8.5.1, 2.8.5.2.1, 2.8.5.2.2, 2.8.5.2.3, 2.8.5.3.1, 2.8.5.4.3, 2.8.5.5 and 2.8.5.6.1, guides the NRC staff to reach a conclusion regarding reactor coolant pressure boundary (RCPB) pressure limits not being exceeded. However in Enclosure 13, EPU RS-001 Revised Template Safety Evaluation, the revised template reflecting the BFN licensing basis does not designate acceptance criteria in the "Regulatory Evaluation" portion of each of these sections related to the RCPB. Provide a discussion of the corresponding draft GDC design requirement(s) and acceptance criteria for RCPB, and provide a markup of the SE template accordingly.

17. Sections 2.8.4.4 of RS-01 provides guidance regarding the RHR system. However, in Enclosure 13, EPU RS-001 Revised Template Safety Evaluation, the revised template contained in the letter dated February 23, 2005, reflecting the BFN licensing basis does not designate acceptance criteria in the "Regulatory Evaluation" portion of this section related to the RHR design requirements contained in GDC-34. Provide a discussion of

the corresponding draft GDC design requirements and acceptance criteria for RHR and provide a markup of the SE template accordingly.

18. Matrix 8 of Section 2.1 of RS-001 addresses new fuel and spent fuel storage. Draft GDC-40 and GDC-66, these GDC require that protection for ESFs shall be provided against dynamic effects and missiles that might result from plant equipment failures and criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. Provide a discussion on criticality of new and spent fuel storage for all intended fuel types.

19. Section 3.9.1 of the PUSAR discusses the shutdown cooling (SDC) analysis for EPU. It indicates that it takes [ ] for cooling down the reactor with two RHR pumps and heat exchangers in service. Furthermore, in Section 4.2.5 of the PUSAR, it was stated, “One RHR pump is required to operate during either the SBO or an Appendix R fire event.” The staff, however, notes that shutdown cooling with single SDC loop operation was not discussed in the PUSAR. Obviously, it is expected that SDC with one RHR pump and heat exchanger in service will take [ ] Clarify which criteria apply to SDC with single SDC loop operation, and whether the criteria are satisfied at EPU conditions. The response should also include whether EPU conditions will comply with the safe shutdown requirements of 10 CFR 50.63, Loss of all alternating current power, 10 CFR 50.48, Fire protection, GDC-3, Fire protection of Appendix A, General Design Criteria for Nuclear Power Plants to 10 CFR Part 50 and III.G, III.J, and III.L of Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, to 10 CFR Part 50, which may require a unit to achieve cold shutdown conditions within a given time.

20. In Section 2.1 of the PUSAR, it is stated that “[t]he additional energy requirements for EPU are met by an increase in bundle enrichment, an increase in the reload fuel batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length.” In order to obtain the additional 20-percent power, roughly 660 MWt at EPU condition, discuss whether the maximum enrichment level of fuel assemblies increased compared to the pre-EPU enrichment level. If so, by how much; and what impact does this have on new and spent fuel storage facilities?

21. Table 1-3 of the PUSAR, lists computer codes used for the EPU transient analyses. Clarify which code was used for the overpressure protection analysis.

22. In Section 4.3 of the PUSAR, it is stated that the Unit 1 licensing basis PCT at EPU for GE-14 fuel is a small recirculation discharge line break with battery failure, and that the break size is 0.06 ft^2. The PCT increases by 70 $^\circ$F (1830-1760 $^\circ$F) from the pre-EPU value. Provide the following additional information:

    (a) Discuss whether the current limiting LOCA event is the same as that for EPU;

    (b) Discuss whether a full break spectrum small break LOCA analysis performed to determine the limiting break size of 0.06 ft^2 based on a equilibrium core. If not, discuss why; and

(c) Discuss whether TVA intends to perform a full break spectrum small break LOCA analysis for cycle-specific LOCA analysis. If not, discuss why.

23. Provide the following information for the current SLMCPR calculation, as well as for the coming EPU calculation.

(a) The Reference Core Loading Pattern for the current cycle and the coming EPU cycle,

(b) The major differences with respect to reload batch fraction, core average weight percent enrichment, Core MCPR for limiting rod pattern, MCPR importance parameter and R-factor importance parameter,

(c) The core power shape experiencing through the operation for the current cycle and the coming EPU cycle, and

(d) The approved methodologies used for the SLMCPR calculation.

24. Based upon NRC staff review of GE methods on other EPU applications, additional penalties are required when evaluating the MCPR value that will result in the MCPR value increasing by 0.01 for the EPU condition. Discuss why the current SLMCPR value is still valid for the EPU operation.

25. Confirm that TVA did not take any deviations from Licensing Topical Report, NEDC-32424P-A, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, or ELTR1 and ELTR2. Also, confirm that TVA performed, or will perform, all the analyses required by the guidelines of ELTR1 and ELTR2. Identify the analyses to be performed before the EPU operation. Especially confirm that the transients listed in Appendix E of ELTR1 will be performed for the first EPU core.

Mr. Karl W. Singer
Tennessee Valley Authority

BROWNS FERRY NUCLEAR PLANT

cc w / Enclosures 1 and 2:
Mr. William D. Crouch, Manager
Licensing and Industry Affairs
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

cc w / Enclosure 1 Only:
Mr. Ashok S. Bhatnagar, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Larry S. Bryant, Vce President
Nuclear Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Brian O'Grady, Site Vice President
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Robert J. Beecken, Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

General Counsel
Tennessee Valley Authority
ET 11A
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. John C. Fornicola, Manager
Nuclear Assurance and Licensing
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Bruce Aukland, Plant Manager
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Masoud Bajestani, Vice President
Browns Ferry Unit 1 Restart
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Robert G. Jones, General Manager
Browns Ferry Site Operations
Browns Ferry Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Decatur, AL 35609

Mr. Scott M. Shaeffer
Browns Ferry Unit 1 Project Engineer
Division of Reactor Projects, Branch 6
U.S. Nuclear Regulatory Commission
61 Forsyth Street, SW.
Suite 23T85
Atlanta, GA 30303-8931

BROWNS FERRY NUCLEAR PLANT

Mr. Karl W. Singer
Tennessee Valley Authority

Mr. Glenn W. Morris, Manager
Corporate Nuclear Licensing
and Industry Affairs
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, AL 35611-6970

State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 303017
Montgomery, AL 36130-3017

Chairman
Limestone County Commission
310 West Washington Street
Athens, AL 35611