October 3, 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, 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 (BFN), Unit 1. The proposed amendment would change the operating license to increase the maximum authorized power level from 3293 to 3952 megawatts thermal. This change represents 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 gage (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.

Additionally, since the shutdown of BFN Unit 1 in 1985, the reactor has been defueled and systems have not operated. As part of the recovery plan for the restart of BFN Unit 1, many modifications and replacements are being made to plant equipment, programs, and procedures. The Extended Power Uprate (EPU) Review Standard RS-001 makes certain assumptions regarding the condition of components and implementation of programs. As Unit 1 was shut down, the NRC staff did not perform an evaluation of the Unit 1 motor-operated valve program under Generic Letter (GL) 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance, or GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves. Therefore, these evaluations will need to be completed, in addition, to reviewing any impact from the request to operate at EPU conditions. The NRC staff is currently reviewing background material provided by the licensee in support of a planned on-site inspection, which is tentatively scheduled for October/November 2005. Additional information regarding the inspection will be provided at a later date.

Consistent with NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 19 and Regulatory Guide 1.174, the licensee's risk analyses used to support a license application and the level of detail of the review of those analyses, should be commensurate with the degree of rigor needed to provide a valid technical basis for the NRC staff's decision. As for EPUs, those licensees who do not request the relaxation of

any deterministic requirements for their proposed power uprates, the NRC staff's approval is primarily based on the application meeting the current deterministic engineering requirements. Thus, the purpose of the NRC staff's risk review is to determine if there are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the deterministic requirements in the regulations. Such issues could represent the "special circumstances" that would call for a more detailed risk review. Based on the uniqueness of BFN's issues and the fact that there has never been an industry peer review of the Unit 1 probabilistic risk assessment, the NRC staff intends to perform an audit to determine if any issues adversely affect the presumption of adequate protection.

This request was discussed with your staff on August 29, 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 and risk audit. If you have any questions, please contact Ms. Eva Brown at (301) 415-2315.

Sincerely,

/RA by Eva Brown for/

Margaret H. Chernoff, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosure: As stated

cc w/encl: See next page

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

Date: October 3, 2005

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REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE (EPU)

TENNESSEE VALLEY AUTHORITY (TVA)

BROWNS FERRY NUCLEAR PLANT (BFN), UNIT 1

DOCKET NO. 50-259

EMCB-C

Flow Accelerated Corrosion (FAC)

1. The FAC monitoring program includes the use of a predictive method to calculate the wall thinning of components susceptible to FAC. Provide a sample list of components for which wall thinning is predicted and measured by ultrasonic testing or other method. Include the initial wall thickness (nominal), current (measured) wall thickness, and a comparison of the measured wall thickness to the thickness predicted by the CHECWORKS™ FAC model.

2. EPU will affect several process variables that influence FAC. Identify the systems that are expected to experience the greatest increase in wear as a result of EPU and discuss the effect of individual process variables (i.e., moisture content, temperature, oxygen, and flow velocity) on each system identified.

Protective Coating Systems

3. TVA's February 23, 2005, response states:

> Previous testing was performed which bounded peak accident conditions for all but one specific coating configuration. Therefore, TVA is performing confirmatory testing to ensure that all qualified coating configurations have been tested.

In regards to this statement provide a discussion explaining what the specific coating configuration is, how large the affected area is, what specific testing was performed, the results of the confirmatory testing, and how the confirmatory testing is correlated to the coating's original design basis accident qualification.

EEIB-B

1. Address and discuss the following points:

    a. Identify the nature and quantity of Mega volt-amp reactive (MVAR) support necessary to maintain post-trip loads and minimum voltage levels.

    b. Identify what MVAR contributions BFN Unit is credited for providing to the grid.

c. After the power uprate, identify any changes in MVAR associated with Items a and b above.

d. Address the compensatory measures that the licensee would take to compensate for the depletion of the nuclear unit MVAR capability on a grid-wide basis.

e. Evaluate the impact of any MVAR shortfall listed in Item d above on the ability of the offsite power system to maintain minimum post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document information exchanges with the transmission system operator.

2. Page 6-1 of Enclosure 4 of the June 28, 2004, submittal states that the study documented that no additional changes are required for BFN's offsite power system to continue to meet Title 10 the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criteria (GDC)-17 requirements. Because the BFN construction permits were issued prior to the May 21, 1971, effective date of the GDC, compliance to these criteria may not be required as part of the BFN Units 1, 2, and 3 licensing basis.

   State whether BFN Unit 1 is consistent with GDC-17 or the Atomic Energy Commission Criterion 39.

3. The submittal states that transmission system operating guides will be issued to the load dispatcher prior to EPU operation, detailing any system operating constraints and any actions that may be required, including prompt communication with the control room. What protocol has been established with the transmission system operator to communicate to the licensee the availability of the transmission lines to provide sufficient voltage following a plant trip or when voltages would not be adequate?

4. Provide in detail and compare the **existing ratings** with the uprated ratings and the effect of the power uprate on the following equipment:

   a. Main generator rating and power factor
   b. Isophase bus, and modifications to the cooling system
   c. Detailed description of the replaced main power transformers
   d. Unit Auxiliary/Start-up transformers
   e. Main Generator breaker

5. Provide the list of loads affected by the power uprate change. Identify the motor loads before and after the power uprate change.

6. Provide the coping duration and recovery time expected from a station blackout (10 CFR 50.63). Discuss whether there is any change in the coping duration and recovery time for station blackout (10 CFR 50.63).

7. Page 6-2 of Enclosure 4 of the June 28, 2004, submittal and Page 6-2 of Enclosure 5 of the June 28, 2004, submittal state that Units 1 and 2 share four independent safety-related diesel generator units coupled as an alternate source of power, to four independent 4160 volt buses. Have the design and operation changed since Unit 1 was shut down in 1985? Describe the onsite alternating current power system for Unit 1.

EMCB-A

1. Section 10.7, Plant Life, of Enclosure 4 of the June 28, 2004, submittal identifies irradiation-assisted stress-corrosion cracking (IASCC) as a degradation mechanism influenced by increases in neutron fluence and reactor coolant flow. This section indicates that the current inspection strategy for reactor internal components is expected to be adequate to manage any potential effects of EPU operating conditions. Note 1 in Matrix 1 of Section 2.1 of RS-001, Revision 0 indicates that guidance on the neutron irradiation-related threshold for IASCC in boiling-water reactors (BWRs) is in Boiling-Water Reactor Vessel and Internals Program (BWRVIP) report BWRVIP-26. The "Final License Renewal SER [Safety Evaluation Report] for BWRVIP-26," dated December 7, 2000, states that the threshold fluence level for IASCC is 5×10^{20} n/cm^2 (E> 1 MeV).

   Identify the vessel internal components whose fluence, at the end of period of operation with the EPU operating conditions, will exceed the threshold level and become susceptible to cracking due to IASCC. For each vessel internals component that exceeds the IASCC threshold, either provide an analysis that demonstrates failure of the component will not result in the loss of the intended function of the reactor internals or identify the inspection program to be utilized to manage IASCC of the component. Identify the scope, sample size, inspection method, frequency of examination and acceptance criteria for the inspection programs.

EMEB-B

1. Discuss the plans to implement an Inservice Testing (IST) Program for restart of BFN Unit 1 that is consistent with the licensee's American Society of Mechanical Engineers (ASME) Code of record and incorporates appropriate changes in light of applicable EPU operating conditions. In particular, discuss, with examples, the evaluation of the impact of EPU conditions on the performance of safety-related pumps, power-operated valves, check valves, and safety or relief valves, including consideration of changes in ambient conditions and power supplies (as applicable), and indicate any resulting adjustments to the IST Program resulting from that evaluation.

2. Section 3.7, Main Steam Isolation Valves, of Enclosure 4 of the June 28, 2004, submittal states that the 24-percent increase in steam-flow rate will result in a decrease in the stroke time for the main steam isolation valves (MSIVs) but that the stroke time will continue to satisfy the Technical Specifications. Describe the basis for this assumption using design, test, and operational experience of the MSIVs.

3. Section 4.1.3, Containment Isolation, of Enclosure 4 of the June 28, 2004, submittal states that parameters for air-operated valves (AOVs) and solenoid-operated valves

(SOVs) were reviewed, and no changes to the functional requirements of any AOVs or SOVs were identified as a result of EPU operating conditions. Discuss, with examples, the evaluation of safety-related AOVs and SOVs used for containment isolation and other safety functions for potential impact from EPU operation.

4. Section 4.1.4, Generic Letter (GL) 89-10 Program, of Enclosure 4 of the June 28, 2004, submittal states that process and ambient parameters for motor-operated valves (MOVs) were reviewed, and no changes to the functional requirements of GL 89-10 MOVs were identified as a result of EPU operating conditions. In support of the EPU review, discuss, with examples, the evaluation of safety-related MOVs for the potential impact from EPU operation, including the impact of increased process flows on operating requirements and increased ambient temperature on motor output.

5. Section 4.1.6, GL 95-07, of Enclosure 4 of the June 28, 2004, submittal states that MOVs used for containment or high energy line break isolation have been reviewed for the effects of operations at EPU conditions, including pressure locking and thermal binding. The licensee provided a response to GL 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, in a submittal dated May 11, 2004. Discuss, with examples, the evaluation of safety-related power-operated gate valves in light of any changes in ambient temperature on the potential for pressure locking or thermal binding resulting from EPU operation.

6. Section 10.4.3, Main Steam Line, Feedwater and Reactor Recirculation Piping Flow Induced Vibration Testing, of Enclosure 4 of the June 28, 2004, submittal discusses the plans for vibration monitoring during initial plant operation for the new EPU operating conditions. Discuss in more detail, the procedures for avoiding adverse flow effects during power escalation and after achieving EPU conditions, including specific hold points and duration, inspections, plant walkdowns, vibration data collection methods and locations, planned data evaluation, and decision criteria for reducing plant power level or initiating plant shutdown.

7. In the submittal dated February 23, 2005, the licensee lists modifications planned to support EPU operation on pages E1-17 to 22. Discuss the modifications planned to safety-related pumps and valves and the actions to provide assurance of their capability to perform the applicable safety functions under EPU conditions.

8. In the submittal dated February 23, 2005, the licensee indicates on page E1-21 that the GL 89-10 MOVs will be modified to accommodate the 30 psi increased reactor operating pressure. The licensee states that MOV setup will be accomplished per MOVATS. Address its implementation of GL 89-10 and GL 96-05 for the safety-related MOVs.

9. In the submittal dated February 23, 2005, the licensee states on page E1-26 that acoustical circuit analyses have been developed to identify the contributions to flow-induced vibration effects from main steam line components, junctions, and connections. Discuss the capability of such analyses to identify the excitation sources for flow-induced vibration effects in light of recent industry experience, and discuss possible alternative methods to identify excitation sources.

10. In the submittal dated February 23, 2005, the licensee states on page E1-27 that TVA had performed a detailed peer review of the General Electric Steam Dryer load definition methodology and analysis, and that the peer review had provided TVA with assurance that all phases of the analysis were adequate. Describe the design-load definition for its steam dryer at BFN Unit 1 and the basis for the adequacy of the load definition.

11. On page E1-28 of the submittal dated February 23, 2005, the licensee states that the uncertainty in its steam dryer analysis will be reduced by the collection of plant-specific data during power ascension. On page E1-30, the licensee states that benchmarking of the acoustic circuit analysis for determining plant-specific loads is in process against a scale model test facility. Provide the details of acoustic circuit methodology and analysis, including validation, results, and uncertainty range of the methodology and analysis. Also, discuss the modifications made to its acoustic circuit model based on lessons learned from recent industry operating experience.

12. On page E1-28 of the submittal dated February 23, 2005, the licensee states that power ascension information will be collected at each of the EPU power ascension test plateaus and compared against the stresses in the design analysis of record. Discuss the specific process for collecting, evaluating, and incorporating plant data into the design stress analysis for the steam dryer during the planned EPU power ascension.

13. On page E1-30 of the submittal dated February 23, 2005, the licensee lists proposed modifications to the steam dryer based on lessons learned from recent BWR dryer modifications. Provide detailed descriptions and diagrams of the proposed modifications to the steam dryer. Also, describe the stress analysis performed for the modified steam dryer, and the resulting changes in predicted stress in comparison to the licensee's acceptance criteria at significant locations on the steam dryer.

14. On pages E1-33 to 36 of the submittal dated February 23, 2005, the licensee discusses the potential impact of temperature changes from resulting from EPU operation on mechanical equipment environmental qualification. The discussion focuses on the impact of temperature changes on non-metallic materials. Discuss the evaluation and potential impact of temperature changes on motor output of applicable safety-related MOVs resulting from EPU operation.

IPSB-B

1. Section 8.6, Normal Operations Off-Site Doses, of Enclosure 4 of the June 28, 2004, submittal states that radiation from shine (offsite) is not presently a significant exposure pathway and is not significantly affected by EPU. This conclusion is based on the experience of earlier 5-percent power uprates for Units 2 and 3. Also, Section 8.2.2, Offsite Doses at Power Uprate Conditions, of the Environmental Report states that N-16 activity in the Turbine Building will increase linearly with EPU.

   The magnitude of the N-16 source term in the Turbine Buildings is not a simple linear increase with reactor power. The equilibrium concentration of N-16 in the Turbine Building systems will be effected (an inverse exponential function) by the decreased

decay resulting from the increased steam/feed flow between the reactor and the Turbine Building. Implementation of hydrogen injection water chemistry also increases N-16 concentrations in reactor steam independently of reactor power.

Provide the present nominal value for the skyshine external dose component (assuming all three units operating at current licensed power levels), the corresponding estimated dose component following EPU (assuming all three units operating at the requested power, and design basis steam activity, levels). Include all parameters (i.e., flow rates, system component dimensions, etc.) used in calculating these values and specify the calculational method used. Identify the limiting dose receptor (i.e., is the dose receptor a member of the public located offsite and, therefore, subject to the dose limits of 40 CFR Part 190) or a member of the public working onsite (subject to the dose limits of 20.1301)). Describe any increases in doses for onsite spaces (i.e., Administrative offices, guard stations, etc.) continuously or routinely occupied by plant visitors or staff.

2. Section 8.5.3, Post Accident, of Enclosure 4 of the June 28, 2004, submittal states that plant specific analysis for NUREG 0737, Item II.B.2. “have been performed” but gives no results or indication they meet the NUREG 0737 acceptance criteria. For each BFN Unit 1 vital area (as defined in Item II.B.2.), provide the calculated pre-uprate and post-uprate mission doses to an operator performing vital tasks following a loss-of-coolant accident (LOCA). Verify that the mission doses to personnel in these vital areas, as well as the calculated dose estimates for personnel performing required post-accident duties in the plant’s Technical Support Center, are within the dose guidelines of GDC-19 (10 CFR Part 50, Appendix A). Is restoring spent fuel cooling a vital action required to mitigate the effects of a design basis LOCA at Unit1?

3. Section 8.4.2, Activated Corrosion Products, of Enclosure 4 of the June 28, 2004, submittal states that the increase in the activated corrosion product activity will be 3-percent higher than the original design basis activity. Provide the basis for this estimated increase. Since Unit 1 has been shut down for 20 years, how was the quantity of loose corrosion products (i.e., available for transport into the reactor) estimated?

   The increased steam EPU flow is likely to result in an increased moisture carryover in the steam, resulting in an increased transport of non-volatile fission products, actinides, and activated corrosion and wear products from the reactor coolant to the balance of the plant. Provide the levels of moisture carry over expected at the EPU steaming rates, and discuss its potential impact on activity buildup and resultant dose rates in the balance of plant.

4. Section 6.3.2, Crud Activity and Corrosion Products, of Enclosure 4 of the June 28, 2004, submittal indicates that the expected increase in spent fuel pool (SFP) crud is 2 percent, based on the expected increase of crud in the reactor coolant system (RCS) due to increased feed flow. Since Unit 1 has been shut down for 20 years, how were the pre-EPU crud levels determined? Describe the impact of a 20-percent increase in feedwater flow has on condensate demineralizer efficiency.

5. Describe the controls implemented throughout the extended shutdown of Unit 1 to minimize the corrosion of reactor water systems.

6. Also, the estimate of the increase in RCS activity does not appear to include pre-outage crud bursts. Recently, a number of BWRs that have implemented hydrogen water and Zinc injection chemistry, have experienced large, unprecedented, crud bursts. Describe any contingencies that will be implemented to compensate for any unexpected build-up and release of crud in Unit 1.

7. Section 6.3.3, Radiation Levels, of Enclosure 4 of the June 28, 2004, submittal states that the normal radiation levels around the SFP may increase slightly, primarily during fuel-handling operations. Explain the reason for, and the magnitude of, these postulated increases in dose-rate levels in the area of the SFP. Verify that these postulated dose-rate increases will be bounded by the current radiation zone designations in the SFP area. If this postulated dose rate increase is due to higher activation of spent fuel assemblies, discuss any effects that the storage of these spent fuel assemblies in the SFP may have on dose rates in accessible areas adjacent to the sides or bottom of the SFP.

8. Section 8.5.1, Normal Operations, of Enclosure 4 of the June 28, 2004, submittal states that, due to the conservative shielding design, the increase in radiation levels resulting from EPU will not affect the radiation zones for the various areas of the plant. This appears to be based on an assumed linear increase in radiation source term with power level. However, the increase in N-16 activity in the turbine building is an inverse exponential function with decay time, not a linear function of reactor power. Verify that the radiation zoning in all areas containing the steam and feed systems will be unaffected by EPU.

9. Section 8.5.2, Normal Post-Operations, of Enclosure 4 of the June 28, 2004, submittal states that the post-operation radiation levels in most areas of the plant are expected to increase by no more than the percentage increase in power level. This section also states, however, that there are a few areas near the reactor water piping and liquid radwaste equipment where the expected radiation level increase could be slightly higher. Provide the specific locations of these areas where higher dose rates are predicted, give the reasons for the expected increase in radiation levels in these areas, and state the percentage increase in dose rates expected.

10. Enclosure 8, Table 2 of the June 28, 2004, submittal states that the objective of test STP 1, Chemical and Radiochemical, is not applicable to EPU and is not required. The Table 1 entry for STP 1 states that “samples will be taken and measurements will be made at selected EPU power levels. . . .” Describe which samples and measurements will be made and at what power levels. Considering that Unit 1 has been shut down for approximately 20 years, justify why the original full startup test STP 1 is not appropriate.

11. Enclosure 8, Table 2 of the June 28, 2004, submittal states that the objective of test STP 2, Radiation Measurements, is not applicable to EPU and is not required. The Table 1 entry for STP 2 states that “Gamma dose rate measurements. . . will be made

at specific limiting locations throughout the plant. . . ." Describe the limiting locations for which measurements will be made and at what power levels. Considering that Unit 1 has been shut down for approximately 20 years, and the uncertainties of predicting the activated corrosion source term, justify why the original full startup test STP 2 is not appropriate to provide a new baseline for dose data on activity buildup.

12. Summarize the major Unit 1 plant hardware or system modifications involved in the requested EPU and discuss the change in occupational doses associated plant operation with the modifications in place.

SPLB-A

Spent Fuel Pool Cooling and Cleanup System (SFPCCS)

1. Section 10.5.5 of the Updated Final Safety Analysis Report (UFSAR), Revision 17 dated August 30, 1999, revised the discussion from the UFSAR that was previously provided regarding the maximum SFP heat load for batch and full core offloads. In order to facilitate NRC review of the capability of the SFPCCS to perform its function for EPU conditions, provide a discussion on the safety-related systems required to maintain fuel pool cooling within design bases temperature limits.

2. For EPU conditions, explain how the SFP water temperature will be maintained below 150 degrees Fahrenheit (F) for the worst-case normal (batch) and full core offload scenarios assuming a loss of offsite power and (for the batch offload only) a concurrent single active failure considering all possible initial configurations that can exist. Include a description of the maximum decay heat load that will exist in the SFP for each case, how these heat loads were determined, such that they represent the worst-case conditions, and what the cooling capacity is for the systems that are credited, including how this determination was made. Also:

    a. Describe any operator actions that are required, how long it will take to complete these actions, and how this determination was made; and

    b. Describe the maximum core decay heat load that will exist at the onset of fuel movement, how this determination was made, how this heat load will be accommodated while also satisfying the SFP cooling requirements over the duration of the respective fuel offload scenarios, and including the situation where the SFP is isolated from the reactor vessel cavity.

3. Discuss how adequate SFP makeup capability is assured for EPU conditions in the unlikely event of a complete loss of SFP cooling capability, including how the maximum possible SFP boil-off rate compares with the assured makeup capability that exists, operator actions that must be taken, how long it will take to complete these actions and how this determination was made, and boron dilution considerations.

4. Provide justification and/or details of the evaluation which concludes that the SFP cooling and makeup systems continue to meet the requirements of draft GDC-4 for EPU conditions, in so far as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

<u>Service Water Systems</u>

5. In Section 6.4.1.1, of Enclosure 4 of the June 28, 2004, submittal, regarding the emergency equipment cooling water (EECW) system, it is stated that: "EPU does not significantly increase equipment cooling water loads, and thus, the capacity of the EECW system remains adequate." Discuss, in more detail, the impact of the proposed EPU on EECW heat loads, flow rates, and flow velocities for the worst-case conditions, including limiting assumptions, input parameters, and available margin that will remain.

6. In Section 6.4.1.1.2, of Enclosure 4 of the June 28, 2004, submittal regarding the residual heat removal service water (RHRSW) system, it is stated that:

   > The post-LOCA containment and suppression pool responses have been calculated based on an energy balance between the post-LOCA heat loads and the existing heat removal capacity of the RHR and RHRSW systems. As discussed in Sections 3.11 and 4.1.1, the existing suppression pool structure and associated equipment have been reviewed for acceptability based on this increased suppression pool temperature. . . . The RHRSW system flow rate is not changed.

   Discuss, in more detail, the impact of the proposed EPU on the RHRSW system heat loads (including SFP cooling considerations), flow rates, and flow velocities for the worst-case conditions, including limiting assumptions, input parameters, and available margin that will remain.

7. Provide a description of any impacts that the proposed EPU will have on the issues described in GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," and GL 96-06, Supplement 1, including the basis for your determination. In particular, confirm that the assumed heat transfer capabilities of heat exchangers are consistent with heat exchanger performance testing that has been completed in accordance with GL 89-13 and corrected for worst-case conditions; and that water-hammer and two-phase flow analyses that were completed in accordance with GL 96-06 continue to be valid.

8. For EPU conditions, provide justification and/or details of the evaluation which concludes that the safety-related service water systems will continue to meet the requirements of draft GDC-4, in so far as it requires that reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

SPLB-B

1. Discuss whether any administrative controls or fire protection responsibilities of plant personnel are affected by an increase in decay heat. Also, address why an increase in decay heat will not result in an increase in the potential for a radiological release from a fire.

2. Section 6.7.1, of Enclosure 4 of the June 28, 2004, submittal states that:

> . . . a plant-specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions. . . . The results of the Appendix R evaluation for EPU provided in Table 6-5 demonstrate that fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions.

Upon reviewing Table 6-5, Browns Ferry Appendix R Fire Event Evaluation Results, the NRC staff was able to find references for all but the following values in the EPU submittal:

C Cladding Heatup (peak clad temperature (PCT)), degrees F = 1428 (EPU)
C Suppression Pool Bulk Temperature, degrees F = 227 (EPU), # 227 (Appendix R Criteria), including Note 3
C Primary Containment Pressure, pounds per square inch gage = 13.6 (EPU)

Provide references, including appropriate extracts from the UFSAR, plant-specific Appendix R evaluation, etc., for these values in Table 6-5, including Note 3.

3. Section 6.7.1 of Enclosure 4 of the June 28, 2004, submittal states that:

> . . . [f]or this [bounding PCT] case, the time available to the operator to open three MSRVs [main steam relief valves] is reduced from 30 minutes to 25 minutes at the EPU conditions. This reduction in the time available does not have any effect because the current procedures require this action to be completed within 20 minutes. Although the analysis assumes the time available to perform this operator action is reduced by five minutes. . ., five minutes of margin remain compared to the present analysis.

Discuss the time-line analyses, including any assumptions, that may have been made in determining that the action can confidently be accomplished within 20 minutes, such that the 5-minute reduction in available time “does not have any effect.”

4. The June 6, 2005, Reply 6 of Enclosure 4, states that:

> . . . 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.

Discuss whether that latter phrase “alternate. . . available” is intended as additional to the former phrase “a fire. . . area” or as a contingency if the first phrase does not apply.

That is, does Volume 1 of the BFN Fire Protection Report (FPR) ensure “that a fire in one area will not affect the equipment in another area” exclusively, or does it do so only if “alternate shutdown paths capable of controlling each of the units are [not] available?”

5. Section 6.7.1 of Enclosure 4 of the June 28, 2004, submittal as supplemented by the reply dated June 6, 2005 (including the discussion for the ATRIUM-10 fuel), states that “spurious operation of HPCI [high pressure coolant injection] was reviewed in accordance with [Volume 1 of 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 [main steam lines] is greater than 6 minutes. Therefore, the procedures will require HPCI isolation prior to 6 minutes during an Appendix R event.” Volume 1 of the BFN FPR addresses pre-EPU conditions, so the conclusion regarding the greater than 6-minute time for the reactor vessel water level to reach the MSLs presumably applies to pre-EPU conditions.

   Discuss whether the conclusion with regard to the timing for isolation of HPCI still remains valid at EPU conditions.

6. Enclosure 13 of the June 28, 2004, submittal states,

   > Because the BFN construction permits were issued prior to the May 21, 1971, effective date of the GDC, compliance to these criteria [i.e., the acceptance criteria contained in RS-001] is not required as part of the BFN Units 2 and 3 licensing basis.

   Correspondingly, the submittal contains a modified version of Section 2.5.1.4, Fire Protection, of Insert 5 for “Section 3.2 - BWR Template Safety Evaluation” from RS-001. However, Section 1.3, Basis of the Fire Protection Plan, of Volume 1 of the BFN FPR, states the following.

   > This Fire Protection Plan has been developed for BFN to satisfy the requirements of General Design Criterion (GDC) 3 of Appendix A to 10 CFR 50. . . . On November 19, 1980, the Nuclear Regulatory Commission (NRC) published its final 10 CFR 50.48, ‘Fire Protection,’ which established fire protection requirements for operating nuclear power plants. This regulation, which imposed the requirement to have a fire protection plan to satisfy GDC 3, became effective on February 17, 1981. This regulation is applicable to BFN.

   Furthermore, Section 6.7.1 presents an analysis based on the BFN FPR, which acknowledges GDC 3 as the basis for the current Fire Protection Program. Address the discrepancy between the submitted information and the FPR.

7. Some plants credit aspects of their Fire Protection System for other than fire protection activities (e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems). Identify the specific situations and discuss to what extent, if any, the EPU affects these “non-fire-protection” aspects of the plant Fire Protection System.

SPSB-A

1. The second paragraph of Section 10.5 of Enclosure 4 of the June 28, 2004, submittal indicates that all associated plant modifications were systematically reviewed to identify their effect on the elements of the probabilistic risk assessment (PRA) model. Provide the details of these systematic reviews, including the effect of each modification on the PRA model.

2. Provide the following information related to the treatment of a loss of offsite power (LOOP) in the PRA model:

   a. Describe how the frequencies of LOOP events were determined.

   b. Describe how the recovery of offsite power is modeled in the PRA (e.g., use of specific representative times, probabilistic convolutions).

   c. Describe how the probabilities of offsite power recovery events were determined.

   d. Describe how the probability of consequential LOOP was determined.

   e. Provide the contribution to the total core damage frequency (CDF) from consequential LOOP events.

3. Section 10.5.1 of Enclosure 4 of the June 28, 2004, submittal indicates that the Unit 1 PRA uses more detailed initiating event categories as compared to the Unit 2 and Unit 3 PRAs in order to facilitate the tracing of success criteria in the PRA model. Explain why it was necessary to adopt this approach for the Unit 1 PRA and describe (in terms of the PRA modeling) how the approach actually facilitates the tracing of success criteria. Explain why it was not necessary to use more detailed initiating event categories in the Unit 2 and 3 PRA models.

4. Identify the specific sources of the data used in the Unit 1 PRA (including initiating event frequencies, basic event failure probabilities, split fractions, and common cause data). If any data based on the operating experience of Unit 1 has been used, justify its applicability to the post-EPU plant, considering that Unit 1 has been shut down for almost two decades. If any data based on the operating experience of Units 2 and 3 has been used, justify its applicability to Unit 1.

5. The following questions/requests relate to the internal flooding initiating event frequencies:

   a. For “emergency equipment cooling water (EECW) flood in reactor building – shutdown units,” the Unit 1 frequency is given as 1.2E-3. For Unit 2, this frequency is given as 1.2E-5, and for Unit 3, as 1.2E-2. Provide an explanation and bases for these widely different estimates.

b. For the remaining flooding initiators (EECW flood in reactor building – operating unit, flood from the condensate storage tank, flood from the torus, large turbine building flood and small turbine building flood), the Unit 1 frequencies are higher than the corresponding Unit 2 and 3 frequencies. Explain and provide a basis for these differences.

6. Section 10.5 of Enclosure 4 of the June 28, 2004, submittal states that the Unit 1 PRA assumes that Units 2 and 3 are operational at EPU power levels. Provide the following information related to the treatment of multi-unit interactions in the Unit 1, 2, and 3 PRA models:

    a. Describe how various combinations of plant operating states (at-power, shutdown, transition) are addressed.

    b. Describe which initiating events impact more than one unit and describe how these are modeled.

    c. Identify the systems that are shared among units and describe how these shared systems are modeled in the PRA. Specifically address when credit is taken to recover failed key safety functions by using cross-connects among units.

7. Provide the detailed human reliability analysis (HRA) calculation sheets, (e.g., as generated by the Electric Power Research Institute (EPRI) HRA calculator) for all human interactions ("operator actions") that have a Fussell-Vesely importance measure greater than 0.005 or a risk-achievement worth greater than 2. Include a discussion of how performance shaping factors were modified for the Unit 1 human reliability analysis to account for new procedures, lack of familiarity with Unit 1 equipment, the potential for "wrong unit" errors, and other factors unique to starting up a plant that has not operated in almost two decades.

8. Provide a discussion of large early release frequency (LERF) from external events or a basis for concluding that any increases due to EPU are not significant.

9. The frequency-weighted fractional importance to core damage of operator action HORVD2, Manual depressurization of reactor pressure vessel using MSRVs, for the post-EPU plant is 55 percent for Unit 2 and 43 percent for Unit 3 CDF. For Unit 1, the corresponding operator action appears to be HPRVD1, Operator fails to initiate depressurization, which has a frequency-weighted fractional importance to core damage of 26.7 percent. Explain, in detail, why these apparently similar events have such different importance to core damage in light of the similarity of the PRA models. Also, describe the programmatic activities (e.g., training) intended to make this operator action reliable.

10. Section 10.5.3 of Enclosure 4 of the June 28, 2004, submittal states:

    > Recovery actions take credit for those actions performed by the on-shift personnel either in response to procedural direction or as

skill-of-the-craft to recover a failed function, system or component that is used in the performance of a response action in dominant sequences.

Does this include repair of failed equipment? If yes:

a. Provide a list of repair events credited in each PRA model, including the basis for the non-recovery probabilities used.

b. How have these repair human error probabilities been adjusted as the result of EPU?

c. Provide a sensitivity of CDF and LERF to repair activities, if credited, by removing all credit for repair of failed equipment.

11. As part of its EPU submittal, the licensee has proposed taking credit (Unit 1) or extending the existing credit (Units 2 and 3) for containment accident pressure to provide adequate net positive suction head (NPSH) to the ECCS pumps. Section 3.1 in Attachment 2 to Matrix 13 of Section 2.1 of RS-001, Revision 0 states that the licensee needs to address the risk impacts of the extended power uprate on functional and system-level success criteria. The staff observes that crediting containment accident pressure affects the PRA success criteria; therefore, the PRA should contain accident sequences involving ECCS pump cavitation due to inadequate containment pressure. Section 1.1 of Regulatory Guide (RG) 1.174 states that licensee-initiated licensing basis change requests that go beyond current staff positions may be evaluated by the staff using traditional engineering analyses as well as a risk-informed approach, and that a licensee may be requested to submit supplemental risk information if such information is not submitted by the licensee. It is necessary to consider risk insights, in addition to the results of traditional engineering analyses, while determining the regulatory acceptability of crediting containment accident pressure.

    Considering the above discussion, please provide an assessment of the credit for containment accident pressure against the five key principles of risk-informed decision-making stated in RG 1.174 and SRP Chapter 19. Specifically, demonstrate that the proposed containment accident pressure credit meets current regulations, is consistent with the defense-in-depth philosophy, maintains sufficient safety margins, results in an increase in core-damage frequency and risk that is small and consistent with the intent of the Commission's Safety Goal Policy Statement, and will be monitored using performance measurement strategies. With respect to the fourth key principle (small increase in risk), provide a quantitative risk assessment that demonstrates that the proposed containment accident pressure credit meets the numerical risk acceptance guidelines in Section 2.2.4 of RG 1.174. This quantitative risk assessment must include specific containment failure mechanisms (e.g., liner failures, penetration failures, primary containment isolation system failures) that cause a loss of containment pressure and subsequent loss of NPSH to the ECCS pumps.

12. Section 10.1.3 of Enclosure 4 of the June 28, 2004, submittal states that the mass release for reactor water cleanup breaks was calculated using the 30-psi reactor pressure increase, and that the safety-related equipment was evaluated for effects. Was the PRA flooding study updated to reflect this flooding rate? Was the impact of the flood on non-safety related equipment credited in the PRA determined and factored into the risk assessment?

13. The existing fire risk evaluations are based on the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology, which uses a quantitative screening criterion of 10^{-6} per year. This screening criterion appears too large because the core-damage frequency from internal events is of the same order of magnitude. As the fire risk evaluations for Units 2 and 3 have not been updated since the individual plant external event evaluation was performed, provide an updated FIVE analysis for Unit 1 that reflects the post-EPU plant configuration and uses an appropriate screening criterion.

14. Enclosure 7 of the June 28, 2004, submittal identifies planned modifications of the drywell building steel (building steel beams and connections), main steam supports, and torus attached piping (supports and snubbers) due to the EPU conditions. With respect to these planned modifications, address the following issues:

    a. Confirm that these planned modifications will not change the high confidence of low probability of failure values used in the seismic margins analysis.

    b. Describe the impact that the proposed modifications have on the probability distribution function of containment strength used in the LERF analysis.

15. Explain why LERF is less than CDF for interfacing system LOCAs.

16. TVA has previously requested a full-scope application of an alternative source term. As part of this request, it was proposed that the standby liquid control system be used to help control suppression pool pH during severe accidents. Has suppression pool pH control been credited in the LERF analysis? If so, provide the details.

17. Describe the operator actions considered in the estimation of LERF. How are the Severe Accident Management Guidelines accounted for in the LERF analysis?

18. Address the questions in the SRP, Chapter 19, Table III-1 concerning low power and shutdown PRA.

19. With respect to the technical adequacy of the Unit 1 PRA, the letter from T. E. Abney, TVA, to the U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant (BFN) - Unit 1 - Response to Request for Additional Information Related to Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerability," dated August 17, 2004, states:

    > Since the Unit 1 PSA was built from the Units 2 and 3 PSAs, which incorporate the resolution of the peer review comment, the Unit 1 PSA has

> incorporated the findings of the Units 2 and 3 PRAs Peer Review. Thus, the previously conducted Peer Review was effectively an administrative and technical Peer Review of the Unit 1 PSA. Similar models, processes, policies, approaches, reviews, and management oversight were utilized to develop the Unit 1 PSA.

RG 1.174, Section 2.2.3.3, states

> In the current context, technical acceptability will be understood as being determined by the adequacy of the actual modeling and the reasonableness of the assumptions and approximations.

In order to assess the “adequacy of the actual modeling” in the Unit 1 PRA, it is necessary to review the actual Unit 1 PRA model.

Provide an assessment of the PRA’s technical adequacy as discussed in RG 1.200. Note that it is acceptable to perform the assessment by making either (a) a direct assessment against the requirements of the ASME PRA Standard Addendum A (ASME SA-Ra-2003), or (b) a self-assessment using the guidance issued on August 16, 2002, by the Nuclear Energy Institute (NEI) that supplements NEI 00-02.

20. Provide a list of the significant basic events contained in the PRA logic model (including both the basic event name, the basic event description, the Fussell-Vesely importance measure and the Risk Achievement Worth) for the post-EPU plant configuration. Note that term “significant basic event” is defined in RG 1.200, Appendix A, Table A-1, Index Number 2.2.

21. Identify the key sources of uncertainty and the key assumptions in the PRA. Note that the terms “key source of uncertainty” and “key assumption” are defined in RG 1.200, Appendix A, Table A-1, Index Number 2.2.

IROB-B

1. Describe how the proposed EPU will change the plant emergency and abnormal operating procedures.

2. Describe any new operator actions needed as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU.

3. Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g., normal, marginal and out-of-tolerance ranges) on meters will change? What setpoints will change? How will the operators know of the change? Describe any controls, displays, or alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators will be tested to determine they could use the instruments reliably.

4. Describe any changes to the safety parameter display system resulting from the proposed EPU. How will the operators be informed of the changes?

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