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, UNITS 2 AND 3 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE (TS-431) (TAC NOS. MC3743 AND MC3744)

Dear Mr. Singer:

By letter to the U. S. Nuclear Regulatory Commission (NRC) dated June 25, 2004, Tennessee Valley Authority (the licensee) submitted an amendment request for Browns Ferry Nuclear Plant, Units 2 and 3. The proposed amendments would change the Units 2 and 3, operating licenses to increase the maximum authorized power level from 3458 to 3952 megawatts thermal. This change represents an increase of approximately 15 percent above the current maximum authorized power level. The proposed amendments would also change the Units 2 and 3 licensing bases to revise the credit for overpressure from 3 pounds for short-term and 1 pound for long-term, to 3 pounds for the duration of a loss-of-coolant accident, and revise the maximum ultimate heat sink temperature.

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. If you have any questions, please contact me at (301) 415-2315.

Sincerely,

/RA/
Eva A. Brown, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

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

cc w/enclosures: See next page

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Package No.: ML053570284 Enclosure 2: ML053570265
ADAMS Accession No. ML053560177 NRR-106

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SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3 — REQUEST FOR ADDITIONAL INFORMATION FOR EXTENDED POWER UPRATE (TS-431) (TAC NOS. MC3743 AND MC3744)

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REQUEST FOR ADDITIONAL INFORMATION

EXTENDED POWER UPRATE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

CVIB (EMCB-A)

1. a) According to the license renewal submittal for Browns Ferry Nuclear Plant (BFN), it was stated that the effective full-power years (EFPYs) of operation at the end of license (EOL) extended term is 52 EFPYs for BFN, Units 2 and 3. 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 BFN Units 2 and 3.

   (b) Provide the following information related to the past operating history: (a) megawatt thermal power (MWt), (b) calendar years of operation, and (c) capacity factor. In addition, the staff requests that the licensee 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). 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 effect 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
Plate - Heat # A0981-1	40	1.37×10^{18}	7.8×10^{17}	1.2×10^{18}	117.3%
Weld	40	1.37×10^{18}	7.8×10^{17}	1.2×10^{18}	117.3%

Enclosure 1

According to the license renewal submittal for BFN the extended term is for 52 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. Provide information regarding the effect of the EPU on the BFN Units 2 and 3 ISP capsule withdrawal schedules, 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.

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 feedwater (FW) nozzles are inspected in accordance with the requirements of the American Society of Mechanical Engineers (ASME) 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 effect those analyses.

EQVA (IPSB-A)

1. Page E-3 of the April 25, 2005, submittal states that, "BFN UFSAR [Updated Final Safety Analysis Report] 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 (RG) 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 RG 1.68. If the EPU test plan is not in conformance with RG 1.68, provide a justification.

2. Table 1 of the April 25, 2005, submittal describes the original startup test procedure (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.

3. Table 1 of the April 25, 2005, submittal describes the original STP 24, Bypass Valves, and states that, "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 states 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.

4. 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 states 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 submittal dated June 25, 2004 (Power Uprate Safety Analysis Report (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.

5. Provide a table that describes the BFN Units 2 and 3 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).

6. 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 and other factors which could affect the dynamic response of the plant.

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 25, 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 25, 2004.

4. Page 7-2 of Enclosure 2 of the submittal dated June 25, 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 25, 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 Enclosure 4 of the June 25, 2004, submittal, NEDC-33047P, DRF 0000-0011-1328, Revision 2, Browns Ferry Units 2 and 3 Safety Analysis Report for Extended Power Uprate, or 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 Units 2 and 3 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 Standard Review Plan (SRP) Section 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 °F to 95 °F. 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) Part 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.

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 Engineered Safety Feature (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 (LBLOCA), 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 emergency core cooling system (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 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 current licensed thermal power 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 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 Units 2 and 3 are 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 accident pressure.

28. The Hydraulic Institute recommends margin above the required NPSH to suppress cavitation. Discuss the need for the BFN pumps crediting overpressure and how this margin is accounted for in your NPSH calculations. This response 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 safety evaluation 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 Units 2 and 3 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 (psig)) 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 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. On page E1-13 of Enclosure 1 of the June 25, 2004, submittal states that “[n]o increase in allowable peak bundle power is requested for EPU.” Specify the peak bundle power.

2. The EPU analyses contained in Enclosure 4 of the June 25, 2004, submittal, NEDC-33047, DRF 0000-0011-1328, Revision 2, Browns Ferry Units 2 and 3 Safety Analysis Report for Extended Power Uprate, or the PUSAR, are based on Unit 2 operating conditions. The operating conditions and plant features of Units 2 and 3 may not be identical. 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 3. Also, identify all differences in plant design and operating conditions between the units.

3. In Section 2.3.1 of the PUSAR, the thermal hydraulic instability exclusion regions are not shown in Figure-2-1. Provide a figure showing the instability exclusion regions.

4. Section 1.3.2 of the PUSAR, states that there will be no increase in the maximum nominal dome pressure of 1050 pounds per square inch atmosphere (psia) for uprated conditions. Section 2.5.2 of the FUSAR indicates [______________________________ ______________________________]

5. In the NRC Safety Evaluation for Licensing Topical Report, NEDC-32523P-A, Generic Evaluations of General Electric Boiling Water Reactor Power Uprate, or ELTR2, page 12 it is stated: “In order to reduce the possibility of turbine overspeed trips, plant-specific submittals must address the modifications described in GE SIL [Service Information Letter] No. 480 and GE SIL No. 377 (or equivalent modifications).” Confirm that modification recommended by SIL No. 377 are implemented in Units 2 and 3.

6. Section 2.1 of the PUSAR states, that “[t]he additional energy requirements for EPU are met by 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.” Describe how EPU is achieved for BFN Units 2 and 3. Specify the fuel enrichment and the reload batch size before and after EPU.

7. Section 4.3 of the PUSAR indicates that the [______________________________ __________.] However Table 4-5 indicates that the peak clad temperature (PCT) for GE-13 fuel decrease by 30 degrees Fahrenheit (^{o}F) and for GE-14 fuel, the PCT increases by 70 ^{o}F. Describe the reasons behind such a difference and why BFN Units 2 and 3 are an exception in this regard.

8. Section 4.3 page 4-10 states that “[t]he EPU effect on PCT for small recirculation line breaks is larger than the EPU effect on PCT for large line breaks.” Clarify what TVA considers a “small recirculation line break.”

9. Section 6.5 of the PUSAR states that, “there is a corresponding increase in the maximum discharge pressure and a decrease in the operating pressure margin for the pump discharge relief valves.” The peak calculated peak vessel bottom pressure is 1484 psig as shown in Table 9-4. However, the pump discharge pressure value used

for the surveillance test is only 1325 psig. Discuss the reason for specifying 1325 psig when the calculated peak pressure is indicated as 1484 psig.

10. Provide the standby liquid control reactor vessel set point. Discuss the value of the margin after the EPU and provide the pressure at which the relief valve is expected to reclose.

11. Section 9.3.1, Table 9-4 of the FUSAR shows that the peak calculated suppression pool temperature is decreased from 214.6 ^{o}F to 214.1 ^{o}F. Explain why the suppression pool temperature is decreasing due to EPU.

12. Discuss whether the operator actions specified in emergency operating procedures are consistent with the BWR Owners' Group Emergency Procedure and Severe Accident Guidelines, Revision 2 insofar as they apply to the operator actions for an ATWS. Specify the time delay used in the ATWS analysis for starting the standby liquid control pump(s).

13. The Executive Summary of the February 23, 2005 submittal states that, "[a]ll analyses are performed for a reference ATRIUM-10 equilibrium core." Define the "equilibrium core," and explain in detail how this analysis meet all the EPU licensing and regulatory requirements for the first Unit 2 core with partial load of GE fuel and Unit 3 with all ATRIUM-10 fuel.

14. Section 1.2.1 of Enclosure 5, EMF-2982(P), Browns Ferry Units 2 and 3 Safety Analysis Report for Extended Power Uprate ATRIUM™-10 Fuel Supplement, Revision 0, or the FUSAR, states:

> For most of the EPU analyses, the 2 percent power factor discussed in Regulatory Guide 1.49 is accounted for in the analysis methods. Three exceptions are ASME over pressurization, loss of feedwater (LOFW) flow, and LOCA analyses.

However page 3-1, of Section 3.2 states that "[t]he events were analyzed at 102 percent of EPU rated thermal power. . . ." for the overpressure protection analysis. Discuss whether an exception was taken for the overpressure analysis.

Explain in detail why exceptions are taken from the RG 1.49 position of 2-percent power factor for the LOFW and LOCA analyses.

15. Table 1.3 of Section 1.2 of the June 6, 2005, submittal lists all the nuclear steam system codes used for the EPU request. Section 1.2.2 indicates that the Unit 1 application of these codes complies with the limitations, restrictions, and conditions specified in the applicable NRC safety evaluation (SE) report that approved each code, with exceptions as noted in Table 1-3.

Provide a review of the fuel vendor's analytical methods and code systems (neutronic, LOCA, transient and accidents, etc.) used to perform the safety analyses supporting the Units 2 and 3 application and provide the following information to confirm that:

a) 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) 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; and

c) 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 Units 2 and 3 during steady state, transient, or accident conditions remain valid and applicable for the EPU conditions.

16. Section 2.1 of the FUSAR states that “[t]he NRC-approved exposure limits are not exceeded in the ATRIUM-10 equilibrium core design used in the EPU evaluations.” Specify the NRC-approved exposure limits for ATRIUM-10 fuel.

17. Section 2.1.1 of the FUSAR states that, “[s]ince there is no change in the average bundle power with ATRIUM-10 fuel, there is no change to the thermal margin monitoring threshold.” However, the fuel design is changed to achieve EPU and the average power is increased from 4.53 MW/bundle to 5.17 MW/bundle for EPU. Explain the discrepancy stated in the conclusion that a change in fuel design has no impact on average bundle power.

18. Section 3.8 of FUSAR states that, “[l]oss of feedwater flow analyses were performed without reactor isolation since this scenario presents a greater challenge to the system’s ability to maintain water level above the level 1 setpoint.” This implies that Framatome takes credit for systems other than reactor core isolation coolant for reactor core makeup. Discuss the assumptions and systems used in the analysis.

19. For Figures 3-5, MSIV closure with flux scram, and 3-12, Turbine Trip with bypass failure, of the FUSAR, specify the units for the Y-axis for reactivity.

20. In Table 4.1 of the FUSAR, the PCT for ATRIUM-10 fuel at EPU is shown as 1990 ^{o}F. However, in a letter dated April 8, 2005, providing a report of emergency core cooling system evaluation model changes, TVA informed the NRC staff that the calculated PCT for EPU operation is 2007 ^{o}F. Discuss the PCT for Units 2 and 3 EPU operation.

21. In TVA’s response dated June 6, 2005, to NRC Request 11, Table 10 was included listing the EPU Design/Licensing Bases Changes. Change in limiting PCT is identified as one of the licensing bases changes and reference is made to PUSAR Section 4.3. It is not clear from the PUSAR, the limiting event prior to EPU and the limiting event after the EPU. Explain in detail the changes in the limiting event for determining the PCT. Include the impact, if any from the use of ATRIUM-10 fuel.

22. The BFN Units 2 and 3 submittal is the first application of Framatome methods for the transient and accident analysis for EPU. Consistent with Appendix E of Licensing Topical Report, NEDC-32424P-A, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, or ELTR1 provide a justification for the use of

Framatome transient and accident analyses methods for EPU application. List all transients and accidents analyzed in support of EPU for Sections 2.8.5.1 to 2.8.5.6 of Matrix-8 in RS-01. In addition, explain how the limiting transients were selected and discuss in detail how each transient/accident in Sections 2.8.5 to 2.8.5.6 is dispositioned. Specifically:

a) Confirm that the scenario and sequence of event described in UFSAR is still valid for EPU;

b) Identify the supporting analysis for the event, evaluation model used for the analysis, and identify the topical report which describes the event;

c) Specify whether the disposition is based on a generic analysis or the current analysis in the UFSAR, or plant-specific equilibrium analysis or the reload analysis, and

d) Describe how the acceptance criteria is met.

23. Matrix 8 of Section 2.1 of RS-001 address new fuel and spent fuel storage. Draft GDC-40 and GDC-66, these GDC require that protection for engineered safety features 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.

24. 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.

25. 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.

26. Demonstrate quantitatively and qualitatively, that the Lattice/Depletion code systems’ is capable of predicting the power peaking distribution at the upper part of the high powered bundles for operation under high void fractions. For example, show that the requirements in Chapter 4 of EMF-2158(P)-A, “Siemens Power Corporation

Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," can still be met for EPU core designs.

27. Demonstrate quantitatively and qualitatively, that the current uncertainties and biases established in the benchmarkings and presented in Table 9.8 and 9.9 of EMF-2158(P)-A remain valid for the neutronic and thermal-hydraulic conditions predicted for the EPU operation. Specifically, demonstrate the uncertainties and biases used in your reactivity coefficients (e.g., void coefficient) are applicable or remain valid for the neutronic and thermal-hydraulic conditions expected for EPU operation.

28. Demonstrate quantitatively and qualitatively, that the fuel isotopic validations and testing performed in EMF-2158(P)-A remain applicable for prolonged operation under high void conditions for the fuel lattice designs that would be used for the expected EPU core designs.

29. Demonstrate qualitatively and quantitatively that the Framatome-ANP neutronic methodology experience base is applicable to EPU conditions at BFN.

30. Demonstrate that the Framatome-ANP neutronic methodology prediction capability for current fuel designs operated under the current operating strategies and core conditions. Prediction comparison should be made to gamma scans and traversing incore probe (TIP) core follow data. This demonstration applies to any recent fuel, such as the ATRIUM-9 and ATRIUM-10, in particular for first cycle and second cycle fuel. (Refer to Framatome Handout for August 4, 2005, Meeting; ADAMS Accession No. ML052370230.)

31. The first three bullets of slide No. 63 of the CASMO-4/MICROBURN-B2 Methodology, of the August 4, 2005, presentation, alluded to the average value of the Correlation Coefficient (CC) for the Quad Cities cycles 2 and 4. These values were determined for the 8X8 bundle design. Provide quantitative and qualitative technical justification for the use of 8X8 CC to 10X10 bundle design, specifically, demonstrate that the correlation coefficients are independent of fuel bundle design.

32. The radial peaking factors (RPF) uncertainty presented on slide No. 35 of the Safety Analysis, of the August 4 presentation, point to the conclusion that the safety limit minimum critical power ratio (SLMCPR) is not very sensitive to small increases in RPF uncertainty. Provide quantitative and qualitative technical justification in support of this conclusion. Specifically:

    a) Provide statistical information (number of histories, number of bundle or rods, etc.) supporting the 0.0055 value; and

    b) Address whether the SLMCPR value of 1.0855 is rounded up to 1.09 as the final TS Value.

33. Provide qualitative and quantitative description of the gamma scanning process. Specifically address whether the data is obtained via gamma scanning, transformed mathematically and or chemically into providing isotopic burnup/depletion information.

34. Describe qualitatively the cross-section reconstruction process incorporated in CASMO-4 and MICROBURN-B2. The response should reflect the information provided in the slides (1 - 35) of the August 4 presentations, including high void fraction effects and accuracy. Provide flow chart(s), road map(s) and any other means to demonstrate the process, starting from the gathered raw void fraction data, how that data is used by CASMO-4 to generate the required cross-sections. In addition, briefly describe the development of the void fraction correlation and associated uncertainties.

35. Provide qualitative description of the void data base and the associated correlation. Specifically, describe the uncertainty associated with the data gathering, identifying the uncertainties currently applied to the void fraction correlation and justify its applicability for EPU conditions.

36. Demonstrate that the database used to establish the two phase pressure drop for the fuel designs used for the EPU include predicted EPU channel and fuel assembly design conditions. Specifically, demonstrate that current normal power pertinent two-phase flow/pressure drop ranges are still applicable to EPU anticipated ranges of operations.

37. Describe that the methods used in the licensing codes to model the bypass water (e.g., core simulator, steady state and transient codes, LOCA codes).

38. State the bypass voiding criteria (if any) or specification that applies to the TIP and the LPRM.

39. Demonstrate that the capability of the licensing code systems, including the core simulator, to determine the potential for bypass voiding.

40. Provide an evaluation and discussion of the lattice/depletion code (CASMO-4) capability to generate the cross-section with voiding in the in-channel water rods and bypass.

41. Evaluate EPU core neutronic and thermal-hydraulic conditions and state if for EPU core designs and operating conditions, if bypass voiding can occur during steady state or transient events. Consider operation at all limiting statepoints in the maximum extended load line limit analysis (MELLLA) domain.

42. In August 30, 2004, General Electric Nuclear Energy (GENE) issued a 10 CFR Part 21 report (ADAMS ML042720293), stating that using limiting control rod blade patterns developed for less than rated flow at rated power conditions could sometimes yield more limiting bundle-by-bundle MCPR distributions and/or more limiting bundle axial power shapes than using limiting control rod patterns developed for rated flow/rated power in the SLMCPR calculation. The affected plants submitted amendment requests increasing their SLMCPR value. The staff understand that Framatome did not issue a Part 21 reporting on the SLMCPR methodology that addresses the calculation of the SLMCPR at minimum core flow and off-rated conditions similar to GENE’s Part 21 report .

    Reference the applicable sections of the ANF-524P-A SLMCPR methodology that specify the requirement to calculate the SLMCPR at the worst case conditions for minimum core flow conditions for rated power. Demonstrate that the SLMCPR is calculated at different statepoints of the licensed operating domain, including the minimum core flow statepoint and that the calculation is performed for different exposure points.

43. Discuss or reference the applicable sections/chapters of ANF-524P-A that addresses what rod patterns are assumed in performing the nonrated flow SLMCPR calculations. State how it is established that the rod patterns assumed in the SLMCPR calculations for rated power, flow, and minimum core flow conditions, would reasonably bound the planned rod pattern that Units 2 and 3 would operate under EPU conditions.

44. For implementation of Average Power Range Monitor, Rod Block Monitor, Technical Specifications Improvement Program (ARTS)/MELLLA using Framatome methods, show that Units 2 and 3 can operate at all statepoints, including the minimum core flow statepoint, without violating their SLMCPR in the event of an abnormal operating occurrence. The minimum core flow statepoint SLMCPR calculations should demonstrate that Units 2 and 3 can operate at the minimum flow statepoint with some margin.

45. Regarding SAFLIM2 calculations, slides 27 - 31, please provide qualitative description of this process, including the termination of the process if the EPSBTR Criterion is satisfied.

46. Describe the process for establishing the design limit curves for the linear heat generation rate (LHGR) and the maximum average planar LHGR (MAPLHGR). In addition, discuss what impact (if any) high void fractions have on this process.

47. The Siemens Power Corporation- B (SPCB) CHF correlation was approved by the NRC staff to be applicable over specified ranges of mass flow, pressure, and inlet subcooling. Discuss whether the utilization of the SPCB Correlation to EPU take the Correlation beyond its approved applicable ranges?

48. Table 1-3 of the FUSAR, lists computer codes used for EPU transient analyses. Please clarify which code was used for the over-pressure protection analysis, and was the code approved by the NRC specifically for this transient.

49. Section 2.3 of the FUSAR states that, "[c]ycle length and hot excess reactivity are maintained by appropriate selection of initial enrichment, fresh batch size, and burnable neutron absorber design. Sufficient design flexibility exists with the ATRIUM-10 fuel to accommodate operation at EPU conditions while maintaining adequate power distribution control." In order to obtain the additional 14-percent power (roughly 500 MWt) at EPU condition, discuss whether the maximum enrichment level of fuel assembles is increased compared to the pre-EPU enrichment level. If so, by how much, and address the impact this has in fuel storage facility.

50. In Section 4.3 of the FUSAR, the limiting LBLOCA was discussed for EPU with ATRIUM-0 fuel for Units 2 and 3. Provide the following additional information regarding the Units 2 and 3 LBLOCA analysis:

a) Describe the Units 2 and 3 limiting single failure LBLOCA event for the current licensing basis. Typically, the events are same for EPU and pre-EPU conditions. But if the events are different for Units 2 and 3, provide an explanation.

b) The peak cladding temperature (PCT) changes have been [ ] Discuss whether this is also true for Units 2 and 3. If not, then indicate by how much the PCT increased, and why Units 2 and 3 is an exception in this regard.

51. Discuss whether there is any modification in the fuel design of ATRIUM-10 for EPU operation. If so, please describe that in detail, including any changes in the maximum enrichment level.

52. 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.

53. Browns Ferry Units 2 and 3 currently operate under Option III solution. Provide a clarification for the following areas:

a) Describe the expected effects of EPU operation on Option III.

b) Describe any alternative method to provide detection and suppression of any mode of instability other than through the current (Oscillation Power Range Monitor (OPRM) scram (e.g., interim corrective actions).

c) Provide a summary of the Browns Ferry Technical Specifications affected by the Option III implementation and future EPU operation.

d) Provide the approved methodologies used to calculate the OPRM setpoint for the current operation and future Browns Ferry EPU operation.

54. Provide the technical basis that supports the position that the hot channel oscillation magnitude portion of the Option III calculation is not affected by EPU and does not need to be recalculated because the OPRM hardware does not change.

55. Provide the Browns Ferry Delta CPR/Initial CPR Versus Oscillation Magnitude, or DIVOM, curve used for the last three cycles and the next EPU cycle. The purpose is to evaluate the impact of EPU on the safety margins in case of instabilities.

56. 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 which are 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.

57. In Section 9.3.1.1 of the FUSAR, Framatome claims that "[o]peration with ATRIUM-10 fuel results in small changes to parameters important to determining stability."

    Assuming the core power distribution is changed as a result of the EPU, provide the following information:

    a) Provide the results of the analysis and compare the consequences of reactor instability during ATWS before and after the EPU for both mitigated and unmitigated cases;

    b) Quantify the potential core damage percentage before and after the EPU;

    c) Provide the NRC staff approved evaluation model used for the thermal hydraulic stability analyses during ATWS;

    d) Identify any deviation from the methodology stated in the approved BWR Owners Group topical report, NEDO-32047-A, ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability, and NEDO-32164, Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS, and provide justification for its applicability to this analysis if there is a deviation.

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