ATTACHMENT 2

Browns Ferry Nuclear Plant (BFN)
Unit 1

Response to NRC Request for Supplemental Information Regarding
Technical Specification Change TS-467 - Utilization of AREVA Fuel and Associated
Analysis Methodologies
(Non-Proprietary)

Attached is the non-proprietary version of the Response to NRC Request for
Supplemental Information Regarding Technical Specification Change TS-467 -
Utilization of AREVA Fuel and Associated Analysis Methodologies.

Question 1.

Licensing topical report (LTR) ANP-2638P "Applicability of AREVA NP BWR [boiling water reactor] Methods to Extended Power Uprate Conditions," states that loss-of-coolant accident (LOCA) results are only weakly dependent on core average power. However, for the small break LOCA (SBLOCA) the analysis results are highly sensitive to the core average power level.

Since depressurization occurs through the automatic depressurization system (ADS) for SBLOCAs, the timing when low pressure injection systems reach rated flow is extended when the core steam generation rate is higher – as would be the case for EPU conditions. Based on the plant-specific power uprate and the ADS capacity, the limiting break for an EPU plant may be a SBLOCA. This was shown for Browns Ferry Unit 1 in the power uprate safety analysis report.

The EXEM BWR-2000 LOCA analysis methodology is described by LTR EMF-2361(P)(A), "EXEM BWR-2000 ECCS [emergency core cooling system] Evaluation Model." This LTR states:

> SBLOCA PCTs [peak cladding temperatures] are bound when the conservatism included in the EM methodology is applied. This result is acceptable because small break events are not limiting in BWRs and the test evaluated simulated an extremely small break in which core uncovery and the resulting heat-up is minor such that the conservatism (Appendix K coefficients) are not allowed to raise fuel temperature to values of concern.

When the LTR language is considered in the context of the Unit 1 LOCA analyses at EPU conditions, EXEM BWR-2000 does not appear to be applicable. First, for Unit 1 at EPU conditions, the limiting break is a small break. This appears contrary to the basis for the staff approval of EXEM BWR-2000.

Second, at EPU conditions the core heat-up is not rapidly terminated because blowdown times are prolonged for SBLOCA. Therefore, the core uncovery persists for a longer duration and the Appendix K assumptions (e.g., the 20 percent increase in decay heat) will contribute to significant heatup and high fuel temperatures. This appears to conflict with the disposition of the SBLOCA qualification results in the LTR.

Therefore, it does not appear that EXEM BWR-2000 is applicable to analyze the limiting LOCA event for Unit 1. Provide the SBLOCA analyses for Unit 1 using acceptable methods.

Response 1.

For a Pressurized Water Reactor, SBLOCA results may be highly sensitive to core average power. However, for a BWR, SBLOCA results are not nearly as sensitive to initial core power primarily due to the mitigating effects of the

Automatic Depressurization System (ADS). The ADS essentially turns a SBLOCA into a large break steam line LOCA. If the ADS is significantly degraded (or not available), the sensitivity of a SBLOCA to initial power level would be more significant.

As indicated in the NRC request, the rate of depressurization and timing when low pressure injection systems reach rated flow is affected by initial power level. The core steam generation rate (i.e., the steam generated by decay heat) is higher at Extended Power Uprate (EPU) than at pre-EPU conditions. The higher steam generation rate will slow down the depressurization rate slightly; however, the depressurization rate is determined by the net inventory loss rate, that is, the difference between the steam generation rate and the sum of inventory loss out the ADS valves and the break. Because ADS flow rate is much larger than the steam generation rate, the change in depressurization rate is much less than the change in steam generation rate (power level) on a relative basis. For the 0.05 ft^2 recirculation line break (discharge side, top peaked) at EPU conditions, the steam generation rate is 13.8% of the combined ADS and break flow at the time of ADS initiation. A 15% change in the steam generation rate would change the net inventory loss by only 2.3%. This change would not significantly impact the depressurization rate or Low Pressure Core Spray (LPCS) initiation times following ADS actuation.

The impact of initial power level (steam generation rate) on depressurization rate and LPCS initiation times was further assessed by repeating the 0.05 ft^2 recirculation line break at pre-EPU conditions. Relative to pre-EPU, increasing to EPU initial core power delayed the start of LPCS flow after ADS initiation by 0.8 seconds (0.5% of the blowdown time) and delayed reaching rated LPCS flow by 10.7 seconds (3.4% of the blowdown time). The PCT increase for EPU conditions was not significant (25°F). Note, the higher EPU power level also results in an earlier initiation of ADS and LPCS due to a reduced initial liquid inventory and a higher steam generation rate.

The above sensitivities are dependent on the ADS characteristics assumed in the analyses. Degraded ADS performance may increase the sensitivity of a SBLOCA to initial core power level. The EXEM BWR-2000 evaluation model adequately models the important phenomena for SBLOCA analyses and would correctly reflect degraded PCT performance if the ADS system performance is degraded from that assumed in the Browns Ferry Nuclear Plant (BFN) LOCA analyses. Further discussion of SBLOCA analysis results with degraded ADS performance is provided in the response to NRC Question 4.

The statement referred to by the NRC question from the EXEM BWR-2000 LTR (EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model") was intended to indicate that the SBLOCA test resulted in relatively low peak cladding temperatures and therefore it was not surprising that a large amount of conservatism was not predicted in the analysis without the Appendix K model conservatisms included. The discussion in the LTR goes on to demonstrate that with Appendix K model conservatisms included in the analyses of the test, the AREVA methodology produced conservative results

as compared to test data. The statement referred to from the LTR was not intended to imply any restriction on the applicability of the methodology for SBLOCA analyses.

The AREVA LTR clearly indicates that the EXEM BWR-2000 methodology is conservative and applicable for SBLOCA analyses. The NRC specifically approved the method to be applicable and sufficiently conservative for analysis of SBLOCA events, as quoted in Section 4 of the Safety Evaluation for the EMF-2361:

> "*The test results for small breaks show low temperatures, and the EXEM BWR-2000 model using evaluation model options bounds the temperature data. Furthermore, the EXEM BWR-2000 model adequately predicts the important LOCA phenomena...*
>
> *The staff, therefore, concludes that the proposed EXEM BWR-2000 ECCS EM, as documented in References 1, 2, 4, and 5, is acceptable for referencing in BWR LOCA analyses, with the limitation that application of the revised evaluation model will be limited to jet pump plant applications..*"

The EXEM BWR-2000 methodology conservatively predicts the important phenomena that occur during a SBLOCA. The EXEM BWR-2000 methodology has been used for complete LOCA break spectrum analyses submitted and approved by the NRC for two other US BWRs operating at EPU conditions. The LOCA analyses supporting the BFN Unit 1 submittal were performed consistent with the NRC approval of the EXEM BWR-2000 methodology. Therefore, the BFN SBLOCA analyses provided to the NRC were performed using acceptable methods.

Question 2.

Provide the LOCA results for hydrogen generation/core wide oxidation.

Response 2.

The AREVA methodology for core wide metal water reaction (CMWR) analysis calculates CMWR as a function of planar power, axial power shape, and radial power distribution. All assemblies in the core are considered which includes all axial planes. CMWR is calculated as a function of exposure.

The CMWR analysis for BFN was based on a limiting metal water reaction (MWR) LOCA case with a Planar Average MWR (at the peak PCT plane) of 0.4%. The results gave a limiting CMWR and hydrogen generation of 0.05% at a cycle exposure of 18994.7 MWd/MTU.

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After the CMWR analysis was completed, the BFN break spectrum was reanalyzed in 2004 primarily because of a change in the ADS delay time. In the revised break spectrum (documented in Revision 1 of EMF-2950(P)), the limiting MWR LOCA case was more severe due to the additional ADS delay. The planar average MWR was 0.8%. The CMWR analysis results were reviewed to determine if a reanalysis was required with the revised break spectrum results. A reanalysis was not needed because the CMWR results did not challenge the 10 CFR 50.46 acceptance criteria of < 1% total hydrogen generation. [

] Since a new analysis was not performed, the CMWR results were reported as "< 1% hydrogen generation".

Figures 2.1 and 2.2 provide additional information on planar average MWR. The figures are based on the analysis with a PCT planar average MWR of 0.8%. [

] Figure 2.2 shows MWR vs. time at the PCT plane for the limiting LOCA case.

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Figure 2.1 Metal Water Reaction vs. Elevation Above the Bottom of the Active Fuel
Browns Ferry Nuclear Plant Limiting LOCA

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Figure 2.2 Metal Water Reaction vs. Time at the PCT Axial Location
Browns Ferry Nuclear Plant Limiting LOCA

Question 3.

The statements regarding the transition core effects on the LOCA analyses (EMF-2950(P), "Browns Ferry Units 1, 2, and 3 Extended Power Update LOCA Break Spectrum Analysis," Section 2), would inherently impose similar performance conclusions on the legacy fuel. By this logic, the current licensing basis analysis should demonstrate similar performance to the fuel transition analysis. Provide the report describing the previous licensing basis (in this case, the previous licensing basis is at EPU conditions) LOCA analysis and supplement the analysis by accounting for model differences that cause the results in break spectrum, location, geometry, and results to differ.

Response 3.

There are small differences in core volume and total core fluid energy for a full core of ATRIUM-10 fuel compared to a transition core with both ATRIUM-10 and legacy fuel. Since 95% of the reactor system volume is outside the active core region these differences have an insignificant impact on the overall system response during both large and small break LOCAs. The EMF-2950(P) statements (Attachment 14, Section 2, page 2-1, of Technical Specification Change TS-467) are made within the context of the approved vendor methodology. EMF-2950(P) did not intend to imply different methodologies would inherently demonstrate similar performance. Applying different methodologies to the same problem would not necessarily be expected to give identical answers.

AREVA has performed sensitivity studies that confirm the heat transfer behavior of the highest power assembly in the core (hot channel) during a LOCA is not dependent on the core configuration. [

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LOCA calculations performed with different methodologies may produce different results due to conservative assumptions, either required by Appendix K of 10 CFR 50.46 or other conservatisms that are part of the methodology. For example, the requirement to "lock out" boiling heat transfer if the calculated fuel rod surface temperature increases 300°F above saturation temperature will potentially have a much greater impact on LOCA analyses than small differences in core geometry. Unlike best estimate calculations, conservatisms in Appendix K methodologies make it difficult to determine the cause of differences in results between methods.

The requested General Electric (GE) LOCA analysis report has been included in this enclosure as Attachments 3 and 4. This report contains results calculated at Extended Power Uprate (EPU) conditions. Comparisons of the results in this report to the results in EMF-2950 Revision 2 (Attachment 14 of

Technical Specification Change TS-467) indicate a difference in the limiting break size. The AREVA analysis indicates that a break of 0.5 ft^2 is limiting, while the GE analysis indicates a 0.06 ft^2 limiting break size. The NRC had previously inquired as to the reasons for this difference as part of the BFN EPU review. The response to NRC Request for Additional Information SBWB-64 (submitted by letter from W. D. Crouch (Tennessee Valley Authority (TVA)) to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 – Technical Specifications (TS) TS-431 and TS-418 – Extended Power Uprate (EPU) – Response to Round 12 Request for Additional Information (RAI) – (TAC Nos. MC3812, MC3743, and MC3744)," dated February 26, 2007, ADAMS Accession Number ML070600339) provided a discussion from each of the two vendors as to the reasons why their method predicts the relative break size sensitivity that it does for the two break sizes mentioned. The discussion does not attempt to separate out the various components of each methodology in explaining the difference in break size trend, but rather looks at the integrated model predictions of key results such as time and duration of core uncovery, integrated inventory losses, timing of depressurization, and core reflood timing.

As noted in the response to Question 4, an issue has been identified relative to the effects of single failures of the ADS. As the Question 4 response indicates, the size of the limiting small break in the AREVA analysis is expected to shift downward from the current 0.5 ft^2 to approximately half that size. Accounting for a single failure that prevents the automatic initiation of the ADS brings the AREVA result significantly closer to the GE result in terms of the limiting break size. Therefore, the AREVA and Global Nuclear Fuels (GNF) models are producing results that are more consistent than was previously indicated.

Question 4.

Single failure analyses do not account for the ADS unavailability. Please provide the failure modes and effects analysis for the ADS with respect to the limiting postulated SBLOCA, and for the postulated high pressure coolant injection (HPCI) line break. Justify not analyzing the failure of the ADS system in toto, or even a single ADS valve.

Response 4.

Background

For BFN, Unit 1, background information regarding the ADS and HPCI System is provided as follows.

The ADS consists of 6 of the 13 Safety/Relief Valves (S/RVs). BFN Unit 1 Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.5.1, "ECCS-Operating," requires the ADS function of six S/RVs to be operable in Mode 1 and in Modes 2 and 3, except when reactor steam dome pressure is ≤ 150 psig. The ADS is designed to provide depressurization of the Reactor Coolant System during a small break Loss of Coolant Accident (SBLOCA) if

the HPCI System fails or is unable to maintain required water level in the reactor pressure vessel (RPV). ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure Emergency Core Cooling System (ECCS) subsystems (Core Spray (CS) and Low Pressure Coolant Injection (LPCI)), so that these subsystems can provide coolant inventory makeup. Therefore, the ADS provides the backup for the HPCI System. The ADS and HPCI System instrumentation and controls are designed such that no single failure can disable both the ADS and HPCI System functions.

Each of the six S/RVs used for automatic depressurization (i.e., ADS valves) is equipped with an air accumulator and associated inlet check valves. The accumulators provide the pneumatic power to actuate the valves.

Each of the six ADS valves is also provided with a direct current (DC) powered solenoid-operated pilot valve which controls the pneumatic pressure applied to a diaphragm actuator to actuate the ADS valve. The DC power for each of the six ADS solenoid valves is from the 250 VDC Unit Batteries. The normal power for the six ADS solenoid valves are evenly distributed across the three 250 VDC Unit Batteries (i.e., solenoid valves for ADS valves PCV-1-30 and PCV-1-22 are powered from Unit Battery # 1 (through 250 VDC RMOV Board 1A), solenoid valves for ADS valves PCV-1-34 and PCV-1-5 are powered from Unit Battery # 2 (through 250 VDC RMOV Board 1C), and solenoid valves for ADS valves PCV-1-31 and PCV-1-19 are powered from Unit Battery # 3 (through 250 VDC RMOV Board 1B)). In addition, the power for solenoid valves for two of the six ADS valves (i.e., PCV-1-30 and PCV-1-22) transfers to an alternate DC source in the event of the loss of the normal DC source.

The logic to initiate ADS includes two trip systems. BFN Unit 1 TS LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," requires the ADS trip systems A and B to be operable in Mode 1 and in Modes 2 and 3 with reactor steam dome pressure > 150 psig. The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High; High Drywell Pressure Bypass Timer; and Pump Discharge Pressure - High. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory). Either the Drywell Pressure - High or the Drywell Pressure Bypass Timer contacts and all remaining contacts in both logic strings must close and the ADS initiation timer must time out to initiate an ADS trip system. Either the A or B trip system will cause all the ADS valves to open. Sufficiently redundant instrumentation components are provided such that no single instrument failure will preclude initiation of all ADS (e.g., four channels of Reactor Vessel Water Level - Low Low Low, Level 1 are provided; two channels input to ADS trip system A and the other two channels input to ADS trip system B). However, the logic for ADS trip system A and ADS trip system B are powered from the same DC source (i.e., 250 VDC RMOV Board 1B which is powered by Unit Battery # 3). As such, the ADS is not single failure proof, but the design is such that no single failure shall prevent the integrated operations of the ECCS from providing adequate core cooling (e.g., a single

failure shall not result in the failure of both the ADS and the HPCI System functions).

For the HPCI System, the controls for the HPCI components (e.g., HPCI pump discharge valve) are powered from the 250 VDC RMOV Board 1A (which is powered by Unit Battery # 1). BFN Unit 1 TS LCO 3.5.1, "ECCS-Operating," requires the HPCI System to be operable in Mode 1 and in Modes 2 and 3, except when reactor steam dome pressure is ≤ 150 psig. The HPCI initiation logic is arranged such with Division II is powered by 250 VDC RMOV Board 1A (which is powered by Unit Battery # 1) and Division I is powered from 250 VDC RMOV Board 1B (which is powered by Unit Battery # 3). The arrangement is such that a loss of power to 250 VDC RMOV Board 1B does not result in loss of HPCI System initiation capability. However, loss of power to 250 VDC RMOV Board 1A does result in loss of HPCI System initiation capability. BFN Unit 1 TS LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," requires the HPCI System initiation instrumentation functions to be operable in Mode 1 and in Modes 2 and 3 with reactor steam dome pressure > 150 psig.

LOCA Analyses with Single Failure of ADS Automatic Actuation

The LOCA analyses supporting use of AREVA fuel for BFN Unit 1 is documented in Reference 1. The LOCA analyses are based on the plant parameters and potentially limiting ECCS single failures reflected in Reference 1 and Updated Final Safety Analysis Report (UFSAR) Table 6.5-3, "Single Failure Evaluation Used for LOCA Analysis." The BFN LOCA analyses were included as part of Technical Specification Change TS-467 supporting the transition to ATRIUM™-10 fuel at BFN Unit 1.

In defining the plant characteristics for the LOCA analysis, it had been previously identified that the ADS could not be disabled by a single failure. Therefore, it was not identified as a potentially limiting single failure that needed to be considered in the LOCA analyses. Subsequent investigation of the issue by TVA has concluded that the previous position was incorrect; a potential single failure was identified that could disable automatic initiation of the ADS (all valves). In addition, TVA determined that the potential failure of automatic initiation of the ADS had been considered in previous LOCA analyses and a requirement for operator action to initiate ADS (4 valves) within 10 minutes had been identified and credited in the BFN licensing basis. Current BFN emergency operating procedures are consistent with this requirement.

As a result, an assessment of the impact of a single failure of automatic ADS actuation on the BFN LOCA analyses documented in Reference 1 was performed.

<u>Assessment Basis</u>

For the single failure of automatic ADS actuation, the available ECCS systems as follows:

- For recirculation discharge line breaks: HPCI and 1 LPCS
- For recirculation suction line breaks: HPCI, 1 LPCS, and 2 LPCI (one loop)
- For feedwater line or HPCI injection line breaks: 1 LPCS and 2 LPCI (one loop)
- For LPCS line breaks: HPCI and 2 LPCI (one loop)
- For all breaks with loss of automatic ADS: Manual initiation of 4 ADS valves 10 minutes after event initiation

The analyses for this assessment were performed consistent with previously defined plant parameters with the following exceptions.

- The maximum core power level assumed (3458 MW x 1.02) is consistent with the current licensed operating core power for BFN Unit 1. The Reference 1 and 2 analyses were performed at a higher power level (3952 MW x 1.02) to support extended power uprate.

- The limiting single failure identified in Reference 1 resulted in the loss of several ECCS functions including HPCI. Therefore, HPCI capacity was not a factor in the Reference 1 analyses. The HPCI capacity values assumed for this assessment are consistent with HPCI design and testing requriements.

- The analyses assumed manual initiation of 4 ADS valves at 10 minutes.

<u>Recirculation Line Breaks</u>

For recirculation line breaks, the single failure of the ADS has the potential to be more severe than the single failures considered in Reference 1. ADS operation does not affect large recirculation line breaks (the break flow is high enough to depressurize the reactor and PCT occurs prior to ADS actuation). Intermediate size recirculation line breaks may be affected by ADS (the break flow is high enough to depressurize the reactor but ADS actuation occurs prior to PCT). Small recirculation line breaks may be significantly affected by ADS (the break flow is not high enough to depressurize the reactor).

Recirculation discharge line break LOCA analyses were performed to assess the impact of the ADS actuation single failure. Recirculation suction line breaks are bound by discharge line breaks as discussed in Reference 1. Analysis results are summarized below:

Break Size (ft^2)	PCT (°F)
0.4	1395
0.3	1840
0.25	2057
0.2	1840
0.15	<1000

For recirculation line breaks larger than 0.4 ft^2, the limiting single failure analyzed in Reference 1 will bound or be equivalent to the single failure of ADS actuation. Recirculation line breaks smaller than 0.15 ft^2 will be less limiting.

Non-Recirculation Line Breaks

Breaks in the HPCI injection line are equivalent to feedwater line breaks. In Reference 1, feedwater line breaks were dispositioned as being bounded by a same size break in the recirculation line. The disposition conclusion is valid for the limiting single failure identified in Reference 1; however, for a single failure of ADS actuation the qualitative disposition is not valid because the HPCI flow available for a recirculation line break is assumed to be lost for a break in the feedwater line.

Previous analyses by AREVA and also previous GNF BFN LOCA analyses indicate that a feedwater line break or HPCI injection line break is less severe than a recirculation line break. Feedwater line break LOCA analyses were performed to confirm this position for break sizes of 100% to 10% of the full pipe area. In all cases, the core remained covered through-out the blowdown period and PCT was less than 1000°F. The minimum vessel inventory occurred for a 25% line break (~0.27 ft^2).

Analyses of the LPCS line breaks were performed in Reference 1. For the single failure considered in these analyses, ADS actuation occurs and has a beneficial effect on PCT. The LPCS 0.4 ft^2 line break analysis was repeated for the single failure of ADS actuation. Unlike the single failure analysis in Reference 1, HPCI is available in this analysis and has a significant benefit in depressurizing the reactor faster and increasing liquid mass inventory in the vessel. For the LPCS assessment analysis, the core remained covered through-out the blowdown period and PCT was less than 1000°F. Smaller breaks in the LPCS line would be less severe.

Conclusions

The ADS actuation single failure will increase the PCT reported in Reference 1 for some small break LOCA analyses. However, the assessment demonstrates that the LOCA MAPLHGR limits remain applicable for BFN operation and ensure 10 CFR 50.46 acceptance criteria continue to be met with consideration for a single failure of the automatic actuation of the ADS.

As a result of this identified single failure, additional evaluations of the various impacted breaks will be performed at EPU conditions. Both the AREVA LOCA analysis provided in Reference 1, and the GE analysis provided in Attachments 3 and 4, will be supplemented by addendum reports. These addendum reports will provide the results of the additional evaluations of this identified single failure performed in accordance with NRC approved methodologies. The evaluations will credit the manual actuation of four ADS valves ten minutes into the event. In addition, the HPCI flow rates used in these evaluations will be revised to be consistent with existing Technical Specifications requirements. These evaluations will be completed by March 2010.

In addition, TVA is investigating the feasibility of modifying the ADS system such that a single battery failure will not result in the total loss of ADS function. If this modification is feasible, then the modified plant configuration would be consistent with the analysis assumptions used in the original LOCA analyses provided in Reference 1, as well as Attachments 3 and 4. In this event, the addendum reports would not be required to be relied upon as part of the design basis, since the LOCA design basis would be entirely supported by the original analyses provided in Reference 1 and Attachments 3 and 4.

Question 5.

Table 2.3, Cycle Specific Reload Evaluation Methodologies, of the Reload Safety Analysis Report (RSAR, ANP-2864(P), "Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis") lists more transients than Table 2.1 indicates are analyzed on a cycle-specific basis. Provide an explanation for the discrepancy.

Response 5.

Table 2.1 of the RSAR (ANP-2864(P)) provides a disposition summary of BFN FSAR events and analyses. The summary provides the disposition status and applicable comments associated with the event/analysis. The basis of the disposition status is categorized as:

- FSAR analysis.
- Generic analysis. A bounding analysis that is independent of plant type.
- Plant specific analyses. The analysis is based on BFN (independent of unit) and is bounding for cycle-to-cycle variations.
- Cycle specific analysis. The analysis is specific to the Unit and Cycle.

Increased Core Flow (ICF) and Maximum Extended Load Line Limit Analysis (MELLLA) operating regions of the power/flow map are included in the disposition results presented in Table 2.1.

Table 2.3 provides a listing of FSAR events, identified in Table 2.1, that require analyses performed on a cycle specific basis and specifically addressed in the RSAR. The table includes the approved evaluation model

and applicable methodology reference for each event/analysis. The table further provides applicable acceptance criteria and comments associated with each event/analysis.

The following event/analysis identified in Table 2.1 as being "addressed each cycle" or "address initial reload" are not included in Table 2.3, based on the following:

- FSAR Section 3.2. The analysis, acceptance criteria, methodology, and evaluation model are provided in Reference 2 of the RSAR.
- FSAR Section 3.6. The analysis, acceptance criteria, methodology, and evaluation model are provided in Reference 1 of the RSAR.
- FSAR Section 10.2. The analysis, acceptance criteria, methodology, and evaluation model are provided in Reference 24 of the RSAR.
- FSAR Section 10.3. The analysis, acceptance criteria, methodology, and evaluation model are provided in Reference 25 of the RSAR.
- FSAR Section 10.11. This item is addressed in Reference 37 of the RSAR.
- FSAR Section 14.5.5.1. Table 2.1 dispositions the event as a potentially limiting Anticipated Transient Without Scram (ATWS) pressurization event. Therefore the cycle-specific analysis becomes ATWS-Pressure Regulator Failure Open (PRFO), considered under Table 2.3, FSAR 7.19.
- FSAR Section 14.6.4. The analysis, acceptance criteria, methodology, and evaluation model are provided in Reference 27 of the RSAR.

Question 6.

The approved AREVA nuclear design method topical reports do not appear to describe a method for calculating the standby liquid control system (SLCS) cold shutdown margin (CSDM).

Provide a description of the SLCS CSDM calculation method. The methodology description should include:

a. A discussion of the codes used;
b. A discussion of the nuclear data that must be generated (e.g. lattice parameters for given boron concentrations);
c. A list of pertinent analysis assumptions (e.g. active core averaged boron weight),
d. A justification of the method accuracy;
e. A discussion on the treatment of short-lived highly absorbing nuclides (e.g. xenon and samarium); and,
f. A discussion of the assumed core thermal-hydraulic conditions (e.g. temperature of 68 degrees F).

This description should address the application of the method to the co-resident GE14 fuel.

Response 6.

    AREVA performs a cycle specific SLCS shutdown margin calculation as part of the plant reload analyses as specified on page 19 of Reference 2. The result for BFN Unit 1 Cycle 9 is reported in Section 7.3 of Reference 3. The SLCS shutdown margin analysis is performed using the NRC approved AREVA neutronics analysis methodology (i.e., CASMO-4/MICROBURN-B2) documented in EMF-2158(P)(A) (Reference 4).

    The approval of this neutronics methodology was in part based upon benchmarks for a number of reactors and fuel types as evaluated in Reference 4. This included a series of both borated and non-borated CASMO-4 criticality benchmarks based upon KRITZ and Babcock and Wilcox experimental data. These critical experiments include both PWR and BWR fuel assembly designs covering a range of fuel rod diameters and water to fuel ratios. Uniform arrays of rods as well as rod arrays with large gaps are included. The Reference 4 validation includes benchmarks to commercial reactors for hot and cold conditions. It also includes gamma traversing incore probe (TIP) and isotopic data measurement comparisons. Comparisons of fission rate distributions were also made to MCNP as part of the original Reference 4 validation that included lattices from modern AREVA and GE 10x10 fuel designs. BFN Unit 1 and the fuel types contained within the Cycle 9 core are consistent with the benchmarks provided in Reference 4. Furthermore, the results of benchmarking for BFN Unit 1 completed in support of the fuel transition meet the requirements of Reference 4 and include cores with the GE14 fuel that will be co-resident in the Cycle 9 reload design.

    Recent comparisons between CASMO-4 and MCNP have been performed for GE-11 and ATRIUM-10 fuel designs with boron in the coolant. The standard deviation of these results, covering moderator temperatures at 300 K (80.3° F) and 600 K (620.3° F) and boron concentrations between 0 and 1016 ppm, was [   ]. Standard deviations for both fuel types were nearly identical indicating no fuel type dependency.

    The fuel assembly cold and warm lattice nuclear data with boron in the coolant is generated with the CASMO-4 computer code as a function of lattice type, exposure, void history, boron concentration, temperature, and control. The SLCS shutdown margin is calculated with the MICROBURN-B2 reactor simulator code using nuclear data for the lattices generated by CASMO-4 calculations. Additional borated branch cases are performed with CASMO-4 to support the SLCS calculation. These CASMO-4 borated branches are performed for cold, intermediate, and warm temperatures with boron concentrations that correspond to the cold and warm temperature conditions. This additional data is used by MICROBURN-B2 in the same manner as the unborated data except that the code is capable of interpolating or extrapolating to boron concentrations other than those included in the lattice cross section library with high accuracy over a reasonable range. Even so, the amount of interpolation/extrapolation is typically minimized by choosing a warm temperature, usually 360°F, that is near the limiting temperature used for the calculation.

Cold conditions are not limiting for a SLCS calculation. For BFN Unit 1 Cycle 9, the SLCS calculation is performed at the temperature associated with RHR shutdown cooling initiation (i.e., it is based upon the saturation temperature corresponding with the shutdown cooling system initiation pressure). For SLCS conditions, higher temperatures are more reactive since B-10 exhibits a 1/v absorption cross-section (i.e., becomes a less efficient absorber with decreased moderation). The shutdown cooling initiation becomes the most reactive condition due to the dilution impact of adding the Residual Heat Removal (RHR) shutdown cooling volume as described in the BFN UFSAR section 3.8.4.1. The limiting nature of this higher temperature statepoint is illustrated in the answer to Question 7 below, where a direct comparison is made to a cold condition calculation.

The amount of boron assumed in the SLCS analysis is referenced to a natural Boron equivalent at cold conditions. For BFN Unit 1 Cycle 9 this concentration is 720 ppm natural boron equivalent at 70°F. The AREVA standard process is to use 68°F as the reference temperature which is consistent with Improved Standard Technical Specifications (e.g., definition of Shutdown Margin included in Section 1.1, "Definitions," of NUREG-1433). As mentioned in Section 7.3 of Reference 3, the AREVA cold analysis basis of 68°F represents a negligible difference and the results are adequate to protect the 70°F licensing basis of BFN Unit 1. This is further illustrated in the answer to question 7 below, where a comparison is provided between calculations assuming both 68°F and 70°F as the reference temperature.

For BFN Unit 1 Cycle 9, a series of SLCS shutdown margin restart calculations were performed at the desired cycle exposure points throughout the cycle. The MICROBURN-B2 control rod step out restart file used for the SLCS calculation was based upon the short previous cycle energy. In certain cases ,such as performing a SLCS analysis after the previous cycle has shutdown, or when performing a SLCS analysis after the previous cycle has exceeded the short energy window, the restart file may be based on the actual (or expected) previous cycle energy.

The SLCS shutdown margin calculations for BFN Unit 1 Cycle 9 assume a conservative all control rods out (ARO) condition. In some cases, SLCS shutdown margin calculations may be performed with some control rods in the core, as allowed by the licensing basis of the event (i.e., use of the rated power rod pattern is allowed by NUREG-0800, Section 9.3.5). If a rodded calculation is performed then the CASMO-4 calculations will include controlled borated branches. The SLCS shutdown margin is based upon the core k-eff at the required boron concentration and warm moderator temperature with no xenon. The samarium concentration is obtained from the restart file (operating samarium, except at Beginning of Cycle (BOC) which uses shutdown samarium).

Based on uncertainty analysis for the CASMO-4/MICROBURN-B2 methodology, AREVA has determined that the minimum calculated SLCS shutdown margin at the limiting temperature must be greater than [   ] Δk/k. The uncertainties evaluated are generic to the Reference 4 methodology and are applicable to BFN Unit 1. Although not applied to BFN Unit 1 Cycle 9,

a higher acceptance criterion is available for plants that desire to have this calculation performed at cold conditions. The higher acceptance criterion is provided to compensate for performance of the calculation at a non-limiting temperature. For BFN Unit 1 Cycle 9, the result is reported in Section 7.3 of Reference 3 and is conservatively well within the applicable **[   ]** Δk/k criterion.

The pertinent analysis assumptions applied in the BFN Unit 1 Cycle 9 SLCS shutdown margin calculation are the following:

- The cold critical target k-eff values as a function of exposure, based on measured cold critical data, are applicable. It is noted that the use of cold target is conservative since the values are lower than the corresponding hot operating target.
- The reactor is xenon free and has operating samarium (with shutdown samarium at BOC). The most significant of the isotopes is xenon and the assumption of no xenon is conservative for a shutdown margin calculation. The less significant samarium isotope is modeled to be consistent with expected conditions.
- The weight of boron assumed in the calculation at 366° F is referenced to the weight that corresponds to 720 ppm at cold conditions. This ensures that the calculation remains consistent with the Technical Specification bases.
- The saturation temperature consistent with the RHR initiation pressure (366° F) is the appropriate temperature for the analysis. This is shown to be the limiting temperature in the discussion above and demonstrated in the answer to question 7 below.
- The all-rods-out condition was assumed. This is conservative since the licensing basis allows credit for the rated conditions rod patterns.

Question 7.

Provide an analysis of SLCS shutdown margin at 68 degrees F, consistent with definition of rodded shutdown margin appearing in Unit 1 Technical Specifications (TSs). Demonstrate the capability to maintain subcritical configuration in cold conditions.

Response 7.

The SLCS analysis for BFN Unit 1 Cycle 9 was performed at 366°F, which is the reactor water temperature associated with RHR shutdown cooling initiation (i.e., the saturation temperature corresponding to the initiation pressure). As discussed in the answer to question 6, this temperature represents a more limiting condition than cold conditions. AREVA has performed a SLCS calculation at cold conditions (68°F) for the BFN Unit 1 Cycle 9 reload design. Results summarized in the table below confirm that the higher temperature is limiting and therefore provides a bounding result. Additionally, AREVA has performed a SLCS calculation at conditions defined in the BFN SLCS licensing basis documents (720 ppm boron at a cold temperature of 70°F). Results

summarized in the table below confirm that the AREVA cold analysis basis of 68°F represents a negligible difference and the results are adequate to protect the 70°F licensing basis of BFN Unit 1.

Warm (366° F), Boron equivalent to 720 ppm @ 68°F	Warm (366° F), Boron equivalent to 720 ppm @ 70°F	Cold (68°F) 720 ppm
2.98% Δk/k	2.98% Δk/k	3.81% Δk/k

Question 8.

Clarify the following language in the TS: "...latest approved versions applicable to BFN." Several of the references listed have supplements and addenda. Address why the supplements and addenda were not included and provide a listing of the latest approved version as well as the latest approved supplements and/or addenda to the listed topical reports.

Also, clarify the process that is followed when a new version, supplement, or addendum is approved by the NRC in the midst of the generation of the cycle operating limits report for the next cycle.

Response 8.

The term "... latest approved versions applicable to BFN" is a qualifier meaning TVA must review a new version, supplement, or addenda to make sure it is actually applicable to BFN (for example, maybe the vendor revises a report to include language specific to a BWR 6. In this case, the new revision would not be applicable to BFN because it is not a BWR 6. Or perhaps a change is being made relative to a fuel product not actually used by BFN, in which case the new revision supplement, or addenda is not applicable).

Proposed changes to Technical Specification 5.6.5 (provided Attachments 2 and 3 of Technical Specification Change TS-467) are the same as those already approved for BFN Units 2 and 3. This minimizes differences between the three units. Full listings of applicable revisions, supplements, or addenda and dates are provided in the Core Operating Limits Report (COLR), based on actual BFN applicability. An itemized listing of supplements applicable for inclusion in the COLR are shown in Technical Specification Change TS-467 Attachment 17 (ANP-2637, Revision 2, Boiling Water Reactor Licensing Methodology Compendium, Table 1-2, Reference No. column items: 2-3, 2-10, 2-12, 3-1, 4-3, 5-6, and 5-7).

The process that is used when a new version, supplement, or addendum is approved by the NRC in the midst of the generation of the COLR for the next cycle is as follows.

The vendor uses computer code versions corresponding with specific, approved methodologies applicable to BFN. At the beginning of the design process, the vendor is aware of methodology related issues which may be applicable to the design and licensing process. Computer code versions (i.e., approved methodologies) are specified

at the beginning of the cycle design process, independent of potential future revisions.

Results of licensing basis analyses are compiled in the Reload Safety Analysis, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), Fuel Cycle Design, and Mechanical Design Reports. The COLR is based on these reports. The COLR lists all approved methodologies utilized in the licensing basis analyses. At any point in the design and licensing process, the vendor may implement conservative changes to account for known methodology discrepancies and/or errors. Any methods changes used to develop COLR input are processed in accordance with the 10 CFR 50.59 process.

Development of the COLR proceeds through different stages. TVA Nuclear Fuels Engineering performs a technical review of vendor analyses and reports. The COLR and supporting information are assessed in accordance with the 10 CFR 50.59 screening and safety evaluation process. Finally, the COLR must be approved by the plant operations review committee (PORC) prior to inclusion in the Technical Requirements Manual (TRM). The approved COLR is sent to the NRC in accordance with the requirements of Technical Specification 5.6.5.d.

When a topical report revision, supplement, or addendum change is approved prior to the vendor analyses being completed and documented, it may be possible to incorporate the update during the normal COLR development process.

If the approved change constitutes an enhancement or relaxation, not involving a known discrepancy or error, then it would be processed by a future COLR revision based on revised vendor analyses, or wait until the next cycle design and licensing process.

If an approved change involves resolution of a known discrepancy or error, conservative updates to the methodology will have already been included in vendor analyses, and addressed via the 10 CFR 50.59 process, (the vendor is aware of pending NRC reviews, and the potential time line for approval, as well as necessary compensatory action, as discussed up front during the initial design phase). For the special case where the NRC has issued a new Safety Evaluation Report (SER) to close a 10 CFR 21 issue, the new SER can also be noted in the COLR in accordance with the 10 CFR 50.59 process, until such time as the vendor officially issues a revision to the approved methodology report.

Question 9.

Explain why the following topical reports are not included in TS 5.6.5:

- EMF-2209(P), Rev. 2, Addendum 1(P)(A), "SPCB Additive Constants for ATRIUM-10 Fuel," May 1, 2008;
- EMF-CC-074(P)(A), "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," Vol. 4, Siemens Power Corporation, August 2000;
- EMF-85-74(P), Supplement 1(P)(A) and Supplement 2 (P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998;
- BAW-10255(P)(A), Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," Framatome ANP, May 2008; and
- BAW-10247(P)(A), Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA, May 2008.

Response 9.

As reflected in reviewer's notes associated with Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," of NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 3.0, the individual specifications that address core operating limits are to be referenced in Technical Specification 5.6.5.a and the topical reports that reflect the analytical methods used to determine these core operating limits are to be included in Technical Specification 5.6.5.b by number and title. The reviewer's note to Technical Specification 5.6.5.b also indicates that COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

The current BFN Unit 1 Technical Specification 5.6.5.a lists the following individual specifications

(1) The Average Planar Linear Heat Generation Rates (APLHGRs) for Specification 3.2.1;

(2) The Linear Heat Generation Rate (LHGR) for Specification 3.2.3;

(3) The Minimum Critical Power Ratio (MCPR) Operating Limits for Specification 3.2.2; and

(4) The Rod Block Monitor (RBM) setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.

The Limiting Conditions for Operation (LCOs) for these Technical Specifications state that the associated limits are specified in the COLR.

- Consistent with the reviewer's note associated with NUREG-1433 Technical Specification 5.6.5.b, EMF-2209 (P)(A) and its title are included in the proposed BFN Unit 1 Technical Specification 5.6.5.b. On page 4 of the Enclosure to Technical Specification Change TS-467, in paragraph 4,

the status of the EMF-2209 addendum is discussed (the SER is identified as Reference 13 of the Enclosure). Section 4 of the SER relevant to the Addendum 1 of EMF-2209 Revision 2 states"

> "The NRC staff acknowledges that AREVA will combine this safety evaluation with the previously approved TRs, to issue Revision 3 of TR EMF-2209, and Revision 1 of TR ANP-10249. All parts of the latest revisions have been approved by the NRC staff. Therefore, Revision 3 of TR EMF-2209, and Revision 1 of TR ANP-10249, can be submitted as the approved versions of the TRs. This will allow use of current plant technical specification (TS) references without modifications to the standard TSs."

Therefore, the EMF-2209 Rev. 2 Addendum information was incorporated into EMF-2209 Rev. 2 to create the approved topical report EMF-2209PA Rev. 3 dated December 2009. The BFN Unit 1 COLR will incorporate EMF-2209PA Revision 3, consistent with the requirements of proposed Technical Specification 5.6.5.b (i.e., the latest approved versions applicable to BFN).

- The list of specifications, included in current BFN Unit 1 Technical Specification 5.6.5.a, for which core operating limits have been relocated to the COLR does not include limits associated with thermal hydraulic stability. As such, Technical Specification 5.6.5.b does not include a reference to EMF-CC-074(P)(A) since the analytical methods described in EMF-CC-074(P)(A) are associated with stability analysis.

- Consistent with the reviewer's note associated with NUREG-1433 Technical Specification 5.6.5.b, EMF-85-74 (P)(A) and its title are included in the proposed BFN Unit 1 Technical Specification 5.6.5.b. The BFN Unit 1 COLR will incorporate "EMF-85-74(P), Supplement 1(P)(A) and Supplement 2 (P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998," consistent with the requirements of proposed Technical Specification 5.6.5.b (i.e., the latest approved versions applicable to BFN) and the COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date and any supplements).

- As previously stated, the list of specifications, included in current BFN Unit 1 Technical Specification 5.6.5.a, for which core operating limits have been relocated to the COLR does not include limits associated with thermal hydraulic stability. As such, Technical Specification 5.6.5.b does not include a reference to BAW-10255(P)(A) since the analytical methods described in BAW-10255(P)(A) are associated with stability analysis.

- BAW-10247(P)(A) is not listed in proposed Technical Specification 5.6.5.b because the analytical methods described in BAW-10247(P)(A) are not used for the development of core operating limits for BFN Unit 1.

Question 10.

Section 4.8.1 of ANP-2860P, "Browns Ferry Unit 1 Summary of Responses to Requests for Additional Information" states that for fast pressurization transients surface heat flux calculations are provided to the licensee and that appropriate linear heat generation rate factor (LHGRFAC) values are developed based on Global Nuclear Fuel (GNF) thermal-mechanical analyses. Provide a description of the analysis procedures that are used to demonstrate that the GNF fuel meets applicable thermal-mechanical licensing limits. Describe the number of cases that were benchmarked in the safety analysis. This description should address both fast and slow transients.

Response 10.

AREVA Thermal Overpower (TOP)/Mechanical Overpower (MOP) Benchmarking Process

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GNF takes the information provided by AREVA and applies it to the process described in Attachment 5 to produce bounding TOP/MOP limits for application with AREVA's methods. These limits are then provided to AREVA.

Application of GNF Methods

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For analysis of transients, both with or without an early/direct scram, the appropriate GNF supplied TOP/MOP criteria and setdown procedure are implemented. This evaluation is performed for every event and state point analyzed to determine thermal margin.

Question 11.

    Provide the results of the Thermal Mechanical Analyses for GE14 fuel, which
    are not contained in the Reload Safety Analysis Report.

Response 11.

    The following table presents the event and state point specific limiting GE14
    TOP/MOP results used to develop the GE14 Linear Heat Generation Rate
    Factor (LHGRFAC) multipliers for EPU conditions, defined by the following
    events and equipment out of service:

    • LRNB: All LRNB events, including Power Load Unbalance Out of Service
      (PLUOOS), for all combinations of Feedwater Heater Out of Service
      (FHOOS), and Recirculation Pump Trip Out of Service (RPTOOS).
    • FWCF Turbine Bypass Valve in Service (TBVIS): All FWCF events for all
      combinations of FHOOS, and RPTOOS.
    • FWCF Turbine Bypass Valve Out of Service (TBVOOS): All FWCF events
      for all combinations of FHOOS, RPTOOS, and TBVOOS.

    The base case load rejection event already assumes the turbine bypass
    system inoperable. Separate TBVOOS multipliers are only needed for the
    FWCF event.

Power (% Rated)	LRNB MOP/TOP	FWCF TBVIS MOP/TOP	FWCF TBVOOS MOP/TOP
Technical Specification Scram Speed (TSSS) Insertion Times			
100	49 / 49	52 / 50	57 / 57
90	46 / 46	58 / 56	63 / 61
77.6	45 / 45	63 / 63	68 / 68
60	46 / 46	70 / 69	75 / 74
55	75 / 75	74 / 72	78 / 77
50	79 / 79	77 / 77	81 / 81
40	86 / 86	94 / 94	98 / 98
26	101 / 101	128 / 125	136 / 136
26 at > 50%F below P_{bypass}	---	165 / 164	219 / 217
26 at ≤ 50%F below P_{bypass}	---	161 / 151	175 / 174
23 at > 50%F below P_{bypass}	102 / 102	174 / 170	237 / 234
23 at ≤ 50%F below P_{bypass}	75 / 75	171 / 155	194 / 190

Power (% Rated)	LRNB MOP/TOP	FWCF TBVIS MOP/TOP	FWCF TBVOOS MOP/TOP
Nominal Scram Speed (NSS) Insertion Times			
100	42 / 42	49 / 47	54 / 52
90	44 / 44	57 / 54	62 / 59
77.6	43 / 43	61 / 61	66 / 66
60	42 / 42	68 / 67	74 / 72
55	74 / 74	72 / 71	76 / 75
50	78 / 78	76 / 75	80 / 80
40	85 / 85	93 / 93	97 / 97
26	100 / 100	124 / 124	133 / 133

Using calculated TOP/MOP data and the process described previously in the response to Question 10, required LHGRFAC multipliers are developed for all the transient events and state points used to determine thermal margin. Two sets of GE14 LHGRFACp multipliers are developed for plant operation, shown in Table 8.10 and Figures A.162 and A.164 of the RSAR. The multipliers are developed to support operation at all cycle exposures, both NSS and TSSS insertion times, and the EOOS conditions identified in Table 1.1 with and without TBVOOS. GE14 LHGRFAC multipliers further take into consideration the previous cycle limits developed by GNF (Cycle 8 was the final cycle prepared by GNF). Cycle 9 multipliers are the most restrictive value of Cycle 9 calculations and Cycle 8 existing limits.

Question 12.

Provide the reference, Mneimneh, GNF, letter to McNelley, TVA, "Revised LHGR Limits for BF1 Transition," MJM-TVA-ER1-09-39.

Response 12.

GNF has generated a document which summarizes the thermal mechanical information they provided to support the transition (Attachments 5 and 6). Sections 2-1 and 2-2 of Attachments 5 and 6 contain all of the technical information from the referenced GNF letter.

Question 13.

Provide additional details regarding the control rod drop accident (CRDA) analysis. Results of the CRDA analysis appear inconsistent with the methodology. Provide an updated description of the analytic method that

explains how the maximum number of rods exceeding 170 cal/g is determined, and what assumptions are used to make this determination.

Response 13.

The NRC-approved CRDA analysis method employed by AREVA is described in Section 4.1 and Section 7.1 of Reference 5. The Reference 5 parameterized methodology is used to determine the maximum enthalpy deposited in fuel rods. For each rod drop case with the maximum enthalpy exceeding 170 cal/gm, the minimum four-bundle local peaking factor (a combination of assembly radial peaking and lattice local peaking with pin power reconstruction effects incorporated) required to achieve a deposited enthalpy at the 170 cal/gm limit can be determined. Any assemblies that experience the combination of radial peaking factor and local peaking factor such that the minimum four-bundle local peaking factor is exceeded are counted as having failed rods. For BFN Unit 1 Cycle 9, the resulting number of assemblies exceeding 170 cal/gm was four (two ATRIUM-10 and two GE14). All rods in those four assemblies were conservatively assumed to be failed, resulting in 366 failed rods, which is well below the number of 850 assumed in the UFSAR. In some other cases where the result of assuming all rods in assemblies that exceed 170 cal/gm are failed is excessively conservative, the number of rods with a local peaking factor equal to or higher than that required to exceed the minimum four-bundle local peaking factor are counted. This less conservative approach of counting individual failed rods was not used in the BFN Unit 1 Cycle 9 analysis. For BFN Unit 1 Cycle 9, the result is reported in Section 6.2 of Reference 3.

The assumptions applied in determining the number of failed rods in the BFN Unit 1 Cycle 9 CRDA calculation are the following:

- All rods are assumed to be failed in any assembly that is determined to deposited enthalpy of 170 cal/gm or greater.
- Pin power reconstruction is assumed in the local peaking factor applied to determine the four-bundle local peaking factor.

Question 14.

The approved methodology relies on parameterization of generic analysis results. Provide the parameterized function. Given the application to GE14, justify the applicability of the generic analyses to modern fuel bundle designs; in particular describe how the differences in bundle fuel mass are accounted for in the parameterized function.

Response 14.

The NRC-approved CRDA parameterized function employed by AREVA is described in Section 7.1 of Reference 5. AREVA has performed analyses to justify the continued applicability of the existing Reference 5 parameterized function to evolutions in fuel design (8X8, 9X9, and 10X10 lattices), core

design, and AREVA neutronics methodology through the years since Reference 5 was approved.

As documented in Section 7.1 of Reference 5, the AREVA method for CRDA analysis [

]. For BFN Unit 1 Cycle 9, the analysis was completed at BOC and at the high reactivity point in the cycle (peak hot excess reactivity), and at end-of-full power (EOFP).

The NRC approved Reference 5 parameterized analysis incorporates additional conservatism by assuming adiabatic conditions during the power excursion (i.e., no direct moderator heating is credited during the analysis), and that the reactor remains at hot zero power conditions for the entire Banked Position Withdrawal Sequence (BPWS). As reported to the NRC in Reference 6, the adiabatic assumption has been evaluated to be conservative by approximately a factor of 2.

The factor in the parameterization most affected by bundle fuel mass is the Doppler coefficient. AREVA evaluates the Doppler coefficient for each fuel type with CASMO-4 calculations and has conservatively determined applicable Doppler coefficients for different fuel designs with different fuel masses, including all of the fuel types resident in the BFN Unit 1 Cycle 9 core design.

Question 15.

Describe the core conditions that are evaluated to determine the limiting values for the following parameters: (1) control blade worth, (2) four bundle local peaking factor, (3) Doppler coefficient, and (4) delayed neutron fraction.

Response 15.

The maximum dropped rod worth is determined assuming core conditions of no xenon and at hot standby conditions (i.e., $T_{fuel} = T_{mod}$ = hot operating temperature, zero void). Rod drop calculations are completed for as many rods in each group as required, based on symmetry, to determine the limiting dropped rod. For rod pull sequences that are quarter-core symmetric, rod drop calculations can be completed for rods in one quarter of the core, and for half-core symmetric situations only rods in half the core can be analyzed. Control rods from each of the planned operating sequences are included in the analysis. For BFN Unit 1 Cycle 9, the determination of maximum dropped rod worth incorporated the effect of inoperable rods, which has the effect of making the maximum dropped rod worth more conservative. As shown in Section 6.2 of Reference 3, the maximum dropped rod worth for BFN Unit 1 Cycle 9 was 11.17 mk.

As shown on page 159 in Section 4.1 of Reference 5, the four bundle local peaking factor consists of the radial peaking factors for the four bundles in the control cell and the local peaking factors associated with those same four fuel assemblies. Both the radial peaking factors and the local peaking factors are taken from MICROBURN-B2 calculations with pin power reconstruction under the same conditions as the dropped rod worth, i.e., no xenon and hot standby conditions (i.e., $T_{fuel} = T_{mod}$ = hot operating temperature, zero void). As shown in Section 6.2 of Reference 3, the four bundle local peaking factor applied in the BFN Unit 1 Cycle 9 limiting case was 1.382.

The Doppler cross-section input used in the parameterized CRDA analysis reported in Reference 5 is correlated to a core average Doppler reactivity coefficient which can be determined for a specific cycle. AREVA has determined conservative generic Doppler coefficients specific to different fuel designs (ATRIUM-10, GE-13, GE-14, etc) for use in applying the CRDA parameterization on the cycle-specific analysis basis. For mixed cores, the core average Doppler coefficient is obtained by volume weighting the values for each fuel type. Alternatively, if additional conservatism is tolerable, the least negative coefficient for the fuel in the core can be used. For new fuel designs and special conditions not covered by the conservative generic Doppler coefficients, CASMO-4 can be used to generate the CRDA Doppler coefficient. Any such CASMO-4 calculations would be done assuming the same conditions assumed in the generation of the generic Doppler coefficients, which is no xenon, 40% void history, and 0% instantaneous voids. As shown in Section 6.2 of Reference 3, the core average Doppler coefficient applicable to BFN Unit 1 Cycle 9 was -10.51×10^{-6}.

For BFN Unit 1 Cycle 9, the End of Cycle (EOC) delayed neutron fraction that is calculated for the plant transient analysis was used for the rod drop calculation. The plant transient value was calculated with an exposure window added to the EOC exposure resulting in additional conservatism. The value is calculated using MICROBURN-B2 and assuming at-power conditions with equilibrium xenon. If plant transient results from MICROBURN-B2 are not available, the core average delayed neutron fraction can be calculated by using CASMO-4 results. For each fuel type at EOC conditions, the CASMO-4 delayed neutron fraction data for the dominant lattice of each fuel type must be determined using interpolation on exposure and void history. Once the individual fuel type β_{eff} values have been determined for the batch average exposure and void history, the core average β_{eff} can then be determined by volume weighting the fuel type β_{eff} values. As shown in Section 6.2 of Reference 3, the effective delayed neutron fraction for BFN Unit 1 Cycle 9 was 0.0052.

Question 16.

Discuss the relationship between the maximum dropped rod reactivity worth and the cycle analyses provided in the fuel cycle design report. This documentation should specify the limiting rod, the method used to identify the limiting rod, the limiting point in exposure, and any consideration given for operational flexibilities (e.g., suppressing power in a leaking fuel bundle).

Response 16.

> The rod drop calculations for BFN Unit 1 Cycle 9 were completed at BOC, at the peak reactivity point in the cycle, and at EOFP, based on the cycle analysis provided in the fuel cycle design report. Rod drop calculations were completed for as many rods in each group as required, based on symmetry, to determine the limiting dropped rod. For rod pull sequences that are quarter-core symmetric, rod drop calculations can be completed for rods in one quarter of the core, and for half-core symmetric situations only rods in half the core can be analyzed. Control rods from each of the planned operating sequences were included in the analysis. For BFN Unit 1 Cycle 9, the limiting rod was found to be rod 18-47 in BPWS group 2 of sequence A at the peak hot excess reactivity point in the cycle.

> The CRDA licensing basis of most plants allows some number of inoperable rods. For these plants, the rod drop calculations are completed for rods on the side of the core opposite the location of the assumed inoperable rods. The location and separation of the assumed inoperable rods is chosen to maximize the worth of the remaining rods while maintaining Technical Specification spacing requirements between inoperable control cells. For the BFN Unit 1 Cycle 9 CRDA, a maximum total of eight inoperable rods were assumed, with a maximum of three in any BPWS group.

> AREVA has had experiences with plants inserting multiple suppression rods and has evaluated the CRDA, with the result being that the effect of the suppression rods is bound by the conservative treatment of the inoperable rods assumed in the CRDA analysis.

Question 17.

> The American Society of Mechanical Engineers overpressure analysis using ATRIUM-10 fuel and AREVA methods credits the failure of a direct scram as the limiting single failure. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for the Nuclear Power Plants", Chapter 5.2.2 assumes a reactor scram on the second safety-grade scram signal. This is a condition of the analysis, not an assumed failure. Identify the limiting single failure assumed in this analysis.

Response 17.

> AREVA performs the American Society of Mechanical Engineers (ASME) overpressure analysis assuming the second safety grade signal from the reactor protection system, the high neutron flux trip signal, initiates the reactor scram. As noted in the question, this is consistent with NUREG-0800 Chapter 5.2.2, as a condition of the analysis.

> Item C, page 5.2.2-6 of NUREG-0800 Chapter 5.2.2 states the following.

"A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power."

This is consistent with the AREVA ASME overpressurization analyses and Abnormal Operational Transient (AOT) analyses in general. The ASME overpressurization analysis does not credit systems if a single malfunction or failure of an active component precludes safety related portions of the system from functioning. No additional single failure considerations are included in the ASME overpressurization analysis.

Question 18.

The RSAR dispositions criticality for new and spent fuel storage based on previous analyses. Address why the use of previous analyses are acceptable.

Response 18.

The TVA requested NRC approval of AREVA ATRIUM-10 fuel design and storage for Units 1, 2, and 3 in a License Amendment Request dated February 13, 2003 (Letter from R. G. Jones (TVA) to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) Change 421 - Framatome Fuel Design and Storage," dated February 13, 2003, ADAMS Accession Number ML030560671). Supplemental information to support the request was provided to the NRC on April 14, 2003 (Letter from T. E. Abney (TVA) to NRC, "Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 – Technical Specifications (TS) Change 421 – Framatome Fuel Design and Storage – Supplemental Information" dated April 14, 2003, ADAMS Accession Number ML031130549).

NRC subsequently issued the associated License Amendments for Units 1, 2, 3 on September 5, 2003. The NRC approval and associated safety evaluation report for ATRIUM-10 fuel design and storage at Units 1, 2, and 3 are included in the letter from K. N. Jabbour (NRC) to J. A. Scalice (NRC), "Browns Ferry Nuclear Plant, Units 1, 2, and 3, Re: Issuance of Amendments (TAC Nos. MB7743, MB7744, MB7745) (TS 421)," dated September 5, 2003, ADAMS Accession Number ML032520003.

Technical approval for the criticality basis of ATRIUM-10 fuel was determined by TVA in accordance with the 10 CFR 50.59 process. Explicit Unit 1, 2, and 3 spent fuel storage pool analyses were performed by AREVA in 2003 prior to the introduction of ATRIUM-10 fuel in 2004. These analyses explicitly evaluated the storage rack configuration for a bounding enrichment design.

Fuel reactivity is calculated in the spent fuel storage pool rack lattice to demonstrate that the k-effective (k-eff) is less than 0.95. The AREVA criticality analyses utilized an accepted computer code (KENO). The analyses concluded a reference lattice design does not exceed an array k-eff of 0.95 in the BFN spent fuel storage pool. Fuel with a lower enrichment and/or more

gadolinia than the reference lattice will also not result in exceeding a k-eff of 0.95. The reference fuel design corresponds to a CASMO-4 cold k-eff reactivity of 0.872. Consequently, if the reference lattice conditions regarding enrichment and gadolinia are not met, cycle specific fuel designs can be verified to be in compliance by performance of criticality analyses with CASMO-4 to confirm the cold k-eff is less than 0.872.

Cycle specific fuel designs are verified to be within the bounding design basis prior fuel manufacture.

The Unit 1 spent fuel rack and fuel designs have been verified to remain within the design basis.

Question 19.

The RSAR dispositions Final Safety Analysis Report (FSAR) Section 14.5.2.8 - Pressure Regulator Failure Downscale - by stating that this event is eliminated as an Anticipated Operational Transient by installation of a digital fault-tolerant main turbine electro-hydraulic control (EHC) system. Address why this event was eliminated and whether the modification was made to support the transition to AREVA fuel.

Response 19.

Section 14.5.2.8 of the BFN UFSAR details the basis for this disposition. The modification was not made as part of the AREVA fuel transition. The modification was made to BFN Unit 1 prior to unit recovery. The UFSAR discussion is as follows.

14.5.2.8 Pressure Regulator Failure
Approval to remove the pressure regulator downscale failure event as an abnormal operational transient was approved by License Amendment Nos. 244, 281, and 239 to Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 by NRC on April 4, 2003, based on the installation of a fault-tolerant electro-hydraulic turbine control system on Units 2 and 3, and a commitment to similarly modify Unit 1 prior to return to power operation. The reliability of the upgraded electro-hydraulic control system is such that a system failure that results in the simultaneous closure of all turbine control valves is not an anticipated failure and, hence, the Pressure Regulator Downscale Failure (PRDF) transient no longer merits evaluation as an AOT.

Question 20.

The RSAR dispositions FSAR Chapter 14.5.6.4 – RCP [Reactor Coolant Pump] Rotor Seizure Accident - based on the fact that its consequences are bounded by the LOCA accident; however, the acceptance criteria are listed in terms of minimum critical power ratio (MCPR) and peak pressure. Provide an explanation for the discrepancy.

Response 20.

UFSAR 14.5.6.4 addresses the one recirculation pump seizure AOT event. The physical phenomena of this event are driven by the core response to the pump seizure. Flow through the affected loop is rapidly reduced due to the large hydraulic resistance introduced by the stopped rotor. This causes the core thermal power to decrease and reactor water level to swell. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer.

The pump seizure analysis in UFSAR 14.5.6.4.3 shows that the peak neutron and heat fluxes did not increase above the initial conditions, nor the peak vessel pressure challenge the nuclear system process barriers. The calculated ΔCPR was well below that for other types of transients analyzed. Therefore, no impact on fuel integrity occurs and no nuclear system process barrier damage results from the pump seizure event. Since there is no release of radioactive material beyond that of normal operation, the acceptance criteria of 10 CFR 50.67, "Accident source term," are met.

The core response during this event is not significantly affected by fuel design. ATRIUM-10 and GE14 fuel are both 10X10 fuel types designed with similar core hydraulics. While differences in the fuel neutronic design will have some impact on the transient response, the affect of any difference will be small compared with the margin to the thermal limits. Since the differences in the ATRIUM-10 and GE14 fuel designs will not significantly affect the transient response, no damage to either fuel type will occur and no nuclear system process barrier damage will result from the pump seizure event. Since there is no release of radioactive material beyond that of normal operation, the acceptance criteria of 10 CFR 50.67 are met.

The pump seizure event has been reclassified as an accident as noted in the footnote in UFSAR 14.5.6.4. The pump seizure event is a very mild accident in relation to LOCA. This is easily verified qualitatively by consideration of the two events. In both accidents, the recirculation driving loop flow is lost extremely rapidly. In the case of the seizure, stoppage of the pump occurs; for the LOCA, the severance of the line has a similar, but more rapid and severe influence. Following a pump seizure event, flow continues, water level is maintained, the core remains submerged and this provides a continuous core cooling mechanism. However, for the LOCA, complete flow stoppage occurs and the water level decreases due to loss of coolant, resulting in uncovery of the reactor core and subsequent overheating of the fuel rod cladding. In addition, for the pump seizure accident the reactor pressure does not significantly decrease, whereas complete depressurization occurs for the LOCA. The increased temperature of the cladding and reduced reactor pressure for the LOCA combine to yield a much more severe stress and potential for cladding perforation for the LOCA than for the pump seizure. Therefore, it can be concluded that the potential effects of the hypothetical pump seizure accident are conservatively bounded by the effects of a LOCA and specific analyses of the pump seizure accident are not required.

The pump seizure event, as described above, meets the acceptance criteria of minimum critical power ratio, peak pressure, and radioactive material release of 10 CFR 50.67.

References

1.   EMF-2950(P) Revision 1, Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis, AREVA NP, April 2004.

2.   XN-NF-80-19(P)(A) Volume 4 Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.

3.   ANP-2864(P) Revision 1, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis, AREVA NP, October 2009.

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

5.   XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Exxon Nuclear Company, March 1983.

6.   Letter from R. A. Copeland (Siemens Power Corporation) to R. C. Jones (U.S. NRC Reactor Systems Branch), "Additional Information on SPC Topical Reports," RAC:95:143, November 2, 1995.