Withhold from Public Disclosure Under 10 CFR 2.390

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

November 9, 2010

10 CFR 50.4(b)(6)
10 CFR 50.34(b)
10 CFR 2.390(d)(1)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Requests for Additional Information

This letter responds to a number of requests for additional information (RAIs) regarding the Unit 2 FSAR.

Enclosure 1 provides the responses to RAIs involving multiple FSAR chapters.

Enclosure 2 contains information proprietary to Westinghouse. TVA requests that the Westinghouse proprietary information be withheld from public disclosure in accordance with 10 CFR § 2.390.

Enclosure 3 contains the nonproprietary versions of the information provided in Enclosure 2 and provides Westinghouse's affidavit for withholding.

Enclosure 4 contains the response to **RAI 15.3.1 - 1.a.**

Enclosure 5 provides *Siemens Steam Turbine Engineering Report – Missile Report CT-27467, Rev 0.* This report was requested by the NRC staff.

Enclosure 6 provides documents noted in various RAI responses.

Enclosure 7 provides the new commitments contained in this letter.

If you have any questions, please contact Bill Crouch at (423) 365-2004.

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NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 9th day of November, 2010.

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

Enclosures:

1. Response to RAIs Regarding Unit 2 FSAR
2. Information Proprietary to Westinghouse
3. Nonproprietary Versions and Affidavit for Withholding
4. Response to **RAI 15.3.1 - 1.a.**
5. Submittal of Siemens Steam Turbine Engineering Report – Missile Report CT-27467, Rev 0
6. Attachments
7. List of New Regulatory Commitments

References:

1. NRC to TVA letter dated July 12, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Licensee's Final Safety Analysis Report Amendment Related to Electrical Engineering Systems (TAC No. ME2731)" (ADAMS Accession No. ML101530354)

2. NRC to TVA letter dated August 19, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Amendment Related to Section 11 (TAC No. ME3945)" (ADAMS Accession No. ML102240513)

3. NRC to TVA letter dated September 8, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Licensee's Final Safety Analysis Report Amendment Related to Chapter 14, 'Initial Test Program' (TAC No. ME2731)" (ADAMS Accession No. ML102380431)

4. NRC to TVA letter dated September 16, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Chapters 4, 5, 6 and 9 (TAC No. ME4074)" (ADAMS Accession No. ML102530464)

5. TVA letter to NRC dated July 31, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Preliminary Requests for Additional Information and Requests For Additional Information" (ADAMS Accession No. ML102290258)

6. NRC to TVA letter dated September 20, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Related to Section 15 (TAC No. ME4074)" (ADAMS Accession No. ML102590244)

7. TVA to NRC TVA letter dated July 2, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 - Submittal of Additional Information Requested During May 12, 2010, Request for Additional Information (RAI) Clarification Teleconference Regarding Environmental Review (TAC No. MD8203)" (ADAMS Accession No. ML101930470)

8. NRC to TVA letter dated September 28, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Amendment Related to Sections 3.2 and 5.2 (TAC No. ME2731 and ME3091)" (ADAMS Accession No. ML102630598)

9. NRC to TVA letter dated September 28, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Amendment Related to Section 7.3 (TAC No. ME2731)" (ADAMS Accession No. ML102640324)

10. NRC to TVA letter dated October 4, 2010, "Watts Bar Nuclear Plant, Unit 2 - Request for Additional Information Regarding Final Safety Analysis Report Related to Section 15 (TAC No. ME4074)" (ADAMS Accession No. ML102700437)

cc (Enclosures):

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ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Chapters 8 and 9 (from NRC letter dated July 12, 2010 (ADAMS Accession No. ML101530354))

8.2 - 3. *FSAR Section 8.2.1 describes the 161 Kilovolt (kV) preferred offsite power supply from Watts Bar Hydro Plant Switchyard (WBHS) for dual Unit operation at the WBN. The FSAR states that transmission system studies were completed to show the adequacy and capability of one 161 kV line and one CSST for starting and running all required safety-related loads for a DBA in WBN Unit 1 and no fuel load in WBN Unit 2. Since WBN Unit 2 loads will be supplied from the same 161 kV preferred power supply, the staff requests the below listed information with regard to the adequacy and capability of the 161 kV preferred offsite power supply for dual-unit operation (DBA in one unit and a concurrent shutdown of the other unit).*

Note: For ease of providing a response, the RAI was split into two parts.

a. *Provide a detailed discussion describing all such transmission system grid conditions and also describe in detail the operating characteristics of the offsite power supply at the WBHS (for dual-unit operation) including operating voltage range, post-contingency voltage drops (including bounding values and post-unit trip values), operating frequency range, etc. In addition, provide the design operating voltage range of the shutdown boards (minimum and maximum voltage) and how low the WBHS voltage can drop (assuming operation of LTCs) while still supplying the worst case shutdown board loading at the minimum design voltage of the shutdown boards.*

Response: The preferred offsite power system at WBN is normally supplied from TVA's 161-kV transmission grid at the Watts Bar Hydro Plant switchyard. Normally the frequency of the grid is 60 Hz, with very small perturbations above and below this value. The TVA Under Frequency Load Shed scheme is compliant with NERC/SERC standards and the first step will begin tripping transmission system load at 59.5 Hz. The final step in the program trips load at 58.7 Hz. Current studies show that the frequency will not drop below 57.5 Hz during any credible extreme contingencies.

The criteria used in the planning of the Transmission system state that the 161-kV voltage should not drop below 95% of nominal voltage for NERC Category B or C events. Normally the 161-kV grid at the WBN offsite power buses operates at 166 kV, with ranges from 161 kV to 170 kV occasionally observed.

Two Transmission System Studies (TSSs), a Planning TSS and an Operations TSS, are performed by Power System Operations (PSO) tri-annually or as needed. The Planning TSS is a 5 year look-ahead study to ensure the transmission network will meet the WBN voltage criteria. Transmission enhancements are made if needed. The Operations TSS is used to assure the network can meet the grid criteria during real time operation. In extreme cases, if the grid is unable to meet voltage criteria, the Transmission Operator will

immediately notify the WBN Generator Operator that offsite power is disqualified.

a. Operating characteristics of the preferred offsite power supply (at Watts Bar Hydro Plant Grid): 164 kV nominal

b. Voltage criteria for WBN for dual-unit operation:

- 161 kV Switchyard: $\geq$ 153 kV and $\leq$ 9 kV drop (post-event)
- 24 kV Generator Buses*: $\geq$ 23 kV and $\leq$ 24.8 kV

* Applicable only when utilizing Unit Board feeders as offsite power (the Unit Station Service Transformers [USSTs] supply offsite power until they transfer to the Common Station Service Transformers [CSSTs] A and B).

c. Post-contingency voltage drops (dual-unit operation): 9 kV Maximum

(The grid studies show that under the worst case scenario the maximum voltage drop will not exceed 5 kV. The auxiliary power system analysis for two unit operation has been performed using a 161 kV grid voltage drop of 11 kV when powered from CSSTs C and D and 9 kV when powered from CSSTs A and B. CSSTs A and B will be used to substitute for CSSTs D and C, respectively, in case of CSST C or D outage.)

d. Bounding value & Post unit trip value: 153 kV (Minimum)

(The grid studies establish that there are no voltage criteria violations under all grid operating conditions.)

e. Operating frequency range (dual-unit operation): Normally the frequency of the grid is 60 Hz with very small perturbations and is compliant with NERC/SERC standards and the first step begins tripping transmission system load at 59.5 Hz.

f. Design operating voltage range of the shutdown boards: 7260 V max; 6570 V min

g. How low the WBHS voltage can drop: 153 kV

b. *Also provide an executive summary of the grid studies completed for dual-unit operation including when the studies were completed. The summary of the grid studies must address dual-unit operation, the transmission network interface available fault current changes, the impact on the switchyard and plant switchgear and cabling. The summary also must include the overall inputs, assumptions, and summary of the output results (with acceptance criteria).*

Response: Attachments 1 and 2 (note that Attachment 2 contains only the first 78 pages of the calculation) provide the requested information with respect to inputs, assumptions, and summary of the output results (with acceptance criteria) of grid studies or planning TSS completed for WBN. This study was recently completed and issued on September 20, 2010; this study includes the impact of dual-unit operation. The study evaluates the WBN offsite power system to confirm that the system adequately meets the requirements of GDC 17. It also documents the load flow analysis of the WBN offsite power system considering all system pre-event outages. The analysis is based on summer 2010 network topology (including future projects) as well as load and generation forecasts.

Based on the auxiliary power system analysis, all plant switchgear and cabling have been determined to be adequate to support dual-unit operation. Switchgear is adequately rated for the available short circuit, dual-unit operation loads, and adequate voltage is available at the switchgear buses for proper operation of all equipment required for safe shutdown of the plant.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Chapter 11 (from NRC letter dated August 19, 2010 (ADAMS Accession No. ML102240513))

11 - 4. *Revising section and table numbers impacts the NRC's Safety Evaluation Report. Provide a cross-walk road map for the revised number of Chapter 11.*

Response: The requested cross-walk road map is provided as Attachment 3.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAI for FSAR Chapter 14 (taken from NRC letter dated September 8, 2010 (ADAMS Accession No. ML102380431))

No number *The Watts Bar Unit 2 (WBN2), FSAR Section 14.2.7, states that the initial test program will be developed and conducted in accordance with Regulatory Guide 1.68, Revision 2 (August 1978) "Initial Test Programs for Water-Cooled Nuclear Power Plants" except for the specific listed exceptions. Regulatory Guide 1.68, Section 5 lists the following test as one that should be included in the power-ascension test phase. "bb Conduct neutron and gamma radiation surveys to establish the adequacy of shielding and identify high-radiation zones as defined in 10 CFR Part 20." The WBN2 FSAR, Table 14.2-2, Sheet 18, summarizes the Radiation Baseline Survey Test to be performed, but doesn't commit to any regulatory guidance or standards for conducting the test.*

Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," Revision 1 (May 2009), Section C.3 endorses American National Standards Institute/American Nuclear Society document ANSI/ANS-6.3.1-1987; R2007, "Program for Testing Radiation Shields in Light Water Reactors (LWR)," which describes a test program to be used in evaluating biological radiation shielding in nuclear reactor facilities under normal operating conditions, including anticipated operational occurrences.

The Commission, in Staff Requirements Memorandum SECY-07-096, directed the staff to encourage TVA to adopt updated standards for Unit 2 where it would not significantly detract from design and operational consistency between Units 1 and 2. Conducting the test program in accordance with ANSI/ANS-6.3.1 may substantially improve the confidence that the WBN2 shielding results in as low as reasonably achievable exposures to personnel and the confidence in equipment qualification assumptions. The staff requests TVA to evaluate conducting the Radiation Baseline Survey Test in accordance with ANSI / ANS-6.3.1-1987; R2007 or otherwise show how adopting the updated standard for the test program significantly detracts from design or operational consistency between Unit 1 and 2.

Response: The Radiation Baseline Survey Test for Unit 2 will be performed consistent with the test performed for Unit 1, RCI-126 (Radiation Baseline Survey). RCI-126 listed ANSI/ANS-6.3.1-1987, "Program for Testing Radiation Shields in Light Water Reactors (LWR)" as a developmental reference. This test will satisfy the requirements of this document.

Note that ANSI/ANS-6.3.1-1987, R2007, is neither an update nor a revision of the 1987 standard. The "R" in the numeric designation denotes that the 1987 standard was reaffirmed as an American National Standard in 2007; it was also reaffirmed in 1998.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Sections 4, 5, 6, and 9 (from NRC letter dated September 16, 2010 (ADAMS Accession No. ML102530464))

4.4.1 – 1. ***Safety Evaluation Report (SER) 4.4.4, "Performance in Safety Criteria" (FSAR 4.4.1)***

FSAR Table 4.4-1 provides the design limit from nucleate boiling ratio (DNBR) values. Provide:

a. *The safety limit DNBR values for typical and thimble cells, for use with the Revised Thermal Design Procedure, and the Standard Thermal Design Procedure methods.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively, of the response to this RAI.

b. *The DNBR uncertainty allowances and margins.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively, of the response to this RAI.

c. *A discussion about how the uncertainty allowances and margins are added to the design DNBR values to develop the DNBR safety limit values.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively of the response to this RAI.

5.2.2 – 2. *SER Section 5.0, Reactor Coolant System and Connected Systems"*

a. *SER 5.2.2, "Overpressurization Protection" (FSAR 5.2.2)*

(2) *Describe the evaluations of the heat and mass input events at low temperature to support conclusion that "the allowable limits will not be exceeded and therefore will not constitute an impairment to vessel integrity and plant safety." Discuss which event (the mass addition event or the heat input event) is more limiting, and why.*

Response: The mass input and heat input transients are evaluated according to the methodology within WCAP-14040-NP-A, Revision 4. This NRC-approved document contains methodology used to calculate Power Operated Relief Valve (PORV) setpoints that will prevent the allowable limits from being exceeded. This methodology is applicable to WBN Unit 2 as stated in Section 5.9.6 of the Unit 2 Technical Specifications.

The Low Temperature Overpressure Protection System (LTOPS) PORV setpoint for WBN is variable with Reactor Coolant System (RCS) temperature and is set such that,

with the calculated setpoint overshoot from the maximum of either the mass addition or heat addition event at any given temperature and with other conservative considerations, the 10 CFR 50, Appendix G, limits will not be exceeded for the design basis events. The PORV setpoint overshoot resulting from the mass addition event is not sensitive to RCS temperature and is essentially constant at all RCS temperatures. The PORV setpoint overshoot for the heat addition event is relatively small at low LTOPS RCS temperatures at approximately one third that of the mass addition setpoint overshoot. As RCS temperature increases, the PORV setpoint overshoot resulting from the heat addition event increases significantly, to approximately double that of the mass addition event at the maximum LTOPS RCS temperature. As stated above, the LTOPS PORV setpoint is selected according to the maximum overshoot at the given RCS temperature, and the peak RCS pressure will therefore remain below the limit for either event. The mass addition event is therefore limiting at lower temperatures and the heat addition event is limiting at higher temperatures.

b. *SER 5.4.3, "Residual Heat Removal System [RHR]" (FSAR 5.5.7)*

Provide a justification for the difference between Units 1 and 2 in the time to cool the reactor coolant system from 350 °F to 140 °F (i.e., Unit 2 RHR time period for cool down is 16 hours which is 3 hours shorter than in Unit 1).

Response: There are two major contributors to the differences in the cooldown times reported in the Unit 1 UFSAR and the Unit 2 FSAR:

1. Unit 1 Technical Specifications surveillance requirement (SR) 3.7.9.1 specifies a limit of "≤ 85°F" for the Ultimate Heat Sink. Per the TS Bases for 3.7.9, "The UHS provides a heat sink ... by utilizing the Essential Raw Cooling Water (ERCW) System and the Component Cooling System." In 2006, TVA was pursuing a Technical Specifications amendment to raise the UHS limit from 85°F to 88°F. A reanalysis of the Unit 1 cooldown was performed to support that Technical Specifications amendment request, and the results were added to the FSAR. The Technical Specifications amendment request was withdrawn in 2007. Since the cooldown analysis using an 88°F UHS is conservative and bounding, TVA chose to keep the 88°F value in the Unit 1 UFSAR. The Unit 2 cooldown analysis was performed using the current proposed Unit 2 Technical Specification limit of 85°F.

2. The ERCW header piping layout feeding the Component Cooling System (CCS) Train A Heat Exchangers is not symmetrical between the two units. This results in the Unit 2 Train A of CCS receiving more flow than the Unit 1 counterpart as shown in the following table:

Parameter	Unit 1	Unit 2
ERCW Inlet Temperature to the CCS Heat Exchangers (°F)	88	85
ERCW Flow to the respective Train A CCS Heat Exchangers (Mlb_m/hr)	4.149	5.022

6.3 – 3. *SER Section 6.3, "Emergency Core Cooling System [ECCS]" (FSAR 6.3)*

a. *FSAR 6.3.1, "Design Bases"*

FSAR Section 6.3.1.3, "Reactivity Required for Cold Shutdown" states that "[d]uring a steam line break outside containment, the refueling water storage tank (RWST) is assumed to rupture. This could be due to a tornado induced steamline break."

Describe the treatment of the assumption of RWST rupture in the steamline break analyses.

Response: A concrete dike around the RWST ensures sufficient borated water is retained in the event the RWST is ruptured due to a tornado missile or a main steam line break (MSLB). Westinghouse safety analysis has identified the required RWST inventory to accomplish the boration function for reactivity control for the required 10 minute timeframe and for plant cool down following the event. TVA has analyzed the design requirement that the RWST must supply sufficient flow to provide negative reactivity for a period of 10 minutes in the event of a break in the steam line outside containment or missile/tornado damage to the RWST. This calculation demonstrates that the RWST contains an adequate protected volume to accomplish the borated function for reactivity control and for plant cool down following this event.

b. *FSAR 6.3.2.2, "Equipment and Component Design"*

Describe the process (inputs, assumptions, methods) used to determine the 3-hour time, at which hot leg recirculation is started.

Response: **Summary of WBN Unit 2 Completion Project Post-Loss of Coolant Accident (LOCA) Hot Leg Switchover (HLSO) Calculations**

The emergency operating procedure (EOP) requirement to realign the emergency core cooling system (ECCS) to hot recirculation is based on conservative calculations of the buildup of boric acid in the core region for the limiting scenario cold leg break. In this scenario, the vessel is a stagnant boiling pot with only water leaving the vessel in the form of steam. The boric acid buildup calculations use 10 CFR 50.46, Appendix K, decay heat to calculate the core steaming rate. No ECCS flow subcooling is assumed and the exiting steam is assumed to contain no boron.

Post-LOCA boric acid buildup calculations indicate that switchover to hot leg recirculation at 3 hours will limit the maximum core region boric acid concentration to 18.59 weight percent allowing for 8.94 weight percent margin to the atmospheric pressure solubility limit of 27.53 weight percent. The core mass boil-off rates were calculated to be 43.86 lbm/sec (328.7 gpm) for an HLSO time of 3 hours. Hot leg recirculation flows were reviewed and found to be adequate to ensure core cooling and to provide core dilution after HLSO realignment. A more detailed description of the analysis (inputs, assumptions, methods) used to determine the appropriate HLSO time is provided in the response to **RAI 15.3.1 - 1.b**.

c. *FSAR 6.3.3, "Performance Evaluation" (SER 6.3.2 and 6.3.4)*

(1) *FSAR Section 6.3.3.3, "Alternate Analysis Methods," for a steamline break states that "the ECCS adequately fulfills its shutdown reactivity addition function." Specify the shutdown reactivity addition function.*

Response: In the steamline break transient, safety injection is initiated 27.7 seconds into the transient. The accumulators actuate 53.6 seconds into the transient. The boron injected via the two systems turns around the transient such that the peak heat flux occurs at 57.4 seconds with the core returning to a subcritical condition at 58.4 seconds. Note that the safety injection model is very conservative. The analysis assumes zero safety injection

flow until the pumps reach full speed (27 seconds after the safety injection signal on low steam pressure). Also, the minimum RWST boron concentration is 3100 ppm, but only 2000 ppm is assumed in the analysis.

(2) *Also, the FSAR states that "[t]he delivery of the borated water from the charging pump results in a negative reactivity change to counteract the increase in reactivity caused by the system cooldown." Chapter 15 steamline break analysis transient plots show the effect of accumulator injection on core reactivity; but the effect of charging pump flow is not evident. Quantify the negative reactivity insertion of the charging pump flow and how it counteracts the increase in reactivity caused by the system cooldown.*

Response: The Chapter 15 steamline break analysis plots show the effect of the combination of accumulators and safety injection. It is difficult to quantify the effect of either system alone. Safety injection begins 27.7 seconds into the transient, accumulator injection occurs at 53.6 seconds, and the limiting time in the transient occurs at 57.4 seconds. Although safety injection begins at 27.7 seconds, the core does not see any boron immediately due to the assumption that unborated water is in the injection lines and must be purged. Also, the charging pump flow is a function of RCS pressure and varies during the transient. As such, the negative reactivity inserted by the system is difficult to separate out from the total reactivity and cannot be easily quantified.

(3) *FSAR 9.2.7, "Refueling Water Storage Tank [RWST]," discusses the two basic requirements for the RWST. In this regard, one requirement is to provide borated water in the event of a loss-of-coolant accident. However, this requirement does not discuss a steamline break accident response. How are steamline break reactivity requirements met by the RWST? How is the boron concentration requirement for the RWST determined?*

Response: The LOCA is the limiting event in terms of RWST requirements such that the LOCA sets the requirements for the RWST. The steamline break analysis credits the boron from the RWST but does not influence the requirements.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Chapter 15 (taken from NRC letter dated September 20, 2010 (ADAMS Accession No. ML102590244))

Safety Evaluation Report (SER) 15.0.0

15.0.0 – 1. *FSAR 15.0.0, "Accident Analyses"*

a. *FSAR Chapter 15 addresses the accident conditions listed in Table 15-1 of Regulatory Guide (RG) 1.70, which apply to WBN Units 1 and 2. The events listed in RG Table 15-1, include Item 5.2, "Chemical and volume control system [CVCS] malfunction (or operator error) that increases reactor coolant inventory." Discuss why this analysis was not included in FSAR Chapter 15.2.14, as a Condition II event.*

Response: The approach used to address this accident condition for WBN Unit 2 is the same as that previously submitted and accepted for WBN Unit 1 and is described in detail below.

As described in Unit 1 UFSAR Section 15.2.14, the Inadvertent Operation of the ECCS (inadvertent ECCS) event as analyzed by Westinghouse conservatively bounds the response to a CVCS malfunction event. Two cases are analyzed to address the acceptance criteria of interest for the event. In both cases, the charging pumps force concentrated boric acid solution from the RWST, through the common injection header and injection lines, and into the cold leg of each reactor coolant loop. The safety injection pumps also start automatically, but provide no flow when the reactor coolant system is at normal pressure. The passive injection system and the low head system provide no flow at normal reactor coolant system pressure. As a result, the same injection flowrates (i.e., charging flow) as would be expected in a CVCS malfunction event are modeled in the inadvertent ECCS analysis.

During a CVCS malfunction event, the injected flow would be expected to originate in the volume control tank (VCT) rather than the RWST; as a result, the boron concentration of the injected flow would be expected to be lower during a CVCS malfunction event compared to an inadvertent ECCS event. As stated in Unit 1 UFSAR Section 15.2.14.1 relative to the departure from nucleate boiling (DNB) case, "the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in primary coolant temperature and coolant shrinkage. Pressurizer pressure and level drop . . . The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip." As such, without the boron in the injected fluid, there would be no power

mismatch and ultimately no pressure drop. The pressure drop is the only adverse impact with respect to DNB concerns caused by the event. Furthermore, the reactor trip is conservatively modeled to not occur as a result of the spurious safety injection signal, rather from the reactor protection system. The reactor protection system would provide the reactor protection during a CVCS malfunction event. Therefore, the inadvertent ECCS event bounds the CVCS malfunction event relative to DNB concerns.

With respect to pressurizer fill concerns, the boron concentration of the injected flow is of little concern as a reactor trip is assumed to occur as a result of the spurious safety injection signal. As discussed above, relative to DNB concerns, a reactor protection system signal (for example, low pressurizer pressure) would be expected to cause the reactor trip. However, since the event for filling concerns is a timed event (i.e., the event duration is a function of an operator action to gain control of the plant), the delaying of a reactor trip would lead to less limiting results. As a result of the reactor trip, the RCS fluid contracts in response to the sudden loss of heat addition to the system. The fluid contraction causes an outsurge from the pressurizer and a decrease in pressurizer water level. As the reactor trip is delayed, the time available between the reactor trip and the assumed operator action time decreases and therefore so does the likelihood that the pressurizer will fill within this time frame. Furthermore, prior to reactor trip, there is no mismatch between the primary and the secondary so there is no heatup and no thermal expansion. After the reactor and the turbine trip, a heatup occurs due to decay heat and pump heat. As such, the modeling of a reactor trip at event initiation is conservative, and the inadvertent ECCS analysis for pressurizer fill concerns conservatively bounds the pressurizer fill response to a CVCS malfunction event.

15.0.0 – 2. *FSAR 15.2, "Condition II – Faults of Moderate Frequency"*

If in answering Question 1 above, it is determined that another Condition II event, CVCS Malfunction, should be added in FSAR Section 15.2 and in Table 15.2-1, provide the proposed description of this analysis for inclusion in the FSAR.

Response: No additional analysis was required. See the response to **RAI 15.0.0 – 1.a**.

15.0.0 – 3. *FSAR 15.3.2, "Minor Secondary System Pipe Breaks"*

a. *Support the claim that a minor secondary system pipe break would be less limiting than the major steam line rupture, since boric acid is supplied to the core by the accumulators during the major steam line rupture; but not necessarily during the minor secondary system pipe break.*

Response: Per the NRC approved topical report - WCAP-9226-P-A, "the largest double-ended steamline rupture at end of life, hot shutdown conditions with the most reactive RCCA in the fully withdrawn position is a limiting and sufficiently conservative licensing basis to demonstrate that the Westinghouse PWR is in compliance with 10 CFR 100 criteria for Condition II, III, and IV steamline break transients."

b. *What is the largest steam line break size that would not rapidly depressurize the reactor coolant system (RCS) to the accumulator delivery setpoint (i.e., rapidly enough to make accumulator injection the primary mitigation function)?*

Response: All breaks assuming offsite power is available will depressurize to the accumulator injection pressure. The smaller (minor) breaks will reach the injection pressure later than the larger (major) breaks and smaller breaks result in less cooldown and a lower peak heat flux and are subsequently less limiting.

SER 15.2.0, "Normal Operation and Anticipated Transients"

15.2.0 – 1. *FSAR 15.1, "Condition I – Normal Operation and Operational Transients"*

Verify that ORIGEN-S is approved for licensing applications by the NRC.

Response: Oak Ridge National Laboratory published "*Validation of ORIGEN-S Decay Heat Predictions for a LOCA Analysis,*" which validates the ORIGEN-S software. This work was funded by the Nuclear Regulatory Commission Office of Nuclear Regulatory Research and may be obtained from the following website: http://www.ornl.gov/sci/scale/pubs/C183.pdf.

Section 3.1 (Fission Product Inventory) of Regulatory Guide 1.183, "*Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors,*" Revision 2, states the following: "The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18)."

ORIGEN-ARP (Automatic Rapid Processing) uses an algorithm that allows the generation of cross-section libraries for the ORIGEN-S code by interpolation over pre-generated SAS2H cross-section libraries.

15.2.0 – 2. *FSAR 15.1.2, "Initial Power Conditions Assumed in Accident Analysis"*

a. *Provide the safety limit values for the departure from nuclear boiling ratio (DNBR), for typical and thimble cells, used in Revised Thermal Design Procedure (RTDP) and Standard Thermal Design Procedure (STDP) analyses, and bases for their determination.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively, of the response to this RAI.

b. *Specify which analyses are evaluated with RTDP and STDP, and why.*

Response: The RTDP methodology is used for DNB events initiated at power. The RTDP methodology is used to analyze the following events:

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, Partial Loss of Forced Reactor Coolant Flow, Loss of External Electrical Load and/or Turbine Trip, Excessive Heat Removal Due to Feedwater System Malfunctions, Excessive Load Increase, Accidental Depressurization of the Reactor Coolant System, Inadvertent Operation of the ECCS, Complete Loss of Forced Reactor Coolant Flow and Locked Rotor.

All other events analyzed for DNB concerns are analyzed using the STDP methodology. All events analyzed to address concerns other than DNB are analyzed explicitly accounting for all applicable initial condition uncertainties (similar to the STDP approach used to address DNB events) unless otherwise noted. The application of RTDP to the events listed above is consistent with the NRC-approved methodology in WCAP-11397-P-A.

c. *Provide the safety limit value for linear power density (kW/ft).*

Response: As presented in Unit 2 FSAR Section 4.3.2.2.5, the safety limit value for the linear power density is 22.4 kW/ft.

15.2.0 – 3. *FSAR 15.1.3, "Trip Points and Time Delays Assumed in Accident Analysis"*

Verify that the time delay, imposed by the trip time delay system, is designed to fail short, not long.

Response: The Trip Time Delay (TTD) described in Unit 2 FSAR Section 15.1.3 is implemented in Eagle 21 which is a digital system. TTD logic and time delays are developed in software. Design failure modes associated with hardware devices are not applicable to software functions.

The WBN Unit 1 TTD modification is described in Westinghouse WCAP 13462, "Summary Report - Process Protection System Eagle 21 Upgrade, NSLB, MSS and TTD Implementation - WBN Units 1 and 2." The NRC reviewed this report in WBN Unit 1 SER NUREG-0847, Supplement 14, and concluded that the implementation of TTD is acceptable as part of the WBN Unit 1 initial license. The Unit 2 TTD design is the same as Unit 1.

<u>SER 15.2.1, "Loss of Cooling Transients"</u>

15.2.1 – 1. *FSAR 15.2.5, "Partial Loss of Forced Coolant Flow"*

a. *The FSAR states that "a partial loss of coolant flow accident can occur from a mechanical or electrical failure in a reactor coolant pump or from a fault in the power supply to the pump or pumps supplied by a reactor coolant bus." The FSAR presents the results of an analysis of loss of one pump.*

If a single fault can cause the loss of two pumps, and the loss of two pumps is expected to be more limiting than the loss of one pump, the FSAR should present an analysis or evaluation of the loss of two pumps. FSAR Chapter 15.3.4 states that the pumps are supplied from individual buses. Verify that the loss of one pump is the only flow-related consequence of a single fault (e.g., one bus failure).

Response: Under normal alignment, each reactor coolant pump (RCP) bus is individually supplied from a normal and an alternate power source. Each of the bus transfer circuits is electrically independent. Therefore, a single bus failure or loss of a single supply would not result in loss of power to more than one RCP.

15.2.1 – 2. *15.2.8 Loss of Normal Feedwater*
(Note: The plots are barely legible. Supply better quality plots for this transient.)

Response: Better quality plots are provided in Attachment 4.

a. *The FSAR states, "Additional sensitivities were performed to determine if it was more conservative to model the pressurizer power operated relief valves [PORVs] as operable or inoperable." What are the results of these studies, and how are they applied to the analysis of this event?*

Response: The additional cases analyzed demonstrated that the modeling of the power operated relief valves as operable led to the overall most limiting maximum pressurizer water volume. The results, figures and tables related to the Unit 2 UFSAR section correspond to this limiting case. The case with the PORVs operable resulted in a peak pressurizer volume of 1472 ft^3 and the case with inoperable PORVs resulted in a peak pressurizer volume of 1467 ft^3.

b. *What information is expected to be gained by analyzing a separate case, with added charging flow, that is not available from the results of the inadvertent actuation of emergency core cooling system (ECCS) event?*

Response: The analysis of the loss of offsite power with charging flow case was included to address a specific failure (described in the response to **RAI 15.2.1 – 2.c.**). It was not clear that the consequential failure in addition to the initiating event of the loss of feedwater flow was bounded by the analysis presented for the inadvertent ECCS event, so an explicit analysis was performed.

c. *What is the control or protection system logic, and objective, that demands charging flow on a loss of offsite power signal?*

Response: As described in Westinghouse Nuclear Safety Advisory Letter NSAL-00-13 (copy provided in Attachment 8), a potential consequence of a loss of offsite power is a loss of power supply to the instrument air system, which may eventually cause letdown isolation valves to close and charging isolation valves to open. Therefore, if the power supply of the charging pumps is sequenced onto the diesel generator and charging and/or seal injection is initiated, there could be a net addition to the reactor coolant system water mass, and thus an increased potential for filling the pressurizer.

d. *In the loss of normal feedwater (a loss of heat sink event), pressurizer level rises due to coolant swell, as RCS temperature rises. What is the net effect of adding relatively cool water to the RCS via the charging system, and then shutting off this flow, on the peak pressurizer water level?*

Response: See the response to **RAI 15.2.1 – 2.e**.

e. *The second peak in pressurizer level occurs at 5.5 minutes. Does the case with charging flow produce another, later peak? If so, what is the level and time of this peak?*

Response: The addition of charging flow would provide a benefit regarding primary side temperature increase (post-trip heatup) but a penalty regarding inventory available for expansion. The net effect of the charging flow in the limiting case is a second peak which occurs approximately 16 minutes after event initiation with approximately 16 ft^3 of margin available to pressurizer fill. Note that as stated in the Unit 2 FSAR, the cases modeling the addition of charging flow are not appropriate to demonstrate the heat removal capacity of the auxiliary feedwater system, which is the intent of the event, and therefore are not the basis for the results, figures and tables presented for the Unit 2 FSAR section.

15.2.1 – 3. *FSAR 15.2.9, "Coincident Loss of Onsite And External (Offsite) AC Power to the Station - Loss of Offsite Power to the Station Auxiliaries"*

Discuss the establishment of natural circulation flow, such that all decay heat is removed in this event.

Response: The limiting case presented in Unit 2 FSAR Section 15.2.8 for the loss of normal feedwater event models a loss of offsite power. This limiting case, which conservatively bounds the response for the Loss of Offsite Power to the Station Auxiliaries event described in Unit 2 FSAR Section 15.2.9, demonstrates that all decay heat is removed while at natural circulation conditions. Figures 15.2-27a through 15.2-27i demonstrate the transient response including the establishment of natural circulation for the more limiting event (Figures 15.2-27c and 15.2-27d).

15.2.1 – 4. *FSAR 15.3.4, "Complete Loss of Forced Reactor Coolant Flow"*

a. *What is the DNBR safety limit value and what is the thermal design procedure used?*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively of the response to this RAI.

b. *What are the minimum DNBR values calculated for the undervoltage and underfrequency cases?*

Response: A comparison of the transient conditions of power, flow, pressure, and temperature (statepoints) was performed which concluded that the underfrequency case is more limiting with respect to DNB. A calculated DNBR is only available for the limiting case. The DNBR for the reference underfrequency loss of flow case is >1.9. Confirmation that the minimum DNBR for this event meets the DNB design criterion for future reloads will be performed for all cycles in accordance with WCAP-9272-P-A (Proprietary) / WCAP-9273-A (Non-Proprietary), "Westinghouse Reload Safety Evaluation Methodology," July 1985.

SER 15.2.2, "Increased Cooling Transients"

15.2.2 – 1. *FSAR 15.2.10, "Excessive Heat Removal Due To Feedwater System Malfunctions"*

a. *The FSAR states, "A generic study performed by Westinghouse demonstrated that the consequences of a hot zero power feedwater malfunction with an increased feedwater flow rate of less that [sic] 150% of the nominal full power flow rate are non-limiting and are bounded by the hot full power feedwater malfunction." Provide a copy (or reference, if previously submitted) of this generic study, and verify that it's applicable to the WBN plant design.*

Response: This section is identical to that previously accepted for WBN Unit 1 and currently documented in Section 15.2.10 of the WBN Unit 1 UFSAR. The study in question concluded that due to the initial plant conditions at zero power and the presence of the high neutron flux (low setting) reactor trip (~35% RTP), the peak heat flux achieved in a conservatively modeled feedwater malfunction event for flow rates ≤150% will not be high enough to make the scenario limiting compared to the full power case. This study and the associated conclusion are applicable to WBN because it is not based on any plant-specific protection settings or plant configuration. The amount of cooldown assuming that flow rate is simply insufficient to make the scenario limiting due to the initial conditions associated with zero power and the availability of the high flux reactor trip to limit the power increase. The generic study is contained in an internal Westinghouse proprietary document. It can be made available at the Westinghouse site at the NRC's request.

b. *It appears, from Figure 15.2-28d that the pressurizer PORVs are not modeled in the analyses. Explain why the PORVs would not be modeled in analyses that are designed to minimize DNBR.*

Response: The pressurizer PORVs are not modeled in the analyses of the excessive heat removal due to feedwater system malfunction events as the pressurizer pressure is not anticipated to increase during the event up until the point at which turbine trip occurs, which is beyond the limiting time in the event. The timing of the transient is such that the minimum DNBR occurs while the reactor is operating at full power and the pressurizer pressure is below the PORV opening setpoint.

15.2.2 – 2. *FSAR 15.2.11, "Excessive Load Increase Incident"*

Specify the low limiting value for DNBR.

Response: This event is typically non-limiting – no reactor trip is generated and the DNB design basis is seldom challenged. As such, the same analysis methodology previously used and approved for WBN Unit 1 was used for WBN Unit 2. This method applies conservatively bounding conditions in generating statepoints that are compared directly to the WBN Unit 2 core thermal limits. If the statepoint conditions remain above the conditions where the DNBR would equal the safety analysis DNBR limit, no further analysis is required. This was the case for WBN Unit 2, and therefore a minimum DNBR was not calculated. A summary of the method follows.

Bounding initial conditions for plant parameters which impact DNBR conditions (i.e., power, temperature, pressure and flow) were determined to be applicable for the WBN Unit 2 Completion Program consistent with the Revised Thermal Design Procedure (RTDP) DNB methods employed for WBN Unit 2. The initial conditions were the licensed core power, nominal T_{avg} temperature, nominal RCS pressure and minimum measured flow, consistent with the RTDP DNB methods.

Conservatively bounding deviations in plant parameters are applied to the WBN Unit 2 initial conditions. The deviations are derived from a bounding set of plant analysis results with appropriate conservatisms applied. By applying these deviations to the WBN Unit 2 initial conditions, a conservative set of statepoints are generated for each case examined.

The combined WBN Unit 2 initial conditions and bounding deviations (i.e., statepoints) were compared directly to the WBN Unit 2 core thermal limit lines that represent the locus of conditions when the DNBR is equal to the DNBR limit value for the uprate. The comparison showed that margin between the bounding statepoint

conditions and core thermal limits exists which demonstrates that the minimum DNBR conditions associated with an Excessive Load Increase Incident for WBN Unit 2 conditions meet the safety analysis DNBR limit. As a result, a calculated limiting DNBR value is not available.

15.2.2 – 3. *FSAR 15.2.13, "Accidental Depressurization of the Main Steam System"*

a. *The FSAR states, "The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief or safety valve. The analyses performed assuming a rupture of a main steam line is given in Section 15.4.2.1." Why is the reader referred to analyses of a main steam line rupture when the transient of interest is an accidental depressurization of the main steam system due to an inadvertent opening of a single steam dump, relief or safety valve?*

Response: The Main Steam System Depressurization is an ANS Condition II event, and the Main Steam Line Rupture is an ANS Condition III or IV event. Westinghouse analyzes the Main Steam Line Rupture event to the more conservative Condition II acceptance criteria. When analyzed to the same acceptance criteria, the Main Steam System Depressurization is bounded by the Main Steam Line Rupture presented in Section 15.4.2.1. When the same acceptance criteria are used, the larger break size is always more limiting.

b. *Provide a discussion to support the claim that the main steam line rupture is more limiting than the accidental depressurization of the main steam system given that boric acid is supplied to the core by the accumulators in the former case; but not in the latter case.*

Response: All breaks assuming offsite power is available will depressurize to the accumulator injection pressure. The accidental depressurizations will reach the injection pressure later than the main steam line rupture, and smaller breaks result in less cooldown and a lower peak heat flux and are subsequently less limiting.

c. *In FSAR Section 5.2.1.5, it is stated that, "A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient." If heat generation is prevented, the implication is that criticality is also prevented. There are no analyses to support this conclusion. Show that the maximum return to power for the accidental depressurization of the main steam system is much lower than that for the main steam line rupture.*

Response: Section 5.2.1.5 refers to the NSSS Design Transients, which are used for the RCS component fatigue stress analyses and evaluations. These transients are defined by the American Society of Mechanical Engineers (ASME) Section III code. The assumptions made for the Small Steam Line (Emergency Condition or ASME Level C) and Large Steam Line Break (Faulted condition or ASME Level D) transients maximize the primary side pressure and temperature variations, which is conservative for the component stress analyses. The assumptions used for the Large Steam Line Break transient in Section 5.2.1.5 are different from the transients discussed in 15.4.2 and 15.2.13 since the analyses are performed for different purposes and have different acceptance criteria. For design transients, the steam line break transient is a cooldown transient. A large shutdown margin with no feedback and no decay heat maximizes the RCS cooldown during this transient. A return to power is not considered in the NSSS Design Transient analyses and the results are included in the appropriate component specifications.

d. *Explain how minimum DNBR would not be a concern, before trip, based on the post-trip return to power.*

Response: The post-trip return to power has no influence on the pre-trip DNBR. Consistent with the licensing basis for WBN Unit 1, only the zero power steamline break event is considered for WBN Unit 2. The assumption made (and accepted by the NRC) when Unit 1 was licensed is that the OPΔT function with the setpoints generated consistent with WCAP-8745-P-A will provide a reactor trip prior to a violation of the DNB design basis. As such, only the post-trip portion of the transient needs to be explicitly evaluated. The post-trip portion of the transient is bounded by the hot zero power steamline break analysis presented in the Unit 2 FSAR in Section 15.4.2.

e. *This FSAR section states that "a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves." FSAR Section 15.4.2 states, "A safety injection signal will rapidly close all feedwater control valves and main feedwater isolation valves, and trip the main feedwater pumps, condensate booster pumps, condensate demineralizer pump, and motor-operated standby feedwater pump if operating." Verify that these functions are the same.*

Response: Feedwater isolation actuated by a safety injection signal has the same functions in both sections and is only described in different detail. In other words, the same actions are attempted for every feedwater isolation actuated by a safety injection signal regardless of the event.

<u>SER 15.2.2, "Change in Inventory Transients"</u>

15.2.3 - 1. *FSAR 15.2.12, "Accidental Depressurization of the Reactor Coolant System"*

a. *The FSAR states, "The most severe core conditions resulting from an accidental depressurization of the reactor coolant system are associated with an inadvertent opening of a pressurizer safety valve." What would cause the inadvertently opening of a pressurizer safety valve?*

Response: The opening of a pressurizer safety valve is not a credible event; rather, it is used to conservatively bound all accidental depressurization of the reactor coolant system events (for example, an inadvertent opening of a pressurizer power operated relief valve).

b. *The FSAR states, "The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip." Why does the pressurizer level increase?*

Response: The increase in pressurizer water level is a result of coolant expansion enabled by a decrease in pressurizer pressure, a direct effect of the stuck open safety valve.

c. *Specify the low limiting value for DNBR.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively, of the response to this RAI.

15.2.3 - 2. *FSAR 15.2.14, "Inadvertent Operation of Emergency Core Cooling System"*

a. *Two different courses of events are considered, since "it cannot be assumed that any single fault that actuates a safety injection signal will also produce a reactor trip." A safety injection signal will generate a reactor trip signal. What scenario is postulated that would prevent the generation of a reactor trip signal from the spurious safety injection signal?*

Response: As described in the response to **RAI 15.0.0 - 1**, the modeling of a reactor trip based on a reactor protection system setpoint rather than as a result of the safety injection signal is conservative for DNB concerns. Consistent with the licensing basis for WBN Unit 1, this bounding scenario is assumed for the analysis of this event for DNB.

b. *The inadvertent ECCS actuation at power event is analyzed to determine the minimum DNBR value. By what mechanism could the actuation of ECCS, including the resultant reactor trip, lead to a degradation of thermal margin? Provide an analysis or evaluation for such a scenario.*

Response: As described above, a scenario that does not result in the generation of a reactor trip signal from the spurious safety injection signal is conservatively modeled. In such a scenario, the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in primary coolant temperature and coolant shrinkage. Pressurizer pressure and level drop. In theory, the decrease in pressure could lead to degradation in thermal margin. However, as described in the *Results* presented in Unit 2 FSAR Section 15.2.14.2, the net effect of the power mismatch and the related RCS implications is an increase in the thermal margin (i.e., the DNBR increases) throughout the transient. The transient associated with an inadvertent safety injection signal that results in an immediate reactor trip would be a less limiting event with respect to DNB concerns as a result of the decrease in power associated with the reactor trip.

c. *The FSAR states, "Should water relief through the pressurizer power-operated relief valves (PORVs) occur, the PORV block valves would be available, following the transient, to isolate the RCS."*

i. *What are the consequences of several minutes of water relief through two PORVs, until the PORVs can be manually isolated?*

Response: The Pressurizer Relief Tank (PRT) volume and the quantity of water stored in the tank are such that no steam or water will be released to containment under

any normal operating conditions or anticipated abnormal occurrences.

The volume of water in the PRT is capable of absorbing a discharge of 110% of the pressurizer steam-water volume above the full power water level setpoint. The steam-water volume requirement is approximately the amount discharged from the pressurizer safety and relief valves if the plant were to suffer a complete loss of load followed by a turbine trip.

PRT water temperature is indicated in the MCR by a temperature detector and associated control loop. A high temperature alarm in the MCR informs the operator that cooling of the tank contents is required. Also, the PRT contains a level indicating control loop to provide a signal to a MCR level indicator. High or low water level activates an alarm.

The water relief through two PORVs for several minutes (classified as abnormal occurrence) has been evaluated by Westinghouse (Mass and Energy Releases for Inadvertent Safety Injection). Based on the mass and energy releases of this analysis, with two PORVs stuck open, the PRT is capable of absorbing the steam-water discharge for a period of time greater than 10 minutes without rupturing the "rupture disks".

The failure of the two PORV(s) to close will be detected by the operator, who will isolate the PORV(s) by closing the block valve(s) located in series with the PORV(s). This action by the operator will terminate the abnormal occurrence.

ii. *Are the PORV block valves qualified to close against the flow of solid water?*

Response: The PORV block valves or isolation valves (MOVs) (2-FCV-68-332 and 2-FCV-68-333) are located between the pressurizer and each PORV. The PORV block valves or isolation valves are qualified to be operated (i.e., opened and closed) for all fluid conditions expected under normal operating and accident conditions. Also, they are qualified to close against the flow of solid water.

Part of the NRC Generic Letter (GL) 89-10 MOV program involves the documentation of the MOV design basis review and assessment of the valve and actuator capability and determination of required trust/torque.

Evidence that these block valves or isolation valves can be opened and closed for all fluid conditions (including solid water) expected under normal operating and accident conditions, is documented as part of the GL 89-10 MOV program.

Should water relief through the PORVs occur, the PORV isolation valves are qualified to perform their design basis function, and are available to isolate the RCS.

iii. *What alternative actions are available, to the operator, if one or both block valves cannot be closed?*

Response: If both a PORV and associated block valve are open, the reactor will experience a loss of reactor coolant. The possibility that a PORV and block valve will not close is considered in the generic Westinghouse Emergency Operating Procedures which are implemented in the WBN Emergency Operating Instructions (EOIs). Emergency Operating Instruction E-0 (Reactor Trip or Safety Injection) contains steps to transition to EOI E-1 (Loss of Reactor or Secondary Coolant) if a pressurizer PORV and block valve are determined to be open. E-1 contains steps to mitigate a Loss of Reactor Coolant by monitoring containment and reactor conditions and initiating actions such as attempting to close the block valve, restoring pressurizer level, etc.

iv. *Show that closing the PORV block valves would not lead to opening the safety valves.*

Response: Based on Emergency Operating Instructions E-0 (Reactor Trip or Safety Injection) and ES-1.1 (SI - Termination), the appropriate recovery following a Safety Injection Initiation is for the Operator to confirm that the pressurizer pressure is less than a setpoint value and then to ensure that the Pressurizer (PZR) PORVs or associated Block Valves (Isolation Valves) are closed.

Per ES-1.1, the plant operator performs the following to terminate Safety Injection and stabilize plant conditions:

- checks RCS subcooling and the PZR water level;
- ensures RCS pressure is stable; and
- controls the charging flow and letdown to maintain PZR water level and RCS pressure.

Since the RCS is stable and is not being further pressurized, closing the PORV block valves would not

lead to opening the safety valves which open at a higher pressure than the PORVs.

SER 15.2.4, "Reactivity and Power Distribution Anomalies

15.2.4 – 1. *FSAR 15.2.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power"*

Correct the titles for the X axes in Figures 15.2-8, 15.2-9, and 15.2-10.

Response: A future amendment to the Unit 2 FSAR will provide corrected titles for the X axes for the cited figure as follows:

- Figure 15.2-8: replace "TIME (SECONDS)" with "REACTIVITY INSERTION RATE (pcm/sec)."
- Figure 15.2-9: replace "TIME (SECONDS)" with "REACTIVITY INSERTION RATE (pcm/sec)."
- Figure 15.2-10: replace "TIME (SECONDS)" with "REACTIVITY INSERTION RATE (pcm/sec)."

Corrected figures are provided in Attachment 4.

15.2.4 – 2. *FSAR 15.2.3, "Rod Cluster Control Assembly Misalignment"*

Provide a legible version of Figure 15.2-11.

Response: A future amendment to the Unit 2 FSAR will provide a legible version of Figure 15.2-11. A legible copy of the figure is provided in Attachment 4.

SER 15.3.1, "Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident) [LOCA]

FSAR 15.3.1, "Loss of Reactor Coolant from Small Ruptured Pipes or Cracks in Large Pipes which Actuate the Emergency Core Cooling System"

FSAR 15.4.1, "Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)"

15.3.1 - 1. *Post-Loss-of-Coolant (LOCA) Boric Acid Precipitation*

a. *Provide the analysis results and write-up for the timing for boric acid precipitation supporting the emergency operating procedure (EOP) input for timing to switch to the simultaneous injection mode following a LOCA.*

Response: The response to this RAI is provided in Enclosure 4.

b. *What is the sump temperature versus time following recirculation, and how does this impact precipitation? Provide an explanation discussing whether the boric acid concentration in the vessel is below the precipitation limit based on the minimum sump temperature at the time the switch to simultaneous injection is performed.*

Response: Sump temperature transients biased for low fluid temperature were not generated for the post-LOCA boric acid precipitation analysis. FSAR Figure 6.2.1-3 does provide active and inactive sump temperature transients, but note that these transients are not biased low and thus do not necessarily address the condition implied in the RAI.

As described in the response to **RAI 15.3.1 - 1**, the WBN Unit 2 post-LOCA boric acid precipitation uses a solubility limit criteria that is conservative for the expected core conditions, that is, saturated conditions at atmospheric pressure. The HLSO time of 3 hours does, however, provide margin to this solubility limit criteria. At HLSO time, the calculated core region boric acid concentration is 18.59 weight percent. This boric acid concentration corresponds to a saturated solution at 173°F.

Although choosing a solubility limit based on saturated conditions may appear to be nonconservative in light of the implications inferred in the RAI, this practice is common to all US PWR boric acid precipitation control analyses. It is also one of the many methodology issues being addressed by a PWR Owner's Group program to develop a post-LOCA boric acid precipitation evaluation model.

The NRC staff has been made aware of this program, the most recent formal update being in October 2009 (NRC Memorandum for Stacey L. Rosenberg from Jonathan Rowley, "Forthcoming Meeting With Pressurized Water Reactor Owners Group (PWROG), October 15, 2009" (ML092790215)).

As discussed in a previous RAI response to the staff for another licensee LAR (L-05-112, Letter from First Energy Nuclear Operating Company to USNRC, "Beaver Valley Power Station Unit Nos. 1 and 2, BV-1 Docket No. 50-334, License NO. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, Responses to a Request for Additional Information in Support of License Amendment Request Nos. 302 and 173," July 8, 2005 (ML051940575)), the injection of subcooled ECCS flow into the reactor vessel would have two effects on the potential for boric acid precipitation. The subcooled ECCS flow would reduce the temperature of the liquid in the region in

which it interacts thus reducing the localized boric acid solubility limit. However, the subcooled SI flow, being at a relatively low boric acid concentration, would dilute the boric acid solution with which it interacts. The net effect can be assessed by calculating the effect of the ECCS flow on the change in localized fluid temperature and boric acid solubility limit versus the change in localized boric acid concentration. Any condensation of steam would be an additional dilution mechanism.

For example, consider conditions calculated for WBN Unit 2 at HLSO time, where the vessel mixture solution is at 218°F with a boric acid concentration of 18.59 weight percent and the injected ECCS flow is at 150°F with a boric acid concentration of 2.0 weight percent. In this example, steam condensation is not credited. Ignoring the density differences, a unit volume of injected ECCS flow mixing with a unit volume of vessel mixture would decrease the liquid temperature to 184°F (218 – (218 + 150) / 2). The boric acid solubility limit at 184°F is approximately 21.0 weight percent. The same ECCS flow unit volume would decrease the boric acid concentration of a unit volume of core region fluid to 10.30 weight percent (18.59 weight percent + 2.0 weight percent) / 2). Thus the boric acid concentration of the combined unit volumes would have 10.7 weight percent (21.0 weight percent - 10.30 weight percent) margin to the solubility limit.

15.3.1 - 2. *Small-Break LOCA (SBLOCA)*

a. *Provide the refueling water storage tank maximum temperature used in the SBLOCA analysis.*

Response: The refueling water storage tank maximum temperature used in the SBLOCA analysis is 105°F.

b. *Provide the head curves for all pumped injection used in the SBLOCA analyses. Confirm that these curves included the error on head and flow rate in generating these curves.*

Response: The flow curves are provided in Figures 15.3-2a and 15.3-2b in the Unit 2 FSAR. Figure 15.3-2a provides the minimum safeguards total flows assumed for the break sizes less than the accumulator line (8.75-inches) where the faulted loop is assumed to spill to RCS pressure. Figure 15.3-2b provides the minimum safeguards total SI flows assumed for accumulator line break.

Table 1 (page E1-30) provides the Spill to RCS Pressure flows per pump and corresponds to Unit 2 FSAR Figure 15.3-2a. Table 2 (page E1-31) provides the Spill to Containment Pressure (0 psig) flows per pump and corresponds to Unit 2 FSAR Figure 15.3-2b. The NOTRUMP model lumps the three intact loops into a single loop; therefore, the intact loop flows represent the total flow that is available for injection into the three intact loops. It has been confirmed that these curves account for the potential measurement error on pump head and flow rate.

TABLE 1
Minimum Safeguards ECCS Flows
Faulted Loop is Assumed to Spill to RCS Pressure

Pressure (psia)	Intact Loop		Faulted Loop	
	CHG (gpm)	SIP (gpm)	CHG (gpm)	SIP (gpm)
14.7	262.2	416.0	107.1	148.5
114.7	255.8	401.4	104.5	143.4
214.7	249.4	386.2	101.9	138.0
314.7	242.9	370.0	99.2	132.2
414.7	236.4	353.0	96.5	126.1
514.7	229.5	335.1	93.8	119.7
614.7	222.2	316.3	91.0	113.0
714.7	215.0	296.3	88.1	105.9
814.7	207.7	274.9	85.1	98.2
914.7	200.2	251.8	82.0	90.0
1014.7	192.5	226.5	78.9	80.9
1114.7	184.7	198.0	75.8	70.7
1214.7	176.7	164.9	72.5	58.9
1314.7	168.5	123.6	69.3	44.2
1414.7	159.6	60.2	65.9	21.5
1514.7	150.6	0.0	62.3	0.0
1614.7	141.3	0.0	58.6	0.0
1714.7	131.4	0.0	54.9	0.0
1814.7	121.0	0.0	50.2	0.0
1914.7	110.4	0.0	46.7	0.0

TABLE 2

Minimum Safeguards ECCS Flows

Faulted Loop is Assumed to Spill to Containment Pressure (0 psig)

Pressure (psia)	Intact Loop		Faulted Loop		
	CHG (gpm)	SIP (gpm)	CHG (gpm)	SIP (gpm)	LHSI (gpm)
14.7	262.2	417.0	97.4	149.5	1123.0
114.7	253.6	400.7	100.3	151.1	2031.0
214.7	244.9	381.4	103.2	153.6	2224.0
314.7	236.1	360.9	106.0	156.7	2223.0
414.7	227.1	339.5	108.9	159.9	2222.0
514.7	218.0	317.6	111.7	163.2	2221.0
614.7	208.4	294.7	114.4	166.6	2221.0
714.7	198.6	269.8	117.2	169.9	2220.0
814.7	188.5	242.7	119.9	173.2	2219.0
914.7	178.2	214.5	122.6	176.9	2218.0
1014.7	167.6	185.1	125.3	180.8	2218.0
1114.7	156.7	150.2	128.1	198.0	2217.0
1214.7	145.4	105.9	130.8	202.9	2216.0
1314.7	131.0	50.8	162.0	208.6	2215.0
1364.7	123.5	6.5	164.0	210.5	2215.0
1389.7	119.7	0.0	165.0	211.5	2214.0
1414.7	115.9	0.0	166.0	212.4	2214.0
1514.7	100.1	0.0	170.2	212.4	2214.0
1614.7	83.0	0.0	174.3	212.4	2214.0
1714.7	63.7	0.0	178.6	212.4	2214.0
1814.7	44.4	0.0	182.8	212.4	2214.0
1914.7	27.0	0.0	186.7	212.4	2214.0
2014.7	5.8	0.0	191.0	212.4	2214.0
2114.7	0.0	0.0	194.9	212.4	2214.

c. *Confirm that the hot leg nozzle gap and core barrel alignment key leakage paths were not credited in the SBLOCA analyses.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively, of the response to this RAI.

d. *Provide the results of SBLOCA from a severed injection line. Also, provide the degraded head curves for each injection location. Since the broken safety injection line dumps to containment at very low pressure, the injection into the other intact lines will be degraded under these conditions.*

Response: The 8.75-inch break size presented in Section 15.3.1 of the Unit 2 FSAR is a severed accumulator line. In this case, no credit is taken for the accumulator flow in the broken loop, and all broken loop SI flow is assumed to spill to the containment. The minimum safeguards ECCS flows per pump generated for this case are provided in Table 2 (page E1-33) and reflect the spill to containment spilling assumption. Note for this case, low head safety injection (LHSI) flow is modeled as spilling flow only. No credit for LHSI flow is taken for injection into the intact loop.

e. *Explain why the SBLOCA analysis assumed the accumulator liquid volume of 1050 ft^3 compare to the minimum value of 1000 ft^3 in the large-break LOCA analysis. Also, confirm the potential negligible impact on the SBLOCA peak clad temperature for this difference in liquid volume.*

Response: For standard application of the approved Westinghouse SBLOCA Emergency Core Cooling System (ECCS) Evaluation Model (NOTRUMP-EM, References 1, 2, and 3), a nominal accumulator liquid volume is assumed. (It is noted that the nominal value assumed in the WBN Unit 2 SBLOCA analysis corresponds to the average accumulator liquid volume of the range presented in Table 15.4-19 of the Unit 2 FSAR for the LBLOCA analysis.)

SBLOCA assumes a minimum gas cover pressure minus uncertainties as given in Table 15.3-2 of the Unit 2 FSAR. Once the RCS pressure reaches the accumulator pressure setpoint and the accumulators begin injecting, the mixture level in the core will begin to recover, if not already recovering on SI flow, and result in the cladding temperatures beginning to decrease. Therefore, the key accumulator parameter is the gas cover pressure, and a change to the assumed liquid volume would not impact the overall PCT results. In addition, unlike LBLOCA, the limiting transients in SBLOCA only use a fraction of the total accumulator volume to mitigate the heat-up phase of the transient. Based on this, the use of 1050 ft^3 is

considered acceptable, and no further evaluation of the minimum value of 1005 ft^3 is necessary.

f. *Confirm/verify that the operating plant ranges given in FSAR Table 15.4-19 are consistent with the Technical Specification limit on these parameters.*

Response: The values provided in Table 15.4-19 are direct inputs to the Large Break LOCA analysis and are consistent with the Technical Specifications.

g. *Identify the loop seal piping locations that clear of liquid for the breaks in the SBLOCA analyses.*

Response: See Enclosures 2 and 3 for the proprietary and non-proprietary versions, respectively, of the response to this RAI.

h. *Discuss the results of a failure of a single bottom mounted instrument tube in the lower head of the reactor vessel*

Response: No plant specific analyses were performed for the WBN Unit 2 Completion Project with regard to the failure of a single bottom mounted instrument (BMI) tube in the lower head of the reactor vessel. However, a comprehensive Westinghouse Owner's Group (WOG; now the Pressurized Water Reactor Owner's Group (PWROG)) program was developed several years ago to assess the impacts of a postulated leak or failure of one or more BMI nozzles.

The WOG program included the following tasks.

- Historical information review to determine the extent to which BMI breaks have been analyzed and to determine the effort required to address the potential consequence of a BMI failure.
- Small Break LOCA analyses to evaluate the potential effect of various failures of BMI tubes. These results are then utilized to support a probabilistic risk assessment of BMI failures.
- A materials assessment which evaluates the potential for failure based on phenomenological considerations. This includes Failure Modes and Effects Analysis (FMEA).
- Evaluation of the effectiveness of the Emergency Response Guidelines (ERGs) in dealing with this postulated scenarios and provisions for recommending modifications to the guidance.

During the execution of this program, various organizations discussed the benefits of providing a coordinated fleet-wide response to BMI related issues. As such, a joint effort between the WOG, B&W Owners Group (BWOG) and MRP (Materials Reliability Program (EPRI)) was developed to provide this response. The effort culminated in the development of internal documentation which supports the various conclusions reached in regards to these issues. A meeting to present the WOG and BWOG results to the NRC was held on September 30 of 2005. A summary of the observed LOCA response is provided below:

- Different plant groups demonstrate similar responses to the BMI small LOCA event. Evaluated thermal hydraulic analysis cases representative of WBN Unit 2 show that a BMI tube break of approximately 1.25 inches equivalent diameter can be withstood under timely operator action (45 minutes) to depressurize without core uncovery.

15.3.1 – 3. *LBLOCA*

Discuss the results of the LBLOCA analyses addressing downcomer boiling. Describe the limiting single failure for this condition.

Response: The limiting case as presented in Unit 2 FSAR Section 15.4 is the basis for this response. The initial blowdown decreases liquid inventory in the reactor coolant system (RCS) to a low value as indicated in Figure 1 of Attachment 5 by the downcomer collapsed liquid level and in Figure 3 of Attachment 5 by the lower plenum collapsed liquid level. Until approximately 23 seconds, any liquid entering the downcomer is bypassed around the downcomer to the break. At the end of bypass, both the downcomer and lower plenum are calculated to begin to refill. At approximately 45 seconds, the lower plenum is calculated to be full and bottom of core recovery is achieved. The calculated ongoing near adiabatic heatup of the hot rod is terminated as the reflood period begins.

At approximately 100 seconds, the calculated downcomer liquid temperature as shown in Figure 1 of Attachment 5 rises to the saturation temperature indicating the start of downcomer boiling. The downcomer, core and lower plenum collapsed liquid levels begin to decrease as shown in Figures 1, 2 and 3 of Attachment 5. The core begins a second heatup until the downcomer, core and lower plenum levels begin to again recover as downcomer boiling abates. Once the core collapsed liquid level reaches approximately 4 ft, the core is predicted to quench.

The limiting single failure for this condition is consistent with that presented in Unit 2 FSAR Table 15.4-19 and in Section 12-3-4 of the ASTRUM Topical ("Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", WCAP-16009-P-A, January 2005 (Westinghouse Proprietary)). As stated in this document:

> "Single-Failure Assumption: The loss of a safety train (that is, the loss of a lowhead pump and a high-head pump) will be assumed for the determination of pumped ECCS flow during the LOCA, while the train is assumed to operate in the calculation of containment backpressure. This will result in a conservatively low containment backpressure. In other plant analyses, sensitivity studies may be performed to permit a less conservative set of assumptions."

Thus, two trains of spray are assumed to be available in the containment backpressure calculation. This assumption leads to the flows listed in Table 15.4-23 and the containment backpressure presented in Figure 15.4.40b.

SER 15.3.2, "Major Secondary System Pipe Rupture: Steamline Break"

15.3.2 - 1. *FSAR 15.4.2.1, "Major Secondary System Pipe Rupture: Steamline Break"*

a. *Specify the NRC-approved DNBR correlation that is used for the steam line break departure from nucleate boiling (DNB) evaluation.*

Response: Consistent with the licensing basis for WBN Unit 1, the Westinghouse W-3 DNB correlation is used (see Unit 2 FSAR Section 4.4.2 for additional discussion).

b. *Provide steam flow plots that depict the steam flow at the time of the break, and at the time of steam line isolation, for the faulted and intact loops.*

Response: Faulted loop steam flow is given in Figure 15.4.11c of the Unit 2 FSAR for Case a (offsite power available) and in Figure 15.4.12c of the Unit 2 FSAR for Case b (offsite power lost). Intact loop steam flow is nearly identical up until the time of steamline isolation occurs at 8.7 seconds. Also, see response to **RAI 15.3.2 - 1.i** which includes transient plots for 600 seconds.

c. *Why is the k_{eff} versus temperature relationship considered at 1110 psi, and not at another pressure (e.g., 1000 psia)?*

Response: The K_{eff} curve is generated at 1110 psia for illustration. It could be generated at any pressure desired from the reactivity feedback coefficients assumed in the steamline break analysis. The pressure of 1110 psia was selected when Unit 1 was originally licensed and that same pressure has been maintained since.

d. *In the discussion of the systems that provide the necessary protection against an accidental depressurization of the main steam system, the FSAR lists "[t]he overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal." Does the reactor trip occur in conjunction with receipt of the safety injection signal, or as a result of the receipt of the safety injection signal?*

Response: The reactor trip occurs as a result of the receipt of a safety injection signal. The safety injection system is discussed in detail in Unit 2 FSAR Section 7.3, and the logic diagram for the Safety Injection System is given in the FSAR as Sheet 3 of Figure 7.3.3. The safety injection signals that are considered for steamline break protection are Low Pressurizer Pressure and Compensated Low Steam Pressure.

e. *Identify the case (a or b) from which the statepoint listed in Table 15.4-7, "Limiting Core Parameters Used in Steam Break DNB Analysis," is taken.*

Response: The limiting statepoint is from Case a (with offsite power available). This case resulted in the limiting combination of thermal-hydraulic conditions (heat flux, inlet temperature, pressure, flow, reactivity, etc). The low flow case is non-limiting as was discussed in the licensing of the SLB topical report; see the USNRC Question 222.11 and Westinghouse response (WCAP-9226-P-A, Rev. 1, Section D). The response identified that using an open channel coupled-code was necessary for evaluation of the low flow case and that the low flow case DNBR is bounded by the full flow case. Thus, although the plant response to a steam line break event with a loss of offsite power is presented in the Unit 2 FSAR, the case itself has been shown to be non-limiting with respect to DNBR.

f. *How was it determined that the statepoint listed in Table 15.4-7, "Limiting Core Parameters Used in Steam Break DNB Analysis," is the limiting statepoint for both cases (a) and (b), and among other statepoints?*

Response: The limiting statepoint is from Case a (with offsite power available). Case b (offsite power lost) is not evaluated for DNB. Consistent with the licensing basis for WBN Unit 1, the without power case is analyzed only for the purpose of presenting transient information in the FSAR.

g. *If "Case b results in a more limiting return to power than Case a", then why is the statepoint of Table 15.4-7 taken from Case (a)?*

Response: As previously noted, the case with loss of offsite power is non-limiting and a DNBR is not calculated for it.

h. *Table 15.4-7 indicates the limiting statepoint occurs just prior to accumulator injection. Describe and evaluate the sensitivity of minimum DNBR to accumulator injection setpoint.*

Response: A minimum accumulator pressure that includes uncertainties is assumed in the analysis. This delays actuation of the accumulators in the steam line break analysis and yields a higher peak heat flux due to the delayed delivery of borated water to the core.

i. *In Case (b), the core becomes critical 12 seconds after boron enters the core, and does not become subcritical until after 200 seconds (the duration reported for the analyzed transient). Explain how the plant design meets General Design Criteria 27 for transients in which boric acid from the accumulators is not provided?*

Response: The ECCS actuates and delivers borated water to the core, consistent with the GDC 27 requirement which requires poison addition by the emergency core cooling system. However, due to the conservative assumptions regarding the available ECCS flow, boron content in the RWST and injection lines, and actuation delays, the flow does not immediately return to the core to a subcritical condition. It does, however, keep the core power from increasing substantially due to the steam line break. Power increases slowly during the transient as the delivered boron works to offset the reactivity insertion due to the cooldown. Eventually, sufficient boron is delivered to the core to shut it down.

j. *Case (b) is deemed to be the limiting case because the accumulators do not inject any boric acid into the RCS. What is the largest break size, with offsite power available, that will not depressurize the RCS to below the accumulator injection pressure? Does this case become the limiting case?*

Response: As is discussed previously, the limiting statepoint is from Case a (with offsite power available). Also discussed previously, all breaks will depressurize to the accumulator injection pressure without mitigation. Smaller breaks obviously take longer to reach the accumulator injection pressure than larger breaks. Smaller breaks result in less cooldown and a lower peak heat flux and are subsequently less limiting.

k. *Demonstrate that DNB does not occur as the result of a MODE 1, inside containment, steam line break, at any time before the reactor trip occurs. Identify the limiting initial power level, and consider environmental effects upon the performance of the overpower reactor trips (neutron flux and ΔT).*

Response: Consistent with the licensing basis for WBN Unit 1, only the zero power steamline break event is considered for WBN Unit 2. The assumption made (and accepted by the NRC) when Unit 1 was licensed is that the OPΔT function with the setpoints generated consistent with WCAP-8745-P-A will provide a reactor trip prior to a violation of the DNB design basis. As such, only the post-trip portion of the transient needs to be explicitly evaluated. The post-trip portion of the transient is bounded by the hot zero power steamline break analysis presented in Section 15.4.2 of the Unit 2 FSAR.

l. *Show all the transient results (plots and sequences of events), for all cases, extending to at least ten minutes, when manual actions might reasonably be expected to begin. Include plots of transient heat flux.*

Response: The analysis was run for 10 minutes. The plots were truncated at 200 seconds because core heat flux had peaked in both cases before 200 seconds. The only event that could be added to the sequence of events is the time at which the core returns to a subcritical condition which is 58.4 seconds for the with-power case and 185 seconds for the without-power case. The requested plots are provided in Attachment 6.

m. *What is the most negative value of the moderator temperature coefficient used for analysis of events occurring at hot full power (e.g., steam line break)?*

Response: The moderator temperature coefficient is modeled in LOFTRAN as a density coefficient. Thus the most negative temperature coefficient is modeled as a most positive density coefficient. The largest positive density coefficient assumed for full power events is 0.43 Δk/gm/cc. As is noted previously, steamline breaks at full power are not considered for WBN Unit 2 consistent with the licensing basis for WBN Unit 1.

n. *What is the limiting break size for the hot full power steam line break? How is that determined?*

Response: Consistent with the licensing basis for WBN Unit 1, only the zero power steamline break event is considered for WBN Unit 2. The assumption made (and accepted by the NRC) when Unit 1 was licensed) is that the OPΔT function with the setpoints generated consistent with WCAP-8745-P-A will provide a reactor trip prior to a violation of the DNB design basis. As such, only the post-trip portion of the transient needs to be explicitly evaluated. The post-trip portion of the transient is bounded by the hot zero power steamline break analysis presented in Section 15.4.2 of the Unit 2 FSAR.

o. *In FSAR Section 5.2.1.5, it is stated that, "A large shutdown margin, coupled with no feedback or decay heat, prevents heat generation during the transient." Provide an analysis or evaluation to support this statement.*

Response: The NSSS design transients (i.e., temperature, pressure, and flow transients) discussed in Section 5.2.1.5 are specified in the RCS component specifications for use in the analyses of the cyclic behavior of the RCS components. To provide the necessary high degree of integrity for the NSSS components, the transient parameters selected for component fatigue analyses are based on conservative estimates of the magnitude and frequency of the temperature and pressure transients resulting from various plant operating conditions. Since the steam line break transient is a cooldown transient, to maximize the cooldown no decay heat and a large shutdown margin with no feedback is assumed for the steam line break design transient analyses. The large steam line break transient is an ASME level D or faulted condition transient, which has only one event for the design life of the plant. The small steam line break is an ASME level C or emergency condition transient that is assumed to have 5 events for the life of the plant (See Unit 2 FSAR Tables 5.2-2 and 5.2-3).

SER 15.3.3, "Major Secondary System Pipe Rupture: Feedwater System Line Break"
(FSAR 15.4.2.1)

15.3.3 – 1. *What are the acceptance criteria by which this accident analysis is judged?*

Response: The acceptance criteria used for the analysis are those described in Section 2.0.C. of WCAP-9230 "Report on the Consequences of a Postulated Main Feedline Rupture, Proprietary," which are as follows:

1. Maximum pressures do not exceed those specified for service limit D as defined in ASME Nuclear Power Plant Components Code, Section III.

2. The core remains in place and geometrically intact with no loss of core cooling capability because

   a. the DNB ratio is such that there is a 95 percent probability that the limiting fuel rod does not go through DNB, with a 95 percent confidence level.

   b. the core remains covered with water.

3. Any activity release must be such that the calculated doses at the site boundary are within the guidelines of 10 CFR Part 100.

To conservatively assure meeting these basic criteria, the criterion established is that no boiling occurs in the primary coolant system following a feedline rupture prior to the time that heat removal capability of the steam generators being fed auxiliary feedwater exceeds the core heat generation assuming prudent operator actions.

15.3.3 – 2. *What is the basis for the assumption of 15 percent for the initial break flow quality?*

Response: The modeling of the break flow quality is consistent with that previously submitted and accepted for WBN Unit 1. The basis for the 15% break flow quality assumed in this method is described below.

The initial break flow quality of 15% is based on a previously completed detailed analysis of a Model D steam generator during a feedline break transient. This value simulates the increase in quality of the steam generator inventory as it reverses flow and passes back through the preheater prior to exiting out the break. Additional discussion of the sensitivity studies encompassing a spectrum of feedwater line break sizes and levels of water entrainment is documented in Section 5.C.16 of WCAP-9230 "Report on the Consequences of a Postulated Main Feedline Rupture, Proprietary."

15.3.3 – 3. *How is it determined that the core remains covered?*

Response: The analysis conservatively demonstrates that no bulk boiling will occur in the RCS prior to the time when the heat removal capability of the steam generators, being fed auxiliary feedwater, exceeds NSSS heat generation. The prevention of bulk boiling in the RCS is shown by demonstrating that the margin to hot leg saturation remains above 0°F during the time frame of interest.

15.3.3 – 4. *For the feedwater line break inside containment, what is the effect of the steam environment upon the steam generator water level measurement?*

Response: Steam generator water level is measured with differential pressure transmitters. The high temperature steam environment that results from a feedwater line break inside containment causes heatup of the transmitter reference leg heatup. This heatup results in a decrease in the water density in the reference leg which results, in turn, in an increase in the indicated water level.

Since the steam generator water level reference legs are not insulated, the Low-Low steam generator level reactor trip signal is not credited for feedwater line breaks inside containment. Analysis shows that the high containment pressure Safety Injection (SI) signal will be generated in less than 1 second after the postulated feedwater line break. The high containment pressure SI signal will initiate the safeguards functions required to mitigate this event, including a reactor trip and auxiliary feedwater system actuation.

The FSAR analysis which credits the Low-Low steam generator level reactor trip signal is retained as the analysis for the feedwater line break outside containment and as a bounding analysis for the inside containment break event. The NRC reviewed this and approved this design as part of the WBN Unit 1 initial license. The Unit 2 design is the same as Unit 1.

15.3.3 – 5. *How is it determined that the double-ended break area (0.223 ft^2) is the limiting break size?*

Response: Based on the design of the pre-heater steam generator and on the flow restrictor in particular, the break size of 0.223 ft^2 is the largest that can be postulated for the plant. Smaller break sizes are possible, but these would be anticipated to result in some main feedwater being delivered to the intact steam generators (WCAP-9230) which would provide a significant benefit. In the analysis of the 0.223 ft^2 break, all main feedwater to the intact and faulted steam generators is conservatively terminated. Therefore, this break size as analyzed by Westinghouse is bounding compared to smaller ones.

Additional discussion of the sensitivity studies encompassing a spectrum of feedwater line break sizes is documented in Section 5.C.15 of WCAP-9230, "Report on the Consequences of a Postulated Main Feedline Rupture, Proprietary."

15.3.3 – 6. *How does steam generator heat transfer area decrease in relation to the shellside liquid inventory decrease?*

Response: The steam generator effectiveness is reduced using the methods described in Section 4.1.1 of WCAP-7907-P-A, "LOFTRAN Code Description."

15.3.3 – 7. *Why is the case assuming failure of one motor-driven auxiliary feedwater pump more limiting than the case assuming failure of the turbine-driven auxiliary feedwater pump (TDAFWP)?*

Response: In the case assuming the failure of the turbine driven auxiliary feedwater pump (TDAFWP), the one MDAFWP provides 410 gpm total to 2 intact SGs. The MD pump connected to the faulted SG is also assumed to deliver half of its flow to the other intact SG prior to steamline isolation (SG pressures are approximately uniform, so equal delivery is assumed during this brief period). After steamline isolation, the pump connected to the faulted SG is initially assumed to deliver no flow to the intact SG. However, after the pressure in the faulted SG drops below 360 psig, the branch line to the faulted SG is automatically restricted, thus permitting some flow (60 gpm) to the intact SG.

In the motor-driven auxiliary feedwater pump (MDAFWP) failure scenario, during the first 12 minutes following the first low-low SG level signal, only the 60 gpm from the remaining MDAFWP is delivered to the intact SGs. As in the MDAFWP failure scenario, the MDAFWP connected to the faulted SG is initially assumed to deliver no flow to the intact SG. However, after the pressure in the faulted SG drops below 360 psig, the branch line to the faulted SG is automatically restricted, thus permitting some flow (60 gpm) to the intact SG. The TDAFWP is then made available 12 minutes after the low-low SG level signal.

As such, during the approximately 11 minutes from the startup of the diesel generators and auxiliary feedwater pumps to the assumed operator action time, very little flow (60 gpm after steamline isolation) is delivered to the intact SGs in the MDAFWP failure case, whereas 410 gpm plus 60 gpm after steamline isolation is delivered in the TDAFWP failure case. As a result, the net effect of the AFW flow in the early portion of the transient provides more of a benefit than the larger amount of AFW flow at a later time in the event.

15.3.3 – 8. *Verify that the analysis models the TDAFWP start on receipt of low-low level signals from two SGs, not just the faulted SG.*

Response: The signal associated with the TDAFWP start is not relevant to the analysis as the analysis conservatively does not model AFW flow delivered to the SGs from the TDAFWP until operator action is taken to isolate the affected SG 12 minutes after the time of the first low-low SG level signal.

15.3.3 – 9. *Show that this case, which assumes the operation of the pressurizer PORVs, yields more conservative results than a case that does not assume the operation of the pressurizer PORVs.*

Response: The modeling of the operation of the pressurizer PORVs for the feedwater line break event has been previously accepted for WBN Unit 1. WBN specific sensitivity studies have shown that modeling the PORVs results in more severe results for this transient. For plants with high-head safety injection systems that do not inject substantial amounts of flow at pressures above the PORV setpoint, cases without PORVs are typically limiting. These cases are limiting because modeling the PORVs reduces the saturation pressure of the RCS, and therefore reduces the margin to hot leg saturation. Modeling the PORVs also results in an increase in safety injection flow, a result of the decrease in pressure. The increased safety injection flow provides a benefit to cooling the RCS, but since the safety injection flow is low around the PORV setpoint, the benefit is smaller than the penalty associated with the reduction in pressure caused by the opening of the PORVs. This is the case for WBN, and as such, the PORVs are conservatively modeled.

SER 15.3.4, "Reactor Coolant Pump Rotor Seizure (or Locked Rotor)" (FSAR 15.4.4)

(Note: Provide legible plots depicting the results of the event analyses.)

Response: Legible plots are provided in Attachment 4.

15.3.4 – 1. *Provide a discussion of the analysis or evaluation of the radiological consequences of the Reactor Coolant Pump Rotor Seizure event.*

Response: In TVA to NRC letter dated July 2, 2010, "Watts Bar Nuclear Plant (WBN) Unit 2 - Submittal of Additional Information Requested During May 12, 2010, Request for Additional Information (RAI) Clarification Teleconference Regarding Environmental Review (TAC No. MD8203)" (ADAMS Accession No. ML101930470), a similar RAI was answered. The RAI and the response to it as contained in that letter were as follow:

> "3. The NRC indicated that TVA did not analyze other SRP events such as RCP locked rotor and RCP shaft break.
>
> The rod ejection accident was not analyzed for radiation impact. It is noted in FSAR Section 15.4.6.3 that this event is bounded radiologically by the large break LOCA. The current Unit 1 thermal hydraulic analysis for the Chapter 15 locked rotor and shaft break events does not predict any fuel experiencing departure from nucleate boiling (failure); as such, a dose analysis has not been performed for Chapter 15. It is anticipated WBN Unit 2 would have similar results when the final core design and thermal hydraulic analysis are completed for Unit 2. Unit 2 uses the same Westinghouse fuel type as Unit 1 and a lower licensed thermal power. At this time WBN Unit 2 does not intend to perform a locked rotor or RCP shaft break offsite dose analysis."

15.3.4 – 2. *FSAR Section 15.4.4.1 states, "[o]nly one analysis is performed, representing the most limiting condition for the locked rotor and pump shaft break accidents." Which is more limiting, the locked rotor or the pump shaft break, and why? How was this conclusion reached?*

Response: Only one scenario is analyzed, representing the most limiting condition for the locked rotor and pump shaft break accidents. As stated in the Unit 2 FSAR, "the initial rate of reduction of coolant flow is greater for the locked rotor event. However, with a failed shaft, the impeller could conceivably be free to spin in the reverse direction as opposed to being fixed in position as assumed for a locked rotor. The effect of such reverse spinning is a slight decrease in the endpoint (steady-state) core flow when compared to the locked rotor." This is modeled in the analysis by conservatively simulating the rotor as locked for forward flow and free-spinning for reverse flow.

15.3.4 – 3. *Only the locked rotor with a loss of offsite power was analyzed. Justify this case as more limiting than the case with offsite power.*

Response: Assuming a loss of offsite power at the beginning of the transient results in the coastdown of the unfaulted RCPs and a reduction in the total core flow. This further reduction in flow results in more severe results compared to the case in which the RCPs continue to operate. This modeling of the locked rotor with a loss of offsite power is the same as that previously submitted and accepted for WBN Unit 1.

15.3.4 – 4. *Identify the single failure that was assumed in the analysis.*

Response: The single failure assumed in the analysis is the failure of one train of the reactor protection system. The protection function is carried out by the remaining train.

SER 15.4.3 "Steam Generator Tube Rupture" (FSAR 15.4.4)

15.4.3 – 1. *Discuss the classification of an SGTR as a Condition IV occurrence rather than a Condition III occurrence at WBN Unit 2.*

Response: The Steam Generator Tube Rupture (SGTR) accident postulates the complete severance of a steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The SGTR accident will result in the transfer of contaminated reactor coolant into the secondary systems and subsequent release of a portion of the activity to the atmosphere. Therefore, analyses are required to ensure that the radiological consequences resulting from the SGTR are within the 10 CFR 100 limits as discussed in Unit 2 FSAR 15.5.5.

The steam generator tube material is Inconel-600 and is a highly ductile material; thus, it is considered that the assumption of a complete severance of a tube is conservative. Subsequently, the more probable mode of steam generator tube failure would be one or more minor leaks. Therefore, the SGTR event is unlikely to occur during the life of the plant; reference Unit 2 FSAR 15.4.3.

Given the above, the SGTR event is correctly categorized at WBN Unit 2 as a Condition IV Event for the following reasons. SGTR and Condition IV Events, *Limiting Faults* (Fault < 10E-2) are not expected to occur during the life of the plant. However, these events are postulated because their Consequences include the potential for the release of significant amounts of radioactive material to the

environment. Condition IV faults are required to be analyzed to ensure the environmental release in not in excess of the 10 CFR 100 guideline values. As previously stated, the SGTR radiological consequences analysis is provided in FSAR 15.5.5. The SGTR should not be classified as a Condition III Event, *Infrequent Faults* (10E-2 < Fault < 10E-1) because these faults may occur very infrequently during the life of the plant; reference Utility Service Alliance 10 CFR 50.59 Resource Manual, Appendix 7.1.

Furthermore, the industry standard regarding content of the FSAR's subject matter and analyses considers the SGTR accident to be a Condition IV Event. Section 15.6.3 of NUREG- 800 addresses the review of the radiological consequences of an SGTR at a pressurized water reactor making releases to the environment. The acceptance criteria are based on the requirements of 10 CFR 100 as it relates to mitigating the radiological consequences of accident as previously stated.

15.4.3 – 2. *Section 15.4.3.2, "Analysis of Effects and Consequences" refers to the margin to SG overfill for a design basis SGTR for WBN Units 1 and 2. Does this section pertain to Unit 2 or to both units?*

Response: The margin to SG overfill is applicable to both units as demonstrated by NPG Design Criteria Document WB-DC-40-70, "Accident Analysis Parameter Checklist (AAPC)":

- (INTRODUCTION) states, "...This design criteria has been evaluated and determined to be applicable to WBN Units 1 and 2 ..."
- (Steam Generator Tube Rupture) addresses the margin to SG overfill.

15.4.3.2 (Analysis of Effects and Consequences) of the Unit 1 UFSAR states, "The results of this analysis demonstrated that there is margin to steam generator overfill for a design basis SGTR for WBN Units 1 and 2."

WB-DC-40-70 does note that there is a difference between the two units. Specifically, 1.1 (SCOPE) of it states, "In parallel with the Replacement Steam Generator (RSG) Program (Reference 2.2.43), the safety analyses also addressed the impact of reducing the reactor coolant vessel average temperature T_{avg} setpoint by 2°F from the current value of 588.2°F to 586.2°F. In the event the actual steam pressure with the RSGs is higher than desired, this T_{avg} reduction strategy supports a reduction in steam pressure while maintaining the ability to operate at full NSSS power (3,475 Mwt). ..."

15.4.3 – 3. *Explain why the assumed operator action times are precise to two decimal places (e.g., "After depressurization is completed, an operator action time of 4.07 minutes is assumed prior to initiation of safety injection termination.")*

Response: The major operator actions required for the recovery from an SGTR were simulated in a thermal and hydraulic analysis program. The SGTR analysis sequence of events and the program limitations for simulating operator actions are within seconds (see Unit 2 FSAR Table 15.4-21). The two decimal places for minutes on operator action times would put the action within seconds. If operator action time uses only one decimal place for minutes, it would put operator action times in 6-second increment blocks which would not be within the limits of the SGTR Analysis Sequence of Events as shown in Unit 2 FSAR Table 15.4-20.

15.4.3 – 4. *Provide a confirmation and basis, by simulator test results or equivalent, that all assumed operator action times have been justified (see License Condition 41 of SSER 5 regarding a demonstration that the operator action times assumed in the analysis are realistic).*

Response: Operator action times for WBN Unit 1 were validated through simulator test results, emergency drill, etc. The design of Unit 2 is essentially the same as Unit 1. Thus, the operator action times would also be expected to be similar. The Unit 2 operator action times will be validated in the plant and on the simulator through the validation of procedures, the startup test program, and operator training programs.

15.4.3 – 5. *Identify the analysis acceptance criteria that must be satisfied, and discuss how satisfying these criteria are equivalent to meeting the Condition IV acceptance criteria.*

Response: Given below is the NRC NUREG-800, Section 15.6.3-II (Acceptance Criteria), a summary of the Steam Generator Tube Rupture analyses.

NRC NUREG 800 Section 15.6.3-II Acceptance Criteria

The acceptance criteria are based on the relevant requirements of 10 CFR Part 100 as it relates to mitigating the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated steam generator tube failure accident at a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the following exposure guidelines:

(1) for the postulated accident with an assumed pre-accident iodine spike in the reactor coolant and for the postulated accident with

the highest worth control rod stuck out of the core the calculated doses should not exceed the guideline values of 10 CFR Part 100, Section 11, and

(2) for the postulated accident with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem, respectively, for the whole-body and thyroid doses.

The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of Regulatory Guide 1.4 except for the atmospheric dispersion factors which are reviewed under SRP Section 2.3.4. Plant Technical Specifications are required for iodine activity in the primary and secondary coolant systems. These specifications are acceptable if the calculated potential radiological consequences from the steam generator tube failure accident are within the exposure guidelines for the above two cases.

ANSI N18.2 - 1973 Condition IV Event Requirements Regarding the SGTR Accident

The Steam Generator Tube Rupture (SGTR) accident postulates the complete severance of a steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The SGTR accident will result in the transfer of contaminated reactor coolant into the secondary systems and subsequent release of a portion of the activity to the atmosphere. Therefore, analyses are required to ensure that the radiological consequences resulting from the SGTR are within the 10 CFR 100 limits as discussed in Unit 2 FSAR Section 15.5.5.

The steam generator tube material is Inconel-600 and is a highly ductile material; thus, it is considered that the assumption of a complete severance of a tube is conservative. Subsequently, the more probable mode of steam generator tube failure would be one or more minor leaks. Therefore, the SGTR event is unlikely to occur during the life of the plant; reference Unit 2 FSAR Section 15.4.3.

SGTR and Condition IV Events, *Limiting Faults* (Fault < 10E-2) are not expected to occur during the life of the plant. However, these events are postulated because their consequences include the potential for the release of significant amounts of radioactive material to the environment. Condition IV faults are required to be

analyzed to ensure the environmental release is not in excess of the 10 CFR 100 guideline values. As previously stated, the SGTR radiological consequences analyses is provided in Unit 2 FSAR Section 15.5.5.

WBN Analyses:

Unit 2 FSAR Table 15.5-19 provided the offsite doses analyses at the Low Population Zone (LPZ) and the Exclusion Area Boundary (EAB) to comply with 10 CFR 100 in the event of an SGTR accident. Given below are calculation WBNTSR-008 (*Control Room Operator and Offsite Doses Due to a Steam Generator Tube Rupture*) parameters used in the SGTR analysis that were also identified in the above NUREG-800 SGTR acceptance criteria:

- The primary to secondary leak rates were based on Reactor Coolant System Operational Leakage Technical Specifications LCO 3.4.13 prior to the accident with 11 gpm (10 gpm identified + 1 gpm unidentified) and a maximum of 150 gpd in the steam generators.
- The activities in the primary coolant are based on Technical Specifications LCO 3.4.16 Required Action Limits with a pre-existing iodine spike of 21 μCi/gm Dose Equivalent I-131 for a pre-accident coolant.
- Technical Specifications LCO 3.4.16 Condition Limit accident initiated iodine spike of 0.265 μCi/gm Dose Equivalent I-131 with a factor of 500 increase in iodine release rate from the fuel.
- The secondary coolant activities were based on Technical Specification LCO 3.7.14 secondary specific activity limit of 0.10 μCi/gm Dose Equivalent I-131.
- The computer code STP was used to determine the radioactive releases during the accident. These releases are used as input to the computer code FENCDOSE to determine the LPZ and EAB doses. The computer runs used in the SGTR analyses utilize the same methodologies and appropriate assumptions for calculating the radiological consequences of the WBN LOCA analyses in accordance with Regulatory Guide 1.4.

Credit is taken for partial flashing of the reactor coolant as it enters the steam generator. For conservatism, no credit is taken for "scrubbing" of iodine in the steam bubbles as the bubbles rise in the water; therefore, it does not matter if the break location is above or below the water for the duration of the accident.

The offsite doses (gamma, beta, thyroid, and TEDE) due to an SGTR with a pre-existing iodine spike do not exceed the 10 CFR 100 limits (25 rem gamma, 300 beta, and 300 rem thyroid) from NUREG 800. The SGTR with accident initiated iodine spike

does not exceed the small fraction 10% of the 10 CFR 100 limits as shown in the response to **RAI 15.4.3 - 6**.

15.4.3 – 6. *Show how the analysis results satisfy the analysis acceptance criteria.*

Response: The Steam Generator Tube Rupture (SGTR) offsite doses (gamma, beta, and thyroid) with a pre-existing iodine spike do not exceed the 10 CFR 100 limits (25 rem gamma, 300 beta, and 300 rem thyroid), and the SGTR with accident initiated iodine spike does not exceed the 10% small fraction of the 10 CFR 100 limits provided in NUREG-800, Section 15.4.3, SGTR Acceptance Criteria . Given below are the SGTR 10 CFR 100 doses from Unit 2 FSAR Table 15.5-19 compared to the SGTR Acceptance Criteria. These values were calculated in Appendix G (*Unit 2 SGTR*) of calculation WBNTSR-008 (*Control Room Operator and Offsite Doses Due to a Steam Generator Tube Rupture*).

Table 15.5-19 10 CFR 100 Doses from Steam Generator Tube Rupture

Pre-Accident Initiated Spike Case (21 μCi/cc Dose Equivalent I-131 Maximum Peak)

(rem)	2 Hr. EAB	30 Day LPZ	10 CFR 100 Limit
Gamma	3.59E-01	8.76E-02	25 rem
Beta	2.06E-01	5.25E-02	300 rem
Thyroid - ICRP -30	1.38E+01	3.28E+00	300 rem

Accident Initiated Spike Case (0.265 μCi/cc Dose Equivalent I-131 Steady State)

(rem)	2 Hr. EAB	30 Day LPZ	10% (10 CFR 100) Limit
Gamma	3.99E-01	9.72E-02	2.5 rem
Beta	2.10E-01	5.36E-02	30 rem
Thyroid - ICRP -30	4.49E+00	1.09E+00	30 rem

SER 15.4.4, "Control Rod Ejection" (FSAR 15.4.6)

15.4.4 – 1. *Provide a discussion of the analysis or evaluation of the radiological consequences of the Rod Ejection accident to show that they are bounded by the radiological consequences of the LOCA (FSAR Section 15.5.7). The information will be reviewed according to NUREG-0800 Section 15.4.8.*

Response: This discussion evaluates the source terms used in the LOCA and Rod Cluster Control Assembly Ejection (RCCAE) accidents to determine if the radiological consequences of the LOCA will bound the radiological consequences of the RCCAE. Evaluating these source terms to determine if the consequences of a LOCA are bounding is acceptable since an RCCAE is a LOCA type accident that will initiate a Safety Injection signal and both accidents will be mitigated by the same ECCS and containment cooling systems.

The design basis LOCA postulates a double-ended rupture of a reactor coolant pipe with subsequent blow-down resulting in rapid depressurization of the reactor core. An increase in fuel cladding temperature is anticipated to the point where [1] some cladding failure may occur in the hottest regions of the reactor core, [2] resulting result in fission products release into the reactor coolant system, and [3] ultimately fission products will enter the ECCS and primary containment. The radiological consequences of a LOCA are based on Regulatory Guide 1.4 and a total of 100% of the noble gas core inventory and 25% of the core iodine inventory is assumed to be immediately available for leakage from primary containment as shown in Unit 2 FSAR Table 15.5-6. The results of the radiological consequences of a LOCA are shown in FSAR Table 15.5.-9.

The RCCAE is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of an RCCAE is a mechanical failure which will result in a rapid positive reactivity insertion together with an adverse power distribution, possibly leading to localized fuel rod damage. Even for a worse case basis, the analyses indicate that the described fuel and clad limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the reactor coolant. The RCCAE event demonstrated that the number of fuel rod entering the DNB amounts to less than 10%, thus satisfactorily limiting fission product release. Subsequently, the assumed radio isotopes activity released to the reactor coolant from failed fuel is composed of 10% noble gases gap activity, 10% iodine gap activity, 0.25% of the noble gas core inventory, and 0.25% iodine core inventory.

A comparison of the LOCA and RCCAE radioisotopes discussed above indicates the LOCA inventories are clearly dominant and, subsequently, the environmental consequences of a LOCA will

bound the consequences of an RCCAE. It is noted that both accidents will initiate a Safety Injection Signal and both accidents will be mitigated by the same ECCS and containment cooling systems. This position is consistent with the conclusion identified in FSAR 15.4.6.3, *Conclusions.*

15.4.4 – 2. *Provide a reference for the revised Westinghouse acceptance criterion, which permits the average clad temperature at the hot spot to exceed 3000 °F.*

Response: The revised Westinghouse acceptance criterion is provided in Westinghouse letter NS-NRC-89-3466 (Reference 48 of the Unit 2 FSAR section, repeated below). Letter NS-NRC-89-3466 documents the Westinghouse position of using fuel rod pellet enthalpy as the acceptance criterion to demonstrate core coolability for the Rod Ejection accident. The paragraph in the FSAR was intended to note the change in acceptance criteria relative to earlier version of the FSAR.

Letter from W. J. Johnson of Westinghouse to R. C. Jones of the NRC, "Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents," NS-NRC-89-3466, October 1989.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Sections 3.2 and 5.2 (from NRC letter dated September 28, 2010 (ADAMS Accession No. ML102630598))

EMCB 3.2-1 *The Nuclear Regulatory Commission (NRC) staff noted a number of instances in the review of Section 3.2 and corresponding tables and figures of Amendment 95 to the Watts Bar Nuclear Plant (WBN), Unit 2, Final Safety Analysis Report (FSAR) (Reference 1), where editorial modifications may be necessitated in subsequent revisions to the WBN, Unit 2 FSAR. Please review the following NRC staff notations and rectify, as necessary.*

1) *In Table 3.2-2, page 9, of Reference 1, 'Upper Containment' and 'CRDM & Instrument Room Cooling' appear not to be components themselves, but rather equipment location/ category headings. If these are in fact headings, then shouldn't the information columns be blank?*

2) *In Table 3.2-2, pages 11 and 12, of Reference 1, was it intentional to delete the 'Rad Source' column information for the Main Steam Relief Valves and Safety Valves when relocating them in Table 3.2-2 from page 12 to page 11? If not, then this information should be restored.*

3) *In Table 3.2-2, page 12, of Reference 1, the TVA/ANS Safety Class' column information is missing for the Spent Fuel Purification Pumps.*

4) *In Table 3.2-2, page 17, of Reference 1, the 'Seismic' column information is missing for 'Essential Raw Cooling Water System Discharge Header Air Release Valves & Piping.'*

Response: It is acknowledged that typographical errors occurred and editorial modifications are required to Unit 2 FSAR Table 3.2-2. These corrections will be incorporated in a future amendment to the Unit 2 FSAR.

Specifically:

- Item 1): "Upper Containment, CRDM, & Instrument Room Cooling" are equipment location/category headings. The information in the corresponding columns will be blanked out.
- Item 2): Column (6) "Rad Source" will be completed with the appropriate "dash" for Main Steam System "Relief Valves" and "Safety Valves."
- Item 3): Column (2) "TVA/ANS Safety Class" will be completed with the appropriate "G" for the "Spent Fuel Purification Pumps."
- Item 4): Column (7) "Seismic" will be completed with the appropriate "I(L)" for the "Valves (Discharge Header Air Release & Piping)."

EMCB 5.2.1.4-1 *In using Code Case 1423-2 for Reactor Coolant Piping/Branch Nozzles, as shown in Table 5.2-8, page 3, of Reference 2, did Tennessee Valley Authority implement the limitations and modifications identified in Regulatory Guide (RG) 1.84 for use of this code case? If not, then please provide a supporting justification/basis for not implementing the limitations and modifications identified in RG 1.84 for the use of Code Case 1423-2. Also, in Section 5.2.1.4 of Reference 2, the Class 1 Code Case List does not include Code Case 1423-2 for Reactor Coolant Piping, even though this Code Case is mentioned in Table 5.2-8. Please confirm whether this Code Case will be added in Section 5.2.1.4 for completeness.*

Response: Amendment 97 to the Unit 2 FSAR inadvertently incorporated Code Case 1423-2 into Table 5.2-8. This resulted from an incorrect interpretation of the applicable sections of a vendor's calculation, "Code Review of Reactor Coolant Loop Piping Pressure Design Calculations," during the development of the originating FSAR change package.

A future amendment to Unit 2 FSAR Table 5.2-8 will remove the reference to Code Case 1423-2 for the branch nozzles material specifications. A change to Section 5.2.1.4 will not be necessary because the future amendment will reconcile Table 5.2-8 and Section 5.2.1.4.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Section 7.3 (from NRC letter dated September 28, 2010 (ADAMS Accession No. ML102640324))

7.3 - 1. *In Amendment 95, Final Safety Analysis Report (FSAR) Section 7.3.2.3 "Further Considerations," the list of signals that would start the auxiliary feedwater motor driven and turbine driven pumps was moved to Table 7.3-1 Item 3, Auxiliary Feedwater. However, Item (6) 'AMSAC' was not included in Table 7.3-1.*

Please explain this omission or state your commitment to correct this in a future amendment to the FSAR.

Response: **NOTE:** This item is also being tracked as item 287 of the I&C Open Items Master List.

The AMSAC start is not included based on Unit 1 UFSAR Change Package 1554S0 item 20 which states:

"**20** (page 7.3-17, 18 and Table 7.3-1): The initiating signals for Auxiliary Feedwater (AFW) are moved from Section 7.3.2.3 to Table 7.3-1, which lists ESF instrumentation. A reference to the Table is added. This change also clarifies that the AFW pumps are started by trip of both Turbine-Driven Main Feedwater (MFW) pumps rather than all MFW pumps as currently stated since trip of the Standby MFW pump does not initiate AFW. This is consistent with the description of the Auxiliary Feedwater System in Section 10.4.9. This change also deletes AMSAC from the list of AFW start signals. As described in Section 7.7.1.12, the AMSAC system is non-safety and provides a diverse means of initiating AFW and turbine trip under conditions indicative of an ATWS event. AMSAC was not designed as an Engineered Safety Feature and is not included in the ESFAS Technical Specification 3.3.2 for AFW start. Therefore, it does not belong in the Table which identifies ESF instrumentation. The change does not alter the AMSAC functions of AFW start and turbine trip. The Switchover from Injection to recirculation and the switchover initiating signals are also added to Table 7.3-1 since they are considered to be part of the ESFAS. The listing of switchover instrumentation is consistent with the description of the switchover function in Section 7.6.9. Also numbered the notes at the bottom of the Table."

7.3 - 2. *In Amendment 95, FSAR Section 7.3.1.1.1 "Function Initiation," item (13) was arranged into paragraph form from what used to be a listing of Items (a), (b) and (c).*

The second bullet under Item (c) was omitted in the new paragraph.

Initiates Phase B containment isolation of the following:

- *"Closure of the main steam isolation valves (MSIV) to limit reactor coolant system cooldown for breaks downstream of the MSIVs."*

Please explain this omission or state your commitment to correct this in a future amendment to the FSAR.

Response: **NOTE:** This item is also being tracked as item 294 of the I&C Open Items Master List.

Closure of the main steam isolation valves is not included based on Unit 1 UFSAR Change Package 1896S0 which states:

"FSAR section 7.3.1.1.1 item 12 contains information which is incorrect and conflicts with information in the same chapter and other sections. Specifically, item 12 indicates that containment spray initiates Phase B containment isolation and that Phase B initiates main steam line isolation. Actually, containment spray, Phase B containment isolation, and main steam isolation are all actuated by high-high containment pressure as shown on Figures 7.3-3 Sheet 4 and 6.2.4-21. Both of these figures depict the functional logic shown on configuration control drawing 1-47W611-88-1. This logic is also described in other sections, e.g., 6.2.1.3.10, 6.2.2.5, 6.2.4.2, 10.3.3. This change will clarify the information in 7.3.1.1.1 to resolve these inconsistencies."

7.3 - 3. *In Amendment 95, FSAR Section 7.3.1.1.2 "Process Protection Circuitry," Item (3) references to Sections 7.6 and 7.7 were removed.*

Please explain the reason for removal.

Response: **NOTE:** This item is also being tracked as item 295 of the I&C Open Items Master List.

References to section 7.6 and 7.7 are not included based on Unit 1 UFSAR Change Package 1554S0 item 5 which states:

"**5** (page 7.3-4): Revise item 3 of Section 7.3.1.1.2 to simplify the discussion of valve position information available during the post-LOCA recovery period."

7.3 - 4. *In Amendment 95, FSAR Section 7.3.1.2.1 "Generating Station Conditions," the new paragraph was arranged from what used to be a listing of Items (1.b), (1.c), and (2.b), leaving out Items (1.a) and (2.a). Even if the paragraph contains the word 'include,' the breaks in Items (1.a) and (2.a) should be listed.*

Please explain this omission or state your commitment to correct this in a future amendment to the FSAR.

Response: **NOTE:** This item is also being tracked as item 296 of the I&C Open Items Master List.

The changes are based on Unit 1 UFSAR Change Package 1554S0 item 8 which states:

"**8** (page 7.3-6): Revise the section summarizing the generating station conditions which require protective action. The list is not intended to be a complete list of the design basis events which the protection system is designed to mitigate. The change simplifies the summary, adds feedwater line break, and adds a reference to Chapter 15 for identification of the conditions requiring protective action. System Description N3-99-4003 is similarly revised."

Deletion of the breaks described in Items (1.a) (i.e., "Rupture in small pipes or cracks in large pipes") and (2.a) (i.e., "Minor secondary system pipe breaks resulting in steam release rates equivalent to a single dump, relief or safety valve") is justified because they are encompassed by the "primary system breaks" and "secondary system breaks", respectively, that are noted in the revised version of 7.3.1.2.1.

7.3 - 5. *In Amendment 95, FSAR Section 7.3.1.2.2 "Generating Station Variables," the following sentence was deleted:*

> *Post accident monitoring requirements and variables are given in Tables 7.5-1 and 7.5-2.*

Please explain the reason for removal.

Response: **NOTE:** This item is also being tracked as item 297 of the I&C Open Items Master List.

The changes are based on Unit 1 UFSAR Change Package 1554S0 item 9 which states:

"**9** (page 7.3-7 and Table 7.3-2, item 3): Revise the summary of the generating station variables which are required for initiation of protective action by the ESFAS. The change simplifies the summary, eliminates repetition, and adds steam generator level and reactor coolant temperature (T_{avg}) as monitored variables. Low-Low SG level starts Auxiliary Feedwater. High-High SG level initiates Feedwater Isolation. Low T_{avg} coincident with a Reactor Trip also initiates Feedwater Isolation. Low T_{avg} with a note to identify the interlock with Permissive P-4 (reactor trip), is also added to Table 7.3-2, item 3, which lists the conditions that initiate Feedwater Isolation. Addition of these variables is consistent with discussions of the Main and Auxiliary Feedwater Systems in Sections 10.4.7, 10.4.9, various Chapter 15 events (e.g., Sections 15.2.10,15.3.1,15.4.2), and Technical Specification Bases 3.3.2 for the P-4 interlock. System Description N3-99-4003 is similarly revised to add steam generator level and reactor coolant temperature."

Unit 2 FSAR Section 7.3 addresses Engineered Safety Features (ESF) Actuation System. Post accident monitoring is not an ESF; thus, a reference to it is not required in 7.3.1.2.2.

7.3 - 6. *U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement Bulletin 80-06 calls for review of engineered safety features with the objective of ensuring that no device will change position solely because of the 'reset' action.*

In Supplement 3 of NUREG-0847, Section 7.3.5, the staff approved the design modifications proposed by the applicant that would allow certain devices to remain unchanged upon an engineered safety feature (ESF) reset. The staff also found acceptable the applicant's justification for some safety-related equipment that does not remain in its emergency mode after an ESF reset.

Please confirm whether or not the equipment that was determined in NUREG-0847 and its supplements to remain unchanged upon an ESF reset will still remain unchanged in Unit 2.

Response: **NOTE:** This item is also being tracked as item 298 of the I&C Open Items Master List.

A review of the original Unit 1 Safety Evaluation Report (SER) and Safety Evaluation Supplements (SSERs) was performed with the following results:

7.3.5 IE Bulletin 80-06 - PAGE 7-10 (from original SER)

"IE Bulletin 80-06 calls for review of engineered safety features, with the objective of ensuring that no device will change position solely because of the reset action. The applicant has stated that the requested reviews have been performed, and has committed to perform the confirmatory tests requested by the Bulletin. The applicant stated that the feedwater isolation valves, the main feedwater check valve bypass valves, the upper tap main feedwater isolation valves, the steam generator blowdown isolation valves, and the RHR heat exchanger outlet flow control valves do not remain in the emergency mode after ESF reset. Design modifications are being made for these components to ensure that the valves will remain in the emergency mode after ESF reset. The staff finds the design modifications to be acceptable subject to its review of the electrical schematics, which are not currently available. The staff will report its findings in a supplement to this report."

7.3.5 IE Bulletin 80-06, Page 7-1, Supplement 3 (from SSER 3)

"In FSAR Amendment 48, the applicant submitted the electrical schematics that implement the following changes:

(1) For feedwater isolation valves (FCV-3-33, FCV-3-47, FCV-3-87, and FCV-3-100), feedwater check valve bypass valves (FCV-3-185, FCV-3-186, FCV-3-187, and FCV-3-188), and upper tap main feedwater isolation valves (FCV 3-236, FCV-3-239, FCV-3-242, and FCV-3-245), a new reset switch and a relay have been added for each steam generator loop. When the

engineered safety feature (ESF) signal is reset, the individual valve will not change state until both the loop and the ESF train reset switches have been reset.

(2) For steam generator blowdown isolation valves (FCV-43-54D, FCV-43-56D, FCV-43-59D, FCV-43-63D, FCV-43-55, FCV-43-58, FCV-43-61, and FCV-43-64), the ESF signal is sealed in by means of a valve-mounted limit switch. The individual valve will not change state until a hand switch in the sample room is used to reopen the individual valve.

(3) For residual heat removal heat exchanger outlet flow control valves (FCV-74-16 and FCV-74-28), the ESF signal is sealed in by the limit switch. A new reset switch has been added at the control room control board. When the ESF signal is reset, the individual valve will not change state until the individual reset switch has been reset.

The staff has reviewed the electrical schematics and finds the design modification acceptable."

1.14.1 Bulletins - PAGE 1-17 "Supplement 6" (from SSER 6)

"Bulletin 80-06, ESF Reset Control"

NRC action: Watts Bar SER (NUREG-0847), Section 7.3.5; Inspection Report 50-390/80-12.

1.8 Summary of Confirmatory Issues - PAGE 1-4 **(from SSER 10**

"(24) IE Bulletin 80-06 Resolved (SSER 3) 7.3.5"

SSER 16

"II.K.1.5: I&E bulletins, assure proper ESF functioning - In Section 7.3.5 of the SER and SSER 3, the staff found the applicant's response to Bulletins 79-06A and 80-06 acceptable, but made no explicit reference to II.K.1.5. Since the technical information was found acceptable, even though there was no mention of Item II.K.1.5, the item is closed."

Based on the results of this review, the valves identified in SSER 3 are the ones to which this RAI applies. Since the original designs of Unit 1 and Unit 2 agreed, the design of the other Unit 2 items is the same as for Unit 1 and a specific review of those items is not required.

A review of the schematic diagrams for the WBN Unit 2 valves listed in SER 3 determined the following:

(1) For feedwater isolation valves (FCV-3-33, FCV-3-47, FCV-3-87, and FCV-3-100), feedwater check valve bypass valves (FCV-3-185, FCV-3-186, FCV-3-187, and FCV-3-188), and upper tap main feedwater isolation valves (FCV 3-236, FCV-3-239, FCV-3-242, and FCV-3-245), the Unit 2 equivalent reset switch and a relay have been added for each steam generator loop. When the engineered safety feature (ESF) signal is reset, the individual valve will not change state until both the loop and the ESF train reset switches have been reset.

(2) For steam generator blowdown isolation valves (FCV-43-54D, FCV-43-56D, FCV-43-59D, FCV-43-63D, FCV-43-55, FCV-43-58, FCV-43-61, and FCV-43-64), the ESF signal is sealed in by means of a seal in relay. The individual valve will not change state until a hand switch in the sample room is used to reopen the individual valve. These changes are implemented in EDCR 53917.

(3) For residual heat removal heat exchanger outlet flow control valves (FCV-74-16 and FCV-74-28), the ESF signal is sealed in by the limit switch. The Unit 2 equivalent reset switch has been added at the control room control board. When the ESF signal is reset, the individual valve will not change state until the individual reset switch has been reset.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR Section 15 (from NRC letter dated October 4, 2010 (ADAMS Accession No. ML102700437))

SER 15.3.6, "Anticipated Transients Without Scram"

FSAR Chapter 15.2, CONDITION II - FAULTS OF MODERATE FREQUENCY, contains a paragraph concerning anticipated transients without scram (ATWS), in which reference is made to a series of generic studies that indicate ATWS analyses would yield acceptable consequences in cases where the turbine trip and auxiliary feedwater actuation functions are successful. This paragraph concludes with a reference to the WBN ATWS mitigation system actuation circuitry (AMSAC) design description in FSAR Chapter 7.7.1.12. Please provide the following additional information, concerning ATWS in WBN Unit 2:

15.3.6 - 1. *Provide a discussion verifying that WBN Unit 2, with the AMSAC installed, meets the requirements of General Design Criteria 13, 14, 16, 35, 38, and 50 (NUREG-0800, Section 15.8).*

Response: The discussion verifying that WBN Unit 2 meets the listed General Design Criteria is contained in Unit 2 FSAR section 3.1, "Conformance with NRC General Design Criteria."

15.3.6 - 2. *Provide a discussion of the analysis or evaluation that shows that the reactor coolant system pressure will not exceed the American Society of Mechanical Engineers Service Level C limits (approximately 3200 psig), with AMSAC installed, for the most limiting ATWS event. If the results of generic analyses are cited, show that they apply to the specifics of the WBN Unit 2 design.*

Response: The ATWS analysis applicable to WBN Unit 2 is the generic analysis summarized in letter NS-TMA-2182 (Reference 2). The WBN Unit 2 design is consistent with the representative 4-loop plant model used in the generic ATWS analysis in Reference 2, as discussed in detail below.

For Westinghouse PWRs, the licensing requirements related to ATWS are those specified in the Final ATWS Rule, 10 CFR 50.62(b). The requirement set forth in 10 CFR 50.62(b) is that all Westinghouse designed PWRs must install AMSAC (ATWS Mitigation System Actuation Circuitry). In compliance with 10 CFR 50.62(b), AMSAC will be installed and implemented at WBN Unit 2.

As documented in SECY-83-293 (Reference 6), the analytical bases for the Final ATWS rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. These generic ATWS analyses were formally transmitted to the USNRC via letter NS-TMA-2182 (Reference 2), and were performed based on the guidelines provided in NUREG-0460 (Reference 3).

In the generic ATWS analyses documented in NS-TMA-2182 (Reference 2), ATWS analyses were performed for the various ANS Condition II events (i.e., Anticipated Transients) considering various Westinghouse PWR configurations applicable at that time. These analyses included 2-, 3-, and 4-Loop PWRs with various steam generator models. For WBN Unit 2, the generic ATWS analyses applicable at that time are those for a 4-Loop PWR with Model D steam generators and a core power of 3411 MWt (NSSS power of 3427 MWt). These conditions are summarized in Table 3-1-a of NS-TMA-2182.

The critical parameters that can affect the results of the ATWS analysis are core power, steam generator type, moderator temperature coefficient, pressurizer safety valve capacity, pressurizer power operated relief valve capacity, and auxiliary feedwater pump capacity. A comparison of these parameters for WBN Unit 2 and those assumed in the generic analysis is presented below.

WBN Unit 2 will be licensed for a power of 3411 MWt and with the original Westinghouse Model D3 steam generators. As delineated below, the analysis was performed for a power of 3475 MWt, and bounds the 3411 MWt level for which Unit 2 is to be licensed.

Although the steam generator type is consistent with that considered in the generic analysis, the core power is higher. The analysis was performed for an NSSS power of 3475 MWt which represents a power increase of 1.4% above that considered in the generic ATWS analysis for the 4-Loop PWRs with Model D steam generators. As documented in NS-TMA-2182, an increase in core thermal power adversely affects the results of the ATWS analyses. As reported for the generic 4-Loop PWR model in Reference 2, an increase in power of 2% increases peak RCS pressure by 44 psi in the limiting loss of load ATWS event. The base peak RCS pressure calculated for the 4-Loop, Model D Loss of Load event is 2780 psia from Reference 2. The resulting pressure, 2824 psia, is well below the acceptance criterion of 3215 psia. This ATWS sensitivity analysis was performed assuming a 2% variation in power, consistent with the typical calorimetric measurement uncertainty on power at the time of these analyses. The increase in power of 1.4% is within the applicable range of the 2% increase in power assumed in the sensitivity analysis since with uprated power the nominal power is increased by 1.4% and the uncertainty is reduced to 0.6% for a total of 2%.

As prescribed by NUREG-0460, the 1979 generic ATWS analyses for Westinghouse PWRs documented in NS-TMA-2182 assumed a full power moderator temperature coefficient (MTC) of -8 pcm/°F. A sensitivity analysis including the use of a MTC of -7 pcm/°F was also

provided as prescribed by NUREG 0460. In 1979, the MTC values of -8 pcm/°F and -7 pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively. The base case of 95% represents a 95% confidence limit on favorable MTC for the fuel cycle. For WBN Unit 2, the Technical Specifications requirement on MTC is limited to $\leq 0\ \Delta k/k$" (0 pcm/°F) at all power levels. Hence, the current MTC Technical Specification for WBN Unit 2 remains the same as that which was applicable for most Westinghouse PWRs in 1979. Additionally, the MTCs for the first cycle of WBN Unit 2 have been confirmed to be equal to or more negative than -8 pcm/°F for 95% of the cycle length. Therefore, the reactivity feedback for WBN Unit 2 remains sufficiently negative to be comparable to the generic Westinghouse ATWS analyses presented in NS-TMA-2182.

The design capacity of each of the three WBN Unit 2 pressurizer safety relief valves is 420,000 lbm/hr. This is consistent with the pressurizer safety valve relief capacity assumed in the 1979 generic ATWS analyses for the Westinghouse 4-Loop plant configuration as documented in NS-TMA-2182.

The WBN Unit 2 pressurizer PORVs are each certified to a maximum flow rate of 210,000 lbm/hr. Therefore, the pressure relief capacities of the PORVs at WBN Unit 2 are consistent with those modeled in the 1979 generic ATWS analyses for the Westinghouse 4-Loop plant configuration as documented in NS-TMA-2182.

The design capacities of the AFW pumps for WBN Unit 2 are shown in Unit 2 FSAR Table 10.4-1 and are as follows:

- Motor-Driven AFW Pump (MDAFW): 450 gpm
- Turbine-Driven AFW Pump (TDAFW): 790 gpm

The WBN Unit 2 AFW system has two Motor-Driven AFW pumps (each pump aligned to 2 steam generators) and one TDAFW pump (aligned to all 4 steam generators). The total design flow for the WBN Unit 2 AFW system (1690 gpm) is therefore a little lower (70 gpm) than the total AFW system capacity of 1760 gpm assumed in the 1979 generic ATWS analyses for the Westinghouse 4-Loop plant configuration as documented in NS-TMA-2182. As documented in NS-TMA-2182, a decrease in AFW flow adversely affects the results of the ATWS analyses. As reported for the generic 4-Loop PWR model in Reference 2, a decrease in AFW flow of 10% increases peak RCS pressure by 12 psi in the limiting loss of load ATWS event. A 10% decrease in flow bounds the 70 gpm delta observed for WBN Unit 2. The addition of 12 psi to the previously calculated pressure of 2824 psia yields a new total of 2836 psia, which is still well below the acceptance criterion of 3215 psia. This ATWS sensitivity analysis was performed assuming a 10% variation in flow

which equates to -176 gpm. The decrease in flow for WBN Unit 2 (70 gpm) is within the applicable range of the sensitivity analysis.

Based on the above, it is concluded that operation of WBN Unit 2 at an NSSS power of 3411 MWt remains within the bounds of the generic Westinghouse ATWS analysis documented in NS-TMA-2182 and, therefore, would remain in compliance with the Final ATWS Rule, 10 CFR 50.62(b).

15.3.6 - 3. *Provide a discussion of the analysis or evaluation showing that the containment safety parameters (e.g., temperature or pressure) will not exceed design limits, with AMSAC installed, for the most limiting ATWS event.*

Response: The containment response to ATWS events is addressed in Appendix D of WCAP-8330 (Reference 1) and Section 7.0 of letter NS-TMA-2182 (Reference 2). It was found that the two limiting ATWS events for containment response, Loss of Feedwater and Accidental Depressurization of the Reactor Coolant System (RCS Depressurization), yielded results that were far below the containment pressure design limit and were much less limiting than the Loss of Coolant Accident.

15.3.6 - 4. *Provide a discussion of the analysis or evaluation showing, with AMSAC installed, that the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," are met for the most limiting ATWS event.*

Response: The ATWS section of the Standard Review Plan, Revision 2, dated March 2007, that discusses 10 CFR 50.46 criteria only applies to evolutionary plant designs (Section 15.8, subsection II.3.C.i), and therefore does not need to be considered for WBN Unit 2.

15.3.6 - 5. *Verify that the moderator temperature coefficient, used in ATWS analyses or evaluations, is consistent with moderator temperature coefficient assumptions used in the bases for 10 CFR 50.62 (see Appendix C to NUREG-0460).*

Response: The use of a statistical MTC value in the WBN Unit 2 ATWS analysis is consistent with previous submittals (References 1, 2, and 3), and with the ATWS analysis guidance documented in References 4 and 5. It was confirmed that the MTCs applicable to the first cycle at WBN Unit 2 will be equal to or more negative than -8 pcm/°F for 95% of the cycle length. This value is consistent with that assumed in the generic ATWS analysis. This value is reconfirmed for every reload at WBN.

References for Responses to **RAIs 15.3.6 - 1.** through **15.3.6 - 6.:**

1. WCAP-8330, "Westinghouse Anticipated Transients without Trip Analysis," August 1974.

2. Letter NS-TMA-2182 from T. M. Anderson of Westinghouse to Dr. Stephen H. Hanauer of the U. S. NRC, "ATWS Submittal," December 30, 1979.

3. WCAP-15831-P-A, Revision 2, "WOG Risk-Informed ATWS Assessment and Licensing Implementation Process," August 2007.

4. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," April 1978.

5. Letter from Roger J. Mattson of the U.S. NRC to Thomas M. Anderson of Westinghouse, February 15, 1979.

6. SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events", W. J. Dircks, July 19, 1983.

15.3.6 - 6. *Provide a discussion indicating that WBN Unit 2 has the capability for long-term shutdown and cooling following the most limiting ATWS event, and identify the operator actions and procedures that are used.*

Response: The Unit 2 procedures have not been approved; they will be based upon the corresponding Unit 1 procedures, and this discussion uses the Unit 1 Emergency Operating Procedures.

The procedure flow is as follows, starting with Emergency Operating Instruction (EOI) E-0, "Reactor Trip Or Safety Injection." Specifically, step 1.0 of Section 3.0 (Operator Actions) of E-0 states:

"Action/Expected Response	Response Not Obtained
1. **ENSURE** reactor trip: • Reactor trip and bypass breakers OPEN. • RPIs at bottom of scale. • Neutron flux DROPPING.	Manually **TRIP** reactor. **IF** reactor will **NOT** trip, **THEN** ** **GO TO** FR-S.1, Nuclear Power Generation / ATWS."

After transitioning to FR-S.1, "Nuclear Power Generation/ATWS," every attempt is made to trip the reactor per the following steps of Section 3.0 (Operator Actions):

- Step 1 includes the following: 1) Manually tripping the reactor; and 2) inserting control rods if the reactor will not trip.
- Step 2 includes the following: Trip the turbine and start Auxiliary Feedwater (AFW).
- Step 4 initiates RCS boration.

NOTE: The actions to trip the turbine and start AFW are also automatically initiated by the AMSAC circuitry on low steam generator water level coincident with turbine load >40%. These actions are mitigative for the most limiting ATWS event, which is an ATWS/Loss of Normal Feedwater Event.

FR-S.1 specifies further actions to prevent pressurizer pressure excursions, isolate dilution paths, and terminate any cooldowns and/or other positive reactivity additions currently in progress.

FR-S.1 continually checks for a reactor trip and/or subcriticality until a transition can be made back to the instruction in effect (usually E-0):

- If Safety Injection (SI) actuated, step 23 of Section 3.0 directs a return to the instruction in effect.
- Otherwise, step 32 of Section 3.0 (last step in the section) directs a return to the instruction in effect.

From there the unit is cooled down and de-pressurized to cold shutdown through normal shutdown procedures. If SI is not actuated, step 4 of Section 3.0 of E-0 specifies two actions depending upon whether SI is required:

- If SI is required and automatic actuation did not occur, E-0 specifies manual actuation of SI. E-0 will diagnose the accident and direct entry into the appropriate procedure to mitigate the accident. The plant is then cooled and de-pressurized to cold shutdown and placed on long-term RHR cooling per normal plant shutdown procedure, GO-6, "Unit Shutdown From Hot Standby to Cold Shutdown."
- If SI is not required, E-0 directs entry into ES-0.1, "Reactor Trip Response."

ES-0.1 provides actions for recovering from the reactor trip. Step 23 of Section 3.0 directs the following actions:

"Action/Expected Response	Response Not Obtained
DETERMINE if natural circulation cooldown is required:	
a. **CHECK** the following:	
1) At least one RCP is available.	1) ** **GO TO** ES-0.2, Natural Circulation Cooldown.
2) Cooldown to Cold Shutdown is desired.	2) ** **GO TO** GO-5, Unit Shutdown From 30% Reactor Power To Hot Standby.
b. ** **GO TO** GO-6, Unit Shutdown From Hot Standby To Cold Shutdown, **OR** GO-5, Unit Shutdown From 30% Reactor Power To Hot Standby, as appropriate."	