September 20, 2010

Mr. Ashok S. Bhatnagar
Senior Vice President
Nuclear Generation Development
and Construction
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING FINAL SAFETY ANALYSIS REPORT RELATED TO SECTION 15 (TAC NO. ME4074)

Dear Mr. Bhatnagar:

By letters dated November 24 and December 14, 2009, and January 11, May 7, and May 27, 2010, Tennessee Valley Authority submitted Amendments 95, 96, 97, 98, and 99, respectively, to the Final Safety Analysis Report for Watts Bar Nuclear Plant, Unit 2.

The U.S. Nuclear Regulatory Commission staff has reviewed the information provided and determined that further information is required to complete its assessment of your submittals. The specific questions are discussed in the enclosed request for additional information.

A response is required within 30 days of receipt of this letter.

If you should have any questions, please contact me at 301-415-1457.

Sincerely,

/RA LRaghavan for/

Patrick Milano, Senior Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosure:
RAI

cc w/encl: Distribution via Listserv

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OFFICIAL AGENCY RECORD

Request for Additional Information

Watts Bar Nuclear Plant, Unit 2

Final Safety Analysis Report Section 15

Tennessee Valley Authority

Docket No. 50-391

By letters dated November 24 and December 14, 2009, and January 11, May 7, and May 27, 2010, Tennessee Valley Authority submitted Amendments 95, 96, 97, 98, and 99, respectively, to the Final Safety Analysis Report (FSAR) for Watts Bar Nuclear Plant (WBN), Unit 2. The Nuclear Regulatory Commission (NRC) staff has reviewed the information provided and has determined that the following information is required to complete its review.

Safety Evaluation Report (SER) 15.0.0

1. FSAR 15.0.0, "Accident Analysis"

   a. FSAR Chapter 15 addresses the accident conditions listed in Table 15-1 of Regulatory Guide (RG) 1.70,[1] 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.

   b. For accident analyses and evaluations in Chapter 15, identify (a) the event classification – Condition II, III, or IV, (b) the specific analysis acceptance criteria to be satisfied for that event, and (c) how each of the criteria are satisfied. If the event is considered to be enveloped by another event, explain the bases and comparisons that underlie the conclusion.

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.

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.

[1] NRC Standard Format and Content Guide, RG 1.70, Revision 3, November 1978

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)?

Safety Evaluation Report (SER) 15.2.0, "Normal Operation and Anticipated Transients"

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

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

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.

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

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

3. FSAR 15.1.3, "Trip Points and Time Delays to Trip Assumed in Accident Analyses"

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

SER 15.2.1, "Loss of Cooling Transients"

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).

2. FSAR 15.2.8, "Loss of Normal Feedwater

   (Note: The plots are barely legible. Supply better quality plots for this transient.)

   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?

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?

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

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?

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?

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.

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?

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

<u>SER 15.2.2, "Increased Cooling Transients"</u>

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.

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.

2. FSAR 15.2.11, “Excessive Load Increase Incident”

   Specify the low limiting value for DNBR.

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?

   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.

   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.

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

   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.

<u>SER 15.2.3, “Change in Inventory Transients”</u>

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?

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

c. Specify the low limiting value for DNBR.

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?

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 a discussion of the analysis or evaluation for such a scenario.

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?
ii. Are the PORV block valves qualified to close against the flow of solid water?
iii. 0What alternative actions are available to the operator, if one or both block valves cannot be closed?
iv. Show that closing the PORV block valves would not lead to opening the safety valves.

d. How does the inadvertent ECCS actuation at power analysis address the concerns raised in Regulatory Information Summary 2005-29[2]?

<u>SER 15.2.4 “Reactivity and Power Distribution Anomalies”</u>

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.

2. FSAR 15.2.3, “Rod Cluster Control Assembly Misalignment”

Provide a legible version of Figure 15.2-11.

[2] NRC Regulatory Issue Summary 2005-29, Anticipated Transients That Could Develop Into More Serious Events, December 14, 2005, ADAMS Accession No. ML051890212

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 from 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)"

1. Post-LOCA Boric Acid Precipitation

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

   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.

2. Small-Break LOCA (SBLOCA)

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

   b. Provide the head flow 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.

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

   d. Provide the results of SBLOCA from a severed injection line. Also, provide the degraded head flow 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.

   e. Explain why the SBLOCA analysis assumed the accumulator liquid volume of 1050 ft^3 compared to the minimum value of 1005 ft^3 in the large-break LOCA (LBLOCA) analysis. Also, confirm the potential negligible impact on the SBLOCA peak clad temperature for this difference in liquid volume.

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

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

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

3. LBLOCA

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

SER 15.3.2, "Major Secondary System Pipe Rupture: Steam line Break"

1. FSAR 15.4.2.1, "Major Secondary System Pipe Rupture: Steam line Break"

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

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.

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

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?

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.

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?

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)?

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.

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?

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?

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).

l. Show all the transient results (plots and sequences of events), for all cases, extending to at least 10 minutes, when manual actions might reasonably be expected to begin. Include plots of transient heat flux.

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)?

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

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 a discussion of the analysis or evaluation to support this statement.

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

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

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

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

4. For the feedwater line break inside containment, what is the effect of the steam environment upon the steam generator (SG) water level measurement?

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

6. How does the SG heat transfer area decrease in relation to the shellside liquid inventory decrease?

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)?

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

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.

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)

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

2. FSAR Section 15.4.4.1 states that "[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?

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.

4. Identify the single failure that was assumed in the analysis.

SER 15.4.3 "Steam Generator Tube Rupture" (SGTR) (FSAR 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.

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?

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.").

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).

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.

6. Show how the analysis results satisfy the analysis acceptance criteria.

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

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 NREG-0800 Section 15.4.8.

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