TVA

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

August 6, 2010

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

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
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**

Reference: 1. Letter from TVA to NRC, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Preliminary Requests for Additional Information and Requests For Additional Information," dated July 31, 2010

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

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

Enclosure 2 provides a minor correction to an RAI response contained in Reference 1.

Enclosure 3 provides the new commitments contained in this letter.

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

D030
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 6th day of August, 2010.

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

Enclosure:

1. Response to RAIs Regarding Unit 2 FSAR
2. Minor Correction to RAI Response Contained in July 31, 2010, Letter to NRC
3. List of New Regulatory Commitments

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 Various Portions of the FSAR [from NRC letter dated 06/23/2010 (ADAMS Accession No. ML101450084)]

Nuclear Performance and Code Review (SNPB)

All references to Watts Bar Unit 1 (WB1) are from the approved UFSAR Amendment 7. All references to Watts Bar Unit 2 (WB2) are from Amendment 95 which is currently under review.

Chapter 4.1

SNPB 4.1 - 1. Why is the density of fuel pellets different between WBN Unit 1 and WBN Unit 2 (compare table 4.4-1 in each FSAR)?

Response: The table that contains the data on the fuel pellets is 4.1-1.

The density of fuel pellets in Unit 2 is the same as the density reported for Unit 1 which is 95% of the theoretical.

Amendment 100 to the Unit 2 FSAR will correct the fuel pellet density (expressed in "% of Theoretical") from "94.5" to "95."

Chapter 4.2.1

SNPB 4.2.1 - 1. Why is the yield strength correlation appropriate for irradiated cladding if the irradiated properties are attained at low exposures and the fuel/clad interactions that lead to minimum margin occur at much higher exposures?

Response: Saturation of increase in irradiated yield strength occurs at low exposures, after which, there is no increase in yield strength with increasing exposure.

SNPB 4.2.1 - 2. In WBN Unit 2 Amendment 95 section 4.2.1.2.2 under the heading 'Guide Thimble and Instrument Tube' (p 4.2-8), should paragraph 3 read "ZIRLO" instead of "Zircaloy"?

Response: Yes. Amendment 100 to the Unit 2 FSAR will replace "Zircaloy" with "ZIRLO" in applicable places.

SNPB 4.2.1 - 4. In WBN Unit 2 Amendment 95 Section 4.2.1.2.2 under the heading 'Grid Assemblies' (p 4.2-9), should paragraph 5 read "ZIRLO" instead of "Zircaloy"?

Response: Yes. Amendment 100 to the Unit 2 FSAR will replace "Zircaloy" with "ZIRLO" in applicable places.

SNPB 4.2.1 - 5. In WBN Unit 2 Amendment 95 Section 4.2.1.3.1 under the heading 'Materials – Fuel Cladding' (p 4.2-10), should paragraph 3 read "ZIRLO" instead of "Zircaloy"?

Response: No, since the reactor tests were done on Zircaloy tubings.

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SNPB 4.2.1 - 6. In WBN Unit 2 Amendment 95 Section 4.2.1.3.2 under the heading 'Stresses and Deflections' (p 4.2-17), should paragraph 1 read "ZIRLO" instead of "Zircaloy"?

Response: Yes. Amendment 100 to the Unit 2 FSAR will replace "Zircaloy" with "ZIRLO" in applicable places.

<u>Chapter 4.2.2</u>

SNPB 4.2.2 - 2. Have the changes to WBN Unit 2 Amendment 95 Table 3.9-5 been reviewed and approved?

Response: In TVA to NRC letter 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], the response to RAI EMCB 3.9.2-3 noted that Amendment 100 to the Unit 2 FSAR will revise Table 3.9-5.

<u>Chapter 4.2.3</u>

SNPB 4.2.3 – 1. Confirm that the burnable absorber rods will be designed so that the absorber material will be maintained below 1492 °F and that the structural elements will be designed to prevent excessive slumping.

Response: The 1,492°F temperature corresponds to the lower limit for excessive slumping of the borosilicate glass burnable absorber material. Since this absorber material will not be used in Unit 2, this number is not applicable.

Amendment 100 to the Unit 2 FSAR will remove any reference associated with the borosilicate glass burnable absorber material.

SNPB 4.2.3 – 4. In WBN Unit 2 Amendment 95 Section 4.2.3.3.1, significant discussion is removed from the FSAR which describes the general methods of analysis, specific methods for analyzing the control rods, and specific methods for analyzing the burnable absorber rods. Provide justification for the reason of this removal, specifically addressing the removal of each paragraph.

Response: Amendment 100 to the Unit 2 FSAR will make the following changes:

- Under 4.2.3.1.3, 4.2.3.2.1, and 4.2.3.3.1, with the subheading Burnable Absorber Rods, the text associated with the borosilicate glass burnable absorber will be removed since this burnable absorber will not be used in Unit 2.

- Under 4.2.3.3.1, the text associated with the B_4C [for example, (14) Gas Generation Pressure] will be removed since this RCCA design will not be used in Unit 2.

SNPB 4.2.3 – 5. In WBN Unit 2 Amendment 95 Section 4.2.3.3.1 (p 4.2-44), should paragraph 1 read "ZIRLO" instead of "Zircaloy"? If so, should the following calculation be performed for ZIRLO instead of Zircaloy?

Response: No. The second full paragraph on the current version of page 4.2-44 (Amendment 99) states, "[I]t is judged that the potential for interference with rod cluster control assembly movement due to unusual local corrosion phenomena of the zircaloy guide thimbles is very low. Operational experience to date and limited PIE data on irradiated thimbles are in support of this data."

Amendment 100 to the Unit 2 FSAR will add the following sentence to this paragraph: "Since ZIRLO® has demonstrated superior corrosion resistance compared to Zircaloy in both autoclave tests and extensive in-reactor irradiation experience, this conclusion is applicable to ZIRLO® guide thimbles as well."

SNPB 4.2.3 – 6. WBN Unit 2 Amendment 95 section 4.2.3.4.1 (p 4.2-52) states:

The rod cluster control assemblies were functionally tested, following initial core loading but prior to criticality to demonstrate reliable operation of the assemblies. Each assembly was operated one time at no flow/cold conditions and one time at full flow/hot conditions. The assemblies were also trip tested at full flow/hot conditions. Those assemblies whose trip times fall outside a certain tolerance were tested an additional 3 times at full flow/hot conditions. Thus each assembly was adequately tested to verify that the assemblies are properly functioning.

WBN Unit 2 Amendment 95 section 4.2.3.4.2 (p 4.2-52) states:

These tests include verification that the trip time achieved by the control rod drive mechanisms meet the design requirement from start of rod cluster control assembly motion to top of dashpot. This trip time requirement was confirmed for each control rod drive mechanism prior to initial reactor operation, as required by Technical Specifications.

Was there an initial core loading for WBN Unit 2 such that this testing was completed? If not, address these two sections.

Response: Amendment 100 to the Unit 2 FSAR will revise the described testing to reflect the following:

There was not an initial core loading for WBN Unit 2. The rod cluster control assemblies will be functionally tested, following initial core loading but prior to criticality to demonstrate reliable operation of the assemblies. Each assembly will be operated four times, once at cold no flow, once at cold full flow, once at hot no flow and hot full flow conditions. Those assemblies whose trip times fall outside a certain tolerance will be tested an additional three times at each failed test condition. Thus each assembly will be adequately tested to verify that the assemblies are properly functioning.

These tests include verification that the trip time achieved by the control rod drive mechanisms meets the design requirement from the time the Reactor Trip Breakers change status until dashpot entry occurs. This trip time requirement will be confirmed for each control rod drive mechanism prior to initial reactor operation, as required by the Technical Specifications (TSs).

Response to RAIs Regarding Unit 2 FSAR

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

RAIs for Various Portions of the FSAR [from NRC letter dated 06/24/2010 (ADAMS Accession No. ML101540250)]

__Nuclear Performance and Code Review (SNPB)__

All references to WBN Unit 1 are from the approved Final Safety Analysis Report (FSAR) Amendment No. 7. All references to WBN Unit 2 are from Amendment No. 95.

SNPB 4.4.2 NRC Information Notice 2009-23 identified that the fuel thermal conductivity experiences a 5-to 7-percent degradation for every 10-gigawatt-days per metric ton of exposure. The thermal conductivity for uranium dioxide provided in equation 4.4-1 does not take this degradation into account. All of the references that are used to generate the thermal conductivity [14, 28, 29, 30, 31, 32, and 33] predate the fuel thermal conductivity experiments performed in 1990 that demonstrate the fuel thermal conductivity degradation effects. Justify the use of equation 4.4-1 given that it will over-predict the fuel thermal conductivity at higher burnups that would lead to an under-prediction of fuel temperatures.

Response: The licensed design models that utilize equation 4.4-1 for fuel thermal conductivity were approved by the NRC in the Safety Evaluation Report (SER) to Topical Report WCAP-15063-P-A, Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)" (TAC No. MA2086). PAD 4.0 was approved by the NRC, and it is documented in Section 4.3 of the SER that an audit calculation was done with the FRAPCON-3 code, which accounts for thermal conductivity degradation. This calculation showed that PAD 4.0 yielded conservative results with respect to the FRAPCON-3 code.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

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

RAIs [taken from NRC letter dated 06/29/2010 (ADAMS Accession No. ML101620006)]:

Nuclear Performance and Code Review (**SNPB**)

SNPB 4.3.2 - 3. In Table 4.3-1 (p 4.3-40) should the Clad Material under the section Fuel Rods read "ZIRLO" instead of "Zircaloy"?

Response: Yes. Amendment 100 to the Unit 2 FSAR will replace "Zircaloy" with "ZIRLO" in applicable places.

SNPB 4.3.2 - 4. Table 4.3-1 (p 4.3-41) in the Rod Cluster Control Assemblies section has information which looks to be carried over from WBN Unit 1, which, is no longer used in WBN Unit 2, such as the information for the boron carbide (B_4C) control rods. According to the table, B_4C control rods will not be used in WBN Unit 2, but all of the parameters are still provided in the table. Correct the table to make it consistent with the Rod Cluster Control Assemblies which will be used in WBN Unit 2.

Response: Amendment 100 to the Unit 2 FSAR will correct this table.

SNPB 4.3.2 - 5. In WBN Unit 2 Amendment No. 95 section 4.3.2.2.4, the definition of axial offset (p 4.3-10) differs from the definition of axial offset from WBN Unit 1 updated FSAR Amendment No.7 section 4.3.2.2.4. Which definition is correct?

Response: Amendment 7 to the Unit 1 UFSAR contained an extraneous character in the definition of axial offset; Amendment 8 corrected this error.

Review of Amendment 8 to the Unit 1 UFSAR and the Amendment 97 version of the Unit 2 FSAR confirms that both use correct and consistent definitions for axial offset.

SNPB 4.3.2 - 6. In WBN Unit 2 Amendment No. 95 section 4.3.2.2.5, show the equation used to determine the average linear power.

Response: The average linear power, or heating rate (ALHR), is determined by:

ALHR = [(Total thermal power (kw)) / (Total active fuel length of all rods (ft))] X HGIF

HGIF is the fraction of the thermal heat generated in the fuel (0.974). Densification is accounted for by use of the densified active fuel length of 143.7 inches.

Using WBN Unit 2 values:

ALHR = [(3411000 kw) / (193 assemblies X 264 fuel rods X 11.975 ft)] X 0.974

ALHR = 5.45 kw/ft

Amendment 100 to the Unit 2 FSAR will add this information to 4.3.2.2.5.

SNPB 4.3.2 - 7. In WBN Unit 2 Amendment No. 95 section 4.3.2.2.5, the total core power is assumed to be limited to 118 percent by a reactor trip, but WBN Unit 1 assumes the total core power to be limited by 121-percent by reactor trip. Provide an explanation for this difference.

Response: The 118% reactor trip is consistent with the safety analyses currently applicable to Unit 2. Unit 1 historical analyses also utilized a 118% reactor trip. Subsequent Unit 1 analyses (circa 1996) resulted in the reactor trip increasing to 121%.

SNPB 4.3.2 - 8. In WBN Unit 2 Amendment No. 95 section 4.3.2.2.6, describe the impacts of using BEACON with fixed incore detectors on the uncertainties listed in this section. Discuss any other impacts of using BEACON with fixed and not moveable incore detectors. Has BEACON been implemented with fixed detectors in other cores?

Response: The determination of the appropriate uncertainty to be applied to peaking factors determined by the BEACON™ Core Monitoring System using fixed incore detectors is explained in WCAP-12472-P-A, Addendum 1-A (rhodium fixed incore detectors) and Addendum 2-A (platinum and vanadium fixed incore detectors). The movable incore detector-based and the fixed incore detector-based implementations of the BEACON Core Monitoring System both use sensor signals arising from fissions in the reactor core to modify a power distribution calculation performed at the reactor conditions to infer the power distribution peaking factors in real time. The major difference in power distribution monitoring based on movable detectors and based on fixed detectors is that the movable detector implementation requires the calibration of the core exit thermocouples, excore detectors, and flux-map determined nodal calibration factors to be performed at intervals not exceeding 180 EFPD from the previous calibration. The fixed incore detector implementation does not require calibration.

As explained in WCAP-12472-P-A, Addendum 1-A and Addendum 2-A (approved by the US NRC in letters dated September 30, 1999, and February 1, 2002, respectively) the peaking factor uncertainty, calculated dynamically and applied to a power distribution surveillance performed by BEACON PDMS, is quantified as a polynomial fit of detector measurement variability and the fraction of inoperable detectors at the time of the surveillance. The coefficients of the polynomial fit are plant dependent and will be developed for WBN Unit 2, prior to startup, using the methodology described in WCAP-12472-P-A, Addendum 1-A. Due to the complexity of the matter, rather than attempting to reproduce the explanation of the methodology presented in Addenda 1-A and 2-A, it is respectfully suggested that the reviewer examine these documents for further details.

The BEACON Core Monitoring System has been implemented using fixed incore detectors at St. Lucie Units 1 and 2 (approximately thirteen reactor-years of BEACON experience for each unit), Temelin Units 1 and 2 (approximately eight and seven reactor-years, respectively, of BEACON experience), and South Ukraine Unit 3 (approximately five reactor-years of operation).

SNPB 4.3.2 - 9. In WBN Unit 2 Amendment No. 95 section 4.3.2.3.2, when the moderator coefficient is calculated for the various plant conditions, is the moderator temperature varied by adding 5 °F to each of the mean temperatures, or by adding and subtracting 5 °F to each of the mean temperature (p 4.3-17 – 4.3-18)?

Response: The mean temperature is varied by adding and subtracting 5 °F.

Amendment 100 to the Unit 2 FSAR will replace "+ 5 °F" with "± 5 °F" in the opening sentence of the fifth paragraph of 4.3.2.3.2.

<u>Chapter 4.3.3</u>

SNPB 4.4.1 - 2. In WBN Unit 2 Amendment No. 95, Table 4.4-1 is inconsistent with Table 4.1-1. Why are the tables inconsistent?

Response: Table 4.1-1 is a summary table. Amendment 95 to the Unit 2 FSAR updated Table 4.4-1; however, Table 4.1-1 was not updated.

Amendment 99 to the Unit 2 FSAR updated Table 4.1-1 to correct the inconsistencies. A review of the current version (Amendment 99) of Table 4.1-1 shows the following typos:

- Item 6: delete the greater than sign.
- Item 8: the exponent should be "6" instead of "5."
- Item 24: less than sign is missing before "3290."

Amendment 100 to the Unit 2 FSAR will correct these errors.

<u>Chapter 4.4.2</u>

SNPB 4.4.2 - 4. In WBN Unit 2 Amendment No. 95 section 4.4.2.5, the conclusion is drawn that the minimum departure from nucleate boiling ratio in the hot channel is relatively insensitive to variations in void models. This conclusion is based on a sensitivity study using the THINC-IV code (which is Reference 52 of the FSAR). The THINC-IV sensitivity study (Section 5.5 in Reference 52) uses void models that will be used in VIPRE. What, then, is the basis for assuming that the sensitivity study remains applicable to the VIPRE-01 code?

Response: Studies using VIPRE-W (Reference 100 in the FSAR) have confirmed similar behavior to THINC-IV. Therefore, the THINC-IV sensitivity study is applied to the VIPRE-W code.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

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SNPB 4.4.2 - 5.1 - 2. Additionally, the changes to this page were not captured and the page is marked 'WBNP-73' which signifies it is from Amendment No. 73 and it is not.

Response: What appears to be a change was actually an error introduced in Amendment 95 on page 4.4-18. Text boxes were being used within the word processing software (Adobe Framemaker) to uniquely identify equations within Section 4.4.2. For some reason, the text box for equation 4.4-15 was erroneously duplicated onto page 4.4-18 during Amendment 95 preparation. This error was subsequently corrected in Amendment 98.

<u>**Chapter 4.4.3**</u>

SNPB 4.4.3 - 1. In WBN Unit 2 Amendment No. 95 section 4.4.3.1.3, clarify the first sentence which references the 'VIPRE-01 THINC-IV computer code'.

Response: The word 'THINCIV' was intended to be deleted. Amendment 98 to the Unit 2 FSAR deleted this word.

SNPB 4.4.3 - 2. In WBN Unit 2 Amendment No. 95 section 4.4.3.2.1 on page 4.4-23, the definition of '$F^{N}_{\Delta H}$' should be the definition of '$F^{RTP}_{\Delta H}$'.

Response: Yes. When defining the terms used in equation 4.4-20, "$F^{N}_{\Delta H}$" should be "$F^{RTP}_{\Delta H}$."

Amendment 100 to the Unit 2 FSAR will make this correction.

<u>**Chapter 4.4.5**</u>

<u>**Quality Assurance RAI**</u>

SNPB QA - 1. The NRC staff has identified multiple inconsistencies, discrepancies, and factual errors in review of WBN Unit 2 Amendment No. 95. Most of the errors identified were inconsistencies (referring to Zircaloy instead of ZIRLO), but some were quite substantial (references to both Ag-In-Cd control rods and B_4C with Ag-In-Cd tips, reference to startup testing performed on the rod cluster control assemblies of WBN Unit 2, discrepancies between Table 4.4.1 and Table 4.1.1). TVA submitted Amendment No. 98 to correct the errors in Amendment Nos. 95 and 97, but none of substantial errors identified by the NRC staff were identified or corrected by Amendment No. 98. Please describe the quality control and assurance process applied to the information contained in Amendment No. 95 (and Amendment No. 98). Explain what assurance TVA can provide the NRC staff that the information contained in the Amendments is factually correct, given the number and magnitude of identified discrepancies and errors identified by the NRC staff. If left uncorrected (especially the discrepancies between Table 4.4.1 and Table 4.1.1) what would the impact of the errors be?

Response: The issue of inaccurate FSAR submittals was entered into the TVA Corrective Action Program. In response, TVA has performed a complete review of the entire FSAR through Amendment 98 using a team of approximately 50 engineers. The review concentrated on the following areas:

1) Technical accuracy/consistency of the text, figures and tables – A review was performed to identify and assess technical discrepancies between Unit 2 design documents (EDCRs, drawings, calculations, system descriptions, design criteria etc.) and the Unit 2 FSAR. Discipline and licensing engineers reconciled the discrepancies, making changes to the FSAR, as appropriate. A review was performed to ensure the information in the final typed version of the FSAR matched the markups provided to the clerical staff. This review was performed by the engineering staff that prepared the initial markups. Previously, the markups were provided to the clerical staff and a typing check was performed administratively but the loop was not closed with the engineers that prepared the markups. By providing this feedback loop, the engineers were able to identify markups that were not correctly incorporated, partially incorporated, or missed in the clerical processing. This effort has improved the integrity of the final product.

2) Administrative consistency – A review was performed to ensure administrative errors were corrected such as section/figure/table/reference numbers, typos introduced through word processing, superfluous information, etc. One source of errors and confusion arose from the manner in which revisions were being marked. In Amendment 98 and prior amendments, TVA attempted to mark each revision and maintain the latest Amendment number that affected that page at the top of the page. However, the amendment number was in a word processing header that did not stay connected to the text and revision bars below it. If text was added on a previous page, the text would roll down but the header would not thus creating a discrepancy between the revision bars and the page headers. Amendment 99 and later revisions will utilize a different marking scheme. The amendment number on a page is based on if anything changed in the entire section including subsections (e.g., Section 1.1, Subsection 1.1.1, etc.).

The above review results were corrected in Amendments 99 (issued) and forthcoming Amendment 100.

ENCLOSURE 1

Response to RAIs Regarding Unit 2 FSAR

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

Preliminary RAIs [taken from NRC letter dated 07/32/2010 (ADAMS Accession No. ML102020281):

FSAR Section 3.11

3.11 - 6 On page 3.11-5 of the FSAR, the licensee stated the following:

> Doses were integrated to determine an equipment exposure for a 100-day period after the accident.

Explain the basis for reducing the period of exposure after an accident from one year to 100 days.

Response: As part of the design and licensing of Unit 1, the Unit 1 Environmental Qualification (EQ) 10 CFR 50.49 Program post accident operating time was determined to be 100 days. The 100-day post accident operating time was incorporated in the Unit 1 EQ program and was reviewed and accepted by the NRC in the Unit 1 SSER 15 review of WBN EQ Program Documentation Files. Consequently, the appropriate exposure time for post accident radiation analyses is 100 days. The Unit 2 EQ program is patterned after the Unit 1 EQ program and also uses a 100-day post accident operating time. Therefore, Unit 2 develops integrated doses for a 100-day time period. NRC Regulatory Guide (RG) 1.89, Appendix D, indicates the accident exposure time should be based on the maximum operating time of the equipment and that the one year post accident exposure period given in the RG is for illustration only.

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RAIs for FSAR 3.9.3 [from NRC letter dated 07/30/2010 (ADAMS Accession No. ML101950610)]

EMCB 3.9.3-1 Provide bases or justification for deleting the following valves from Table 3.9-17, "ACTIVE VALVES FOR PRIMARY FLUID SYSTEMS," and Table 3.9-25, "VALVES REQUIRED TO BE ACTIVE FOR DESIGN BASIS EVENTS."

From Table 3.9-17: Valves RFV-62-518, RFV-62-519, 1-RFV-62-1220, CKV-77-849, and CKV-77-868.

From Table 3.9-25: Valves CKV-18-686A/B, CKV-18-687A/B, RFV-18-688A/B, CKV-18-689A/B, CKV-18-787A/B, 1-33-794,2-33-797, FCV-62-1228, FCV-62-1229, CKV-67-513A/B, FCV-70-168, CKV-70-753, FCV-82-310, FSV-82-310-S, FCV-82-311, FSV-82-311-S, PCV-82-312A/B, PCV-82-313A/B, CKV-82-561, CKV-82-562, CKV-82-567, CKV-82-568, CKV-82-572, CKV-82-573, CKV-82-589, CKV-82-590, CKV-82-591, CKV-82-592, CKV-82-593, SPV-82-593, SPV-82-594, FCV-90-108, and FCV-90-109.

Response: The identified valves (with corrections noted below) were deleted from the FSAR via Amendment 97 (Change Package 2-97-67). The bases/justification for deleting these valves from FSAR Tables 3.9-17 and 3.9-25 are as follows:

<u>RFV-62-518 and CKV-62-519 (identified in question as RFV-62-519)</u>:

Valves RFV-62-518 and CKV-62-519 are associated with the Reciprocating Charging Pump which has been abandoned in place in Unit 1. The Reciprocating Charging Pump is also being abandoned in place in Unit 2. Valves 2-CKV-062-0519 and 2-RFV-062-0518 were determined to be missing in the field and will not be reinstalled.

<u>1-RFV-62-1220</u>:

Valve 1-RFV-62-1220 is associated with the Unit 1 Reciprocating Charging Pump which has been abandoned in place. The Reciprocating Charging Pump is also being abandoned in place in Unit 2. The corresponding Unit 2 valve, 2-RFV-62-1220, was never installed.

<u>CKV-77-849 and CKV-77-868</u>:

Valves CKV-77-849 and CKV-77-868 were retagged as CKV-68-849 and CKV-63-868, respectively. FSAR Table 3.9-17 was revised to remove the old valve numbers (CKV-77-849 and CKV-77-868) and to add the new valve numbers (CKV-68-849 and CKV-63-868).

CKV-18-686A/B, CKV-18-867A/B, RFV-18-688A/B, CKV-18-689A/B and CKV-18-787A/B:

Valves CKV-18-686A/B, CKV-18-867A/B, RFV-18-688A/B, CKV-18-689A/B and CKV-18-787A/B are valves associated with the fuel oil supply for the additional diesel generator. These valves were removed from FSAR Table 3.9-25 since the additional diesel generator has never been placed in service for Unit 1, and it is not currently in the Unit 2 design basis.

1-33-794 and 2-33-797:

Normally closed check valves 1-33-794 and 2-33-797 have no active function during a design basis event. These valves do not provide a containment isolation function for penetration X-40D. The lines through penetration X-40D are isolated during power operations by a blind flange with a dual O-ring type seal installed outside containment, thereby, providing a double barrier against containment leakage.

FCV-62-1228 and FCV-62-1229:

Valves FCV-62-1228 and FCV-62-1229 appeared on both Table 3.9-17 and Table 3.9-25. These valves were removed from FSAR Table 3.9-25 to avoid redundant information. FSAR Table 3.9-17 was considered the more appropriate table to list these valves.

CKV-67-513A/B:

Normally closed check valves CKV-67-513A and CKV-67-513B have no active function during a design basis event. These valves are located in the normally isolated redundant backup flow path of ERCW to the diesel generator jacket water heat exchangers from the opposite train of cooling water.

FCV-70-168 and 0-CKV-70-753 (identified in question as CKV-70-753):

Valves FCV-70-168 and 0-CKV-70-753 have no active function during a design basis event. Valve FCV-70-168 is being removed as part of the abandonment of component cooling to the Unit 2 Gas Stripper and Boric Acid Evaporator Package B. Valve CKV-70-753 is located in the isolated component cooling return line from the Unit 0 Condensate Demineralizer Waste Evaporator which is abandoned in place.

<u>FCV-82-310, FSV-82-310-S, FCV-82-311, FSV-82-311-S, PCV-82-312A/B, PCV-82-313A/B, CKV-82-561, CKV-82-562, CKV-82-567, CKV-82-568, CKV-82-572, CKV-82-573, CKV-82-589, CKV-82-590, CKV-82-591, CKV-82-592, SPV-82-593 and SPV-82-594</u>:

Valves FCV-82-310, FSV-82-310-S, FCV-82-311, FSV-82-311-S, PCV-82-312A/B, PCV-82-313A/B, CKV-82-561, CKV-82-561, CKV-82-562, CKV-82-567, CKV-82-568, CKV-82-572, CKV-82-573, CKV-82-589, CKV-82-590, CKV-82-591, CKV-82-592, SPV-82-593 and SPV-82-594 are associated with the starting air for the additional diesel generator. These valves were removed from FSAR Table 3.9-25 since the additional diesel generator has never been placed in service for Unit 1, and it is not currently in the Unit 2 design basis.

<u>CKV-82-593</u>:

A valve with the number CKV-82-593 does not exist and was not included on or removed from FSAR Table 3.9-25. It is believed that the question was referring to SPV-82-593 which is addressed above.

<u>FCV-90-108 and FCV-90-109</u>:

Valves FCV-90-108 and FCV-90-109 remain on FSAR Table 3.9-25. These valves were listed on the table twice and the table has been revised to only list them once.

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 3.9.3 [from NRC letter dated 07/30/2010 (ADAMS Accession No. ML101940474)]

FSAR 10.4.7

No number

In Amendment No. 75 dated March 4, 2009 (Agencywide Documents Access and Management System Accession No. ML090480566), the Nuclear Regulatory Commission (NRC) approved a revision to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," for Watts Bar Nuclear Plant (WBN) Unit 1. Specifically, this amendment revised the requirements for the auxiliary feedwater system (AFW) auto-start function associated with the trip of the turbine-driven main feedwater (TDMFW) pumps.

The ESFAS circuitry provides an automatic start of the AFW pumps in the event both TDMFW pumps trip. The proposed Final Safety Analysis Report (FSAR) Section 10.4.7 for WBN Unit 2 does not provide a description of this logic. However, FSAR Chapter 15 states that in order to provide the necessary protection against the loss of normal feedwater, the motor-driven AFW and turbine-driven AFW pumps will start automatically upon a trip of both TDMFW pumps.

WBN Units 1 and 2 do not have a block of the automatic AFW pump start circuit upon a trip of both TDMFW pumps to use during startup of the first TDMFW pump. In WBN Unit 1 Amendment 75, Mode 2 applicability for this function added a note stating: "When one or more Turbine Driven Feedwater Pump(s) are supplying feedwater to steam generators."

Discuss whether the changes made to TS 3.3.2 for WBN Unit 1 in Amendment 75 will be made in the proposed FSAR and TSs for Unit 2. If not, provide detailed information regarding the logic to provide an automatic start of the AFW pumps and how that logic is controlled during a plant startup.

Response: As stated above, Amendment 75 to the Unit 1 TS was approved by NRC to modify the requirement related to Mode 1 and 2 Applicability for Function 6.e of TS Table 3.3.2-1 (ESFAS Instrumentation). Additionally, Limiting Condition for Operation 3.3.2, Condition J, was revised to be consistent with the design basis for Unit 1.

Amendment 75 to the Unit 1 TS was incorporated into Developmental Revision B of the Unit 2 TS (submitted February 2, 2010).

Amendment 100 to the Unit 2 FSAR will revise the Unit 2 FSAR based upon Amendment 75 changes made to the Unit 1 UFSAR.

ENCLOSURE 2

Minor Correction to RAI Response Contained in July 31, 2010, Letter to NRC

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

RAIs for FSAR Section 9.1 - SBPB [taken from NRC letter dated 07/07/2010 (ADAMS Accession No. ML101620047)]:

Background:

Many of the fuel storage and handling related structures, systems, and components within the WBN Auxiliary Building are shared between the two units, including the spent fuel pool, the spent fuel cooling and cleanup system, and the spent fuel handling equipment. The WBN Unit 2 FSAR describes the degree of conformance with the NRC General Design Criteria (GDC) of Title 10, Code of Federal Regulations (10 CFR) Part 50, Appendix A. The ability of shared systems to perform their safety functions for credible combinations of normal and accident states is addressed in GDC 5. Pursuant to the requirements of 10 CFR 50.34(b), applicants for operating licenses must include in the FSAR a description and analysis of the structures, systems, and components of the facility, and the evaluations required to show that safety functions will be accomplished.

SBPB 9.1 - 2. Section 9.1.3.3.3, "Pool and Fuel Temperatures," of the WBN Unit 2 FSAR describes that, with a 12 day decay time, the maximum heat load associated with a full core discharge is 28.1E+06 Btu/hr while the maximum heat load for a full core discharge following a normal refueling outage case is 32.6E+06 Btu/hr. This statement is essentially identical to the corresponding Section of the WBN 1 Updated Safety Analysis Report, which was potentially based solely on operation of WBN Unit 1. Since the operation of a second unit would increase the frequency of fuel discharges to the spent fuel pool, the heat load values may not be representative of dual-unit operating conditions. Confirm the expected heat loads for representative dual-unit scenarios and describe the methodology, including decay heat models, used to determine the heat load.

Response: **Response from July 31, 2010:**

"The expected heat loads for dual-unit operating conditions with the current installed SFP capacity of 1386 locations completely filled are as follows:

- 12-day decay time, full core discharge: 39.06E+06 BTU/hr
- Full core discharge following normal outage case: 25.62E+06 BTU/hr

Amendment 100 to the Unit 2 FSAR will update the FSAR with these values.

Methodology Discussion

These spent fuel pool (SFP) decay heat loads are calculated in accordance with ANS Standard 5.1, "Decay Heat Power in Light Water Reactors," and NRC RG-3.54, "Spent Fuel Heat Generation in an Independent Spent Fuel Pool Storage Installation."

Normal Offload Conditions

The SFP will reach an equilibrium situation, alternating refueling offloads between the two units every 180 and 355 days. In this analysis, the pool is filled and the decay heat is calculated based on this offload schedule, beginning with Unit 1. To fill the pool to capacity, an additional 154 assemblies were given the same age as the initial Unit 1 batch and added to the pool. The total decay heat in the SFP at full pool conditions is obtained by adding the heat load from each cycle (referred to as the background heat load) to the total decay heat produced by the most recently offloaded core (referred to as the final offload).

The decay heat produced by the final offload and the background decay heat is computed in accordance with the methodology described above. All data used accounts for additional decay heat due to TPBARs. For each offload, the number of fuel assemblies was also taken into account. Unit 1 offloads are assumed to be 96 assemblies, and Unit 2 offloads are assumed to be 80 assemblies.

Emergency Offload Conditions

For the emergency offload scenario, the background heat load calculation changes slightly. Current Unit 1 licensing basis requires 36 days to elapse after the first unit shutdown before the second unit is shutdown; a period of no greater than 60 days is then allowed to elapse before completion of the second (emergency) core offload. Thus, the decay time for the most recent "normal" offloaded fuel is 96 days. This, along with the slight change in fuel age for the balance of the pool, is factored into the determination of the background heat load (computed as above in the Normal Offload Scenario). The decay heat for the emergency offload core is then taken at 60 days decay. Therefore, the total Emergency Offload decay heat is the sum of the "background" decay heat plus the most recently normal discharge batch (96 assemblies decayed for 96 days) plus the emergency offload (193 assemblies decayed for 60 days).

Correction

The second bulleted item from the previous response should have read "Full core discharge following normal outage case: 25.6**1**E+06 BTU/hr" instead of "Full core discharge following normal outage case: 25.6**2**E+06 BTU/hr."

Amendment 100 to the Unit 2 FSAR will correct this.

ENCLOSURE 3

List of New Regulatory Commitments

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

1. Amendment 100 to the Unit 2 FSAR will implement changes as noted in the applicable preliminary RAI / RAI responses.