WBN2Public Resource

From: Boyd, Desiree L [dlboyd@tva.gov]
Sent: Monday, August 02, 2010 2:32 PM
To: Wiebe, Joel; Raghavan, Rags; Milano, Patrick; Campbell, Stephen
Cc: Crouch, William D; Arent, Gordon; Hamill, Carol L
Subject: FSAR RAI Responses Submittal_ 7-31-10
Attachments: FSAR RAI Responses Submittal_ 7-31-10_NRC copy.pdf

Please see attached letter sent to the NRC today.

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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 Preliminary Requests for Additional Information and Requests For Additional Information**

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

Enclosure 1 provides the response to preliminary RAIs and RAIs involving multiple FSAR chapters.

Enclosure 2 provides the new commitments contained in this letter.

The electronic files of documents noted as being provided by the response to specific RAIs are contained on the enclosed Optical Storage Media (OSM). Enclosure 3 lists the electronic files and the file sizes.

Attachments to Enclosure 3 contain information proprietary to various companies as denoted in specific Attachments. TVA requests that this vendor proprietary information be withheld from public disclosure in accordance with 10 CFR § 2.390.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 31st day of July, 2010.

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

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

Enclosures:

1. Response to Preliminary Requests for Additional Information
2. List of Regulatory Commitments
3. List of Files Provided on Enclosed Optical Storage Media (OSM)

cc (Enclosures):

U. S. Nuclear Regulatory Commission
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NRC Resident Inspector Unit 2
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ENCLOSURE 1

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

Preliminary RAIs (taken from NRC phone call of 05/18):

3.5.1.3.6 - 1. In section 3.5.1.3.6, a sentence was deleted that started like "This analysis…." The deletion of a reference to an analysis seems to the NRC to be a technical change. The change was not reflected in Enclosure 1 as a technical change, and the NRC is questioning why it would not have been included.

Response: This item was previously addressed as the response to RAI 3.5.1-2 on page E1-7 of TVA letter to NRC dated June 3, 2010 (ADAMS Accession No. ML1016004770) as follows:

"2. On page 3.5.22, second sentence of the first paragraph stated: "In this analysis, only Unit 2 containment and the control building appear to be prominent enough to be threatened.

Please discuss how these threats will be addressed and indicate what section of the FSAR addresses this issue?

Response: Amendment 98 deleted the sentence since it was an intermediate issue and not the conclusion of the analysis."

Prior to submittal of Amendment 98 to the Unit 2 FSAR, it was decided by a telecom with the NRC reviewer that this sentence would be removed since it created confusion. TVA chose to delete it in Amendment 98 as an editorial change (The conclusions of the analysis remain as shown in the FSAR.), and chose not to flag it as a change in the cover letter since the explanation was going to be in the RAI response noted above.

3.2-2 - 2. The following questions refer to Table 3.2-2:

a. On page 3.2-27, an asterisk was eliminated from the reference to TVA Class C. Was this a technical change or the deletion of an unused asterisk?

Response: Amendment 98 to the Unit 2 FSAR replaced "TVA Class C*" with "TVA Class C" in Note (18) for Table 3.2-2.

This change was implemented to make this designation consistent with the associated TVA Design Criteria and the definition of vendor-supplied equipment as defined in Note 1) of Table 3.2-4. Thus, this was considered to be an editorial change.

b. The description for "Note 22" says it is not used, but the reference in the safety class column for the Unit 1 PWST says changed to "Note 25."

Response: Amendment 100 to the Unit 2 FSAR will re-instate Note 22 which applies only to the Unit 1 Primary Water Storage Tank. Note 26 applies to the Unit 2 Primary Water Storage Tank. Two separate notes are required to identify the differences between the Unit 1 and Unit 2 tanks.

ENCLOSURE 1

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

Preliminary RAIs for FSAR 3.10 (taken from e-mail from NRC dated 05/04/2010)

<u>Section 3.10</u>

3.10 - 1. Reference is made to IEEE 344–1987 in Tables 3.10-1 and 3.10-2. However, IEEE 344-1987 is not mentioned in Section 3.10.1, “Seismic Qualification Criteria.” Nor is this discussed in the referenced Section 3.7.3.16.

Clarify how IEEE-1987 is used in a similar manner to how you discuss the use of IEEE 344-1971 and IEEE 344-1975.

Response: Amendment 98 to the Unit 2 FSAR deleted equipment heading “PAMS Cabinet and Components and Main Control Room Components” and its reference to IEEE 344-1987 from Table 3.10-1.

Table 3.10-2 (Qualification Of Instrumentation And Control Equipment) lists Qualification Method 9 as IEEE 344-1987. Qualification Method 9 was incorrectly added at Amendment 95. There is no reference to it in the Table 3.10-2 equipment listing in Amendment 95 or any subsequent amendment.

Amendment 100 to the Unit 2 FSAR will delete Qualification Method 9, IEEE 344-1987 from Table 3.10-2.

3.10 - 2. Table 3.10.1, “WBNP Instrumentation and Electrical Equipment,” in WBN-2 FSAR Section 3.10 contains three new rows related to certain equipment and their qualification methods and test methods. The first new row in Table 3.10.1 states that the “Control Instrument Loops” (Unit 2) located at “multiple locations” were qualified by “Test” using “multiaxis” test method performed by “Nuclear Qualification Services.”

Clarify if the “Test” method and the “Test” results were reviewed by the NRC staff and provide a reference that documents the review conclusion. If they were not reviewed by the NRC staff, submit the results of the test for the staff’s review.

The second new row in Table 3.10.1 states that “Panels 2-L-11A and 2-L-11B” were qualified by “Analysis.”

1) Clarify if the Analysis mentioned in the second new row in table 3.10.1 was performed in-house by the TVA staff; if not, complete the Table 3.10.1 giving the name of the company, which performed the Analysis.

2) Also, clarify if the “Analysis” method and the Analysis results were reviewed by the NRC staff and provide a reference that documents the review conclusion. If they were not reviewed by the NRC staff, submit the results of Analysis for the staff’s review.

3) The third new row in Table 3.10.1 states that the qualification method for the equipment (PAMS Cabinet and Components and Main Control Room Components) is “Analysis (to be performed).”

Provide a target date when this analysis will be performed, submit the results of the analysis for the staff's review, and amend the FSAR as needed.

Response: Foxboro Spec 200 Instruments

TVA does not does not know if the test results have been reviewed by the NRC staff. This hardware is widely used in multiple nuclear facilities and may have been reviewed previously. To ensure compliance, the requested documentation is provided as Attachment 1.

1) and 2) The second new row in Table 3.10.1 states that "Panels 2-L-11A and 2-L-11B" were qualified by "Analysis." The analysis is being performed in house, and the analysis results have not been reviewed by the NRC staff. The analysis will be submitted to the NRC by November 30, 2010.

3) Amendment 99 to the Unit 2 FSAR removed this item from Table 3.10-1. Due to hardware changes, the qualification will be by analysis and testing. The vendor is scheduled to provide this documentation to TVA December 27, 2010, and it will be submitted to the NRC by January 14, 2011.

3.10 - 3. In several locations in FSAR Section 3.10 (e.g., pages 3.10-11, 3.10-12, and 3.10-18), the word "LATER" is inserted before a Reference or a report.

If this word LATER refers to future action, provide a target date to provide these reports and the results of the qualification Tests / Analysis included in these reports for the staff's review.

Response: The word LATER is used for the following references:

(26) Westinghouse seismic qualification report for installing Gamma Metrics hardware in Unit 2 NIS cabinets.

This item is EQ-EV-39-WBT, Revision 1 (Seismic Evaluation Of Nuclear Instrumentation System Console 2-M-13 With Gammametrics Equipment For Watts Bar Unit 2, March 2009). The proprietary version of this document is provided as Attachment 2. A non-proprietary version and affidavit for withholding will be provided by November 30, 2010. Amendment 100 to the Unit 2 FSAR will reflect this information.

(29) Ametek seismic qualification report for containment pressure transmitters.

This item is vendor document number Report No. TR-1136 (Qualification Documentation Review Package For Ametek Aerospace Gulton-Statham Products Nuclear Qualified Pressure Transmitter Series Enveloping --- Gage Pressure Transmitter Series PG 3200, Differential Pressure Transmitter Series PD 3200, Differential High Pressure Transmitter Series PDH 3200, Draft Range Pressure Transmitter Series DR 3200, Remote Diaphragm Seal Differential Pressure Transmitter Series PO 3218, Remote Diaphragm Seal Differential High Pressure Transmitter Series PDH 3218). The proprietary version of the document is provided as Attachment 3. A non-proprietary version and affidavit for withholding will be provided by December 17, 2010. Amendment 100 to the Unit 2 FSAR will reflect this information.

(30) Seismic Qualification of Weed Pressure Transmitter.

This item is vendor document number 16690-QTR, Revision 0 (Qualification Test Report For Environmental And Seismic Qualification Of Weed Model DTN2010 Pressure Transmitters). The proprietary version of this document is provided as Attachment 4. A non-proprietary version and affidavit for withholding will be provided by November 30, 2010. Amendment 100 to the Unit 2 FSAR will reflect this information.

3.10 - 4. The numbering of the Unit 2 list on page 3.10-4 is not consistent with the numbering referenced by the text below the list.

1) Correct the numbering to clearly identify the references associated with the items in the list.

Response: Amendment 98 to the Unit 2 FSAR corrected the numbering of the list. Since these were editorial changes, the amendment level remained the same.

2) The Nuclear Instrumentation System Power Range Electronics appears to be a new item added to the list for Unit 2. Clarify which reference documents its qualification testing. Provide the results of the test or analysis.

Response: Amendment 98 to the Unit 2 FSAR corrected the numbering of the list. Since these were editorial changes, the amendment level remained the same.

There is no change to the Power Range Electronics. The electronics are the same and use the same qualification documents as Unit 1. As a result, no qualification documents are submitted for the electronics.

Amendment 95 to the Unit 2 FSAR added the Nuclear Instrumentation System Power Range Electronics. This was done to differentiate the qualification of the <u>cabinets</u> from the <u>electronics</u>. In Unit 2, Westinghouse updated the cabinet qualification to support the installation of the Gamma Metrics hardware. In Unit 1, the cabinet qualification analysis was done by TVA. Having Westinghouse perform the Unit 2 analysis resulted in Reference 26 being added to the reference section. However, Reference 26 was inadvertently omitted from the 3.10.1 text discussion of the equipment qualified by Westinghouse.

Amendment 100 to the Unit 2 FSAR will update the FSAR wording as shown below:

"Seismic qualification testing/analysis of Items 1 through 9 is documented in References [1] through [10] and [26]. Reference [10] presents the theory and practice, as well as justification, for the use of single axis sine beat test inputs used in the seismic qualification of electrical equipment. In addition, it is noted that Westinghouse has conducted a seismic qualification "Demonstration Test Program" (reference Letter NS-CE-692, C. Eicheldinger (W), to D. B. Vassallo (NRC), 7/10/75) to confirm equipment operability during a seismic event. This program is documented in References [12] through [14] (Proprietary) and References [16] through [19] (Non-Proprietary). Seismic qualification testing of Item 10 to IEEE 344-1975 is documented in References [21], [22], [23], [31] and [32]. Reference [26] documents the Westinghouse qualification by analysis of the Nuclear Instrumentation System cabinet 2-M-13 with Gamma Metrics Source and Intermediate Range hardware installed."

The proprietary version of Reference 26, Westinghouse report EQ-EV-39-WBT is provided as Attachment 2. The non-proprietary version and the affidavit for withholding will be submitted by November 30, 2010.

3.10 - 5. Page No. 3.10-4 in FSAR Section 3.10.1 lists several new items of instrumentation and electrical equipment requiring seismic qualification. It is stated on page 3.10-6 of Section 3.10.1 that seismic qualification testing of items 11 and 12 is documented in a new Reference 25. Provide a copy of Reference 25 for the staff's review.

Response: The proprietary version of Thermo Fisher Scientific Qualification Report No. 864, Rev. 0 (Class 1E Qualification of Source Range, Intermediate Range and Wide Range Channels) is provided as Attachment 5. Note that this proprietary version contains some marked-up editorial corrections for cross reference information. A corrected proprietary version, the non-proprietary version, and an affidavit for withholding will be provided by November 15, 2010.

Preliminary RAIs for FSAR Sections 5.2.3 through 6.1.1 (taken from e-mail from NRC dated 05/13/2010)

5.2.3-1 <u>Background</u>

In FSAR Section 5.2.3.4, Chemistry of Reactor Coolant, the applicant added a description of the process for zinc addition. The applicant indicated that zinc would be added for the purpose of reducing radionuclide content in the primary system corrosion films, and that the residual zinc content would be maintained at a concentration of 2-8 parts per billion (ppb). FSAR Table 5.2-10 also indicates zinc will be limited to less than 40 ppb during normal power operation.

The staff reviewed several reports documenting industry experience with zinc addition in Pressurized Water Reactors (PWR's), which indicate that there is no concern with crud deposition for plants with low-duty or medium-duty cores (Reference 1, 2), and, in fact, zinc addition typically leads to thinner, more evenly distributed crud on fuel. However, there is currently insufficient operating experience with zinc addition in plants with high-duty cores to be able to conclude that zinc injection would not cause a problem with crud deposition in such plants. Core duty is a measure of the amount of subcooled nucleate boiling (SNB) occurring in the core. Plants with high-duty cores are those with high fluid temperatures and high surface heat flux at the fuel clad causing a portion of the total heat transfer to the coolant to occur by SNB. Although favorable for thermal efficiency, the combination of high temperature and SNB leads to more surface boiling, which is known to enhance the formation of corrosion product deposits (crud) at the cladding surface. The tendency for SNB can be quantified by means of the High Duty Core Index (HDCI), calculated in accordance with Appendix F of Reference 3. Cores with an HDCI of ≥ 150 are considered to be high duty plants, medium duty plants have HDCI of 120-149, and a plant with HDCI ≤ 119 is considered a low-duty plant. Staff calculations based on thermal-hydraulic data from FSAR Chapter 4 indicate the WBNP-2 core may be considered high-duty. There may be alternate methods to determine the amount of SNB other the HDCI, such as detailed thermal hydraulic computer models.

Potential problems with crud deposition could include excessively thick fuel crud, or uneven crud thickness that could lead to crud induced power shift (CIPS), also known as axial offset anomaly. Reference 2 also indicates that fuel clad corrosion cannot be completely ruled out for high-duty cores exposed to zinc addition even though no problems have been observed to date. Reference 2 recommends a fuel surveillance program for high-duty plants implementing zinc addition.

<u>Requested Information</u>

1. Is the WBNP core design considered a high-duty core when the HDCI is calculated in accordance with Appendix F of Reference 3, or an alternate method of evaluation?

Response: Westinghouse does not recommend the use of HDCI as a measure of plant or core "duty." The formulation is overly sensitive to outlet temperature.

Using the bounding values listed in the FSAR, Watts Bar Unit 2 HDCI falls into the "High Duty Core" category. Appendix F of Reference 3 suggests using best estimate values to calculate HDCI. When best estimate power, flow, and outlet temperatures are used in the HDCI calculation, the resulting HDCI for Watts Bar Unit 2, Cycle 1 is in the "Medium Duty Core" category.

<u>Requested Information</u>

2. If the WBNP core is considered high-duty, describe the measures to be taken to ensure that zinc addition does not increase the risk of CIPS and or clad corrosion, and that the overall risk of adverse fuel effects is minimized. Possible measures could include, but are not limited to:
   a. Implementation of a fuel surveillance program monitoring crud buildup and clad corrosion;
   b. Additional chemistry monitoring
   c. Application of operating experience with similar core designs.

Response: Westinghouse has substantial fuel operating experience with zinc addition in low, medium, and high duty cores. The industry experience with zinc addition in high duty cores has increased substantially since Reference 2 was issued. At Watts Bar, Unit 1 has already operated for two cycles with zinc injection using the same fuel type as will be used in Unit 2 and at a slightly higher power level.

As zinc addition has been applied to plants with boiling duty outside the prior fuel operating experience base for zinc addition, Westinghouse has undertaken fuel examinations including clad corrosion measurements. These measurements have shown no increase in cladding corrosion resulting from zinc addition and include high duty plants using zinc addition. The current operating experience and fuel surveillance experience bounds the duty expected for Watts Bar Unit 2. Therefore, no Watts Bar Unit 2 specific fuel surveillance is required to implement zinc addition in Watts Bar Unit 2 since the plant is bounded by other applicable fuel surveillance campaigns.

Westinghouse performs a fuel crud risk analysis for each operating plant cycle and includes the consideration of zinc addition in that risk analysis if zinc is used. Westinghouse also has zinc addition guidelines that are provided to each zinc addition PWR using Westinghouse fuel. These guidelines describe the chemistry monitoring requirements needed for zinc addition. In addition to monitoring zinc concentration in the coolant, corrosion product concentrations are also required to be measured. These measures have proven successful in avoiding any fuel performance issues associated with zinc addition in PWR cores using Westinghouse fuel.

5.3.1-1 Background

The regulatory acceptance criteria for a reactor vessel material surveillance program are the requirements of Section III of Appendix H of 10 CFR Part 50. Complying with the acceptance criteria satisfies the requirements of 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 32 regarding an appropriate material surveillance program for the reactor vessel. 10 CFR Part 50 Appendix H, paragraph III.B.3 requires:

“A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4. The proposed schedule must be approved prior to implementation.”

Additionally, Generic Letter 96-03, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” contains seven criteria that must be met for the relocation of the pressure-temperature (P-T) limits from the technical specifications to a pressure-temperature limits report (PTLR). One of these criteria is:

“The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.”

In Amendment 97, FSAR Section 5.4.3.6 was modified to state that the tentative schedule for removal of the capsules for post-irradiation testing is as shown in Table 4.0-1 of the Pressure and Temperature Limits Reports (PTLER). The actual proposed schedule was deleted from the FSAR. However, the PTLR (WCAP-17035-NP, Reference 4) does not contain this information.

Requested Information

1. Provide the proposed Table 4.0-1 which incorporates the schedule for withdrawal of the surveillance capsules in the PTLR.

Response: The following table will be added to the Unit 2 PTLR.

TABLE 4.0-1
Watts Bar Unit 2 Surveillance Capsule Removal Schedule[(a)]

Capsule	Orientation of Capsule	Lead Factor	Removal Time	Expected Capsule Fluence (n/cm^2, E > 1.0 MeV)
U	Dual 34°	5.13	1st Refuel Outage	0.50×10^{19}
W	Single 34°	5.18	6.1 EFPY	3.17×10^{19} [(b)]
X	Dual 34°	5.13	6.2 EFPY to 12.5 EFPY[(c)]	3.17×10^{19} to 6.34×10^{19} [(c)]
Z	Single 34°	5.18	Standby	-----
V	Dual 31.5°	4.40	Standby	-----
Y	Dual 31.5°	4.40	Standby	-----

Notes:

(a) This information is taken from the withdrawal schedule contained in WCAP-9455, Revision 3 (Ref. 3).

(b) Approximate Fluence at vessel inner wall at End-of-Life (32 EFPY).

(c) Capsule X should be withdrawn between 6.2 EFPY and 12.5 EFPY, which corresponds to a capsule fluence of not less than once (3.17×10^{19} n/cm^2 (E > 1.0 MeV)) or greater than twice (6.34×10^{19} n/cm^2 (E > 1.0 MeV)) the peak End-of-Life vessel fluence. This is consistent with the recommendations of ASTM E185-82.

The Unit 2 PTLR is included in the Unit 2 System Description for the Reactor Coolant System (WBN2-68-4001). This system description will be revised to reflect required revisions to the PTLR by September 17, 2010.

5.3.1-1 Background

The regulatory acceptance criteria for a reactor vessel material surveillance program are the requirements of Section III of Appendix H of 10 CFR Part 50. Complying with the acceptance criteria satisfies the requirements of 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 32 regarding an appropriate material surveillance program for the reactor vessel. 10 CFR Part 50 Appendix H, paragraph III.B.3 requires:

"A proposed withdrawal schedule must be submitted with a technical justification as specified in § 50.4. The proposed schedule must be approved prior to implementation."

Additionally, Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," contains seven criteria that must be met for the relocation of the pressure-temperature (P-T) limits from the technical specifications to a pressure-temperature limits report (PTLR). One of these criteria is:

"The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves."

In Amendment 97, FSAR Section 5.4.3.6 was modified to state that the tentative schedule for removal of the capsules for post-irradiation testing is as shown in Table 4.0-1 of the Pressure and Temperature Limits Reports (PTLER). The actual proposed schedule was deleted from the FSAR. However, the PTLR (WCAP-17035-NP, Reference 4) does not contain this information.

Requested Information

2. Include a description of the reactor vessel material surveillance program in the PTLR, including a discussion of how the specimen examinations shall be used to update the P-T curves.

Response: The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The results of these specimen examinations shall be used to update the P-T Curves.

The pressure vessel steel surveillance program is in compliance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the ASME Boiler and Pressure Vessel Code . The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

Westinghouse provided a proposed revision of the PTLR. Per this markup, the Reactor Vessel Material Surveillance Program has been updated as indicated above. The Unit 2 PTLR is included in the Unit 2 System Description for the Reactor Coolant System (WBN2-68-4001). This system description will be revised to reflect required revisions to the PTLR by September 17, 2010.

5.3.1-2 Background

10 CFR Part 50, Appendix H, Paragraph III.B.1 requires that the design of the surveillance program and the withdrawal schedule to meet the requirements of the edition of ASTM E 185 that is current on the issue date of the American Society of Mechanical Engineers (ASME) Code to which the reactor vessel was purchased, or later editions through ASTM E 185-1982.

FSAR Section 5.2.4 indicates that changes in fracture toughness of the core region forgings, weldments and associated heat affected zones (HAZ) due to radiation damage will be monitored by a surveillance program which is based on ASTM E 185-82, (Ref. 5) and 10 CFR Part 50, Appendix H. FSAR Section 5.2.4 further indicates that the surveillance program will be in compliance with these documents with the exception that all of the RV irradiation surveillance capsules will receive a neutron flux which is at least 4 times the maximum RV neutron flux. (i.e., the lead factor for all the capsules will be at least 4).

ASTM E 185-82 recommends that the surveillance capsule lead factors (the ratio of the instantaneous neutron flux density at the specimen location to the maximum calculated neutron flux density at the inside surface of the RV wall) be in the range of one to three.

More recent versions of ASTM E 185 acknowledge that it may not be possible to position capsules in low lead factor locations due to the design of the RV internals. ASTM E 185-02 recommends that plants with lead factors greater than five should provide a method of verifying the validity of the accelerated irradiation data. This verification may be accomplished by the inclusion of a reference material.

Requested Information

1. If any surveillance capsules will have a lead factor greater than five, describe how the validity of the accelerated irradiation data will be verified.

Response: Four of the six Unit 2 surveillance capsules have lead factors greater than five. The validity of the accelerated irradiation data will be assessed by comparing the Unit 2 surveillance data to the Regulatory Guide 1.99, Revision 2 (RG 1.99, Rev. 2) predictions. The validity of the data will also be assessed by comparing the Unit 2 surveillance data to results for similar forging and weld material to ascertain that the observed trends are consistent. The

RG 1.99, Rev. 2 predictions and the available data on similar materials from other plants represent data irradiated under a wide range of lead factors, so the Unit 2 surveillance data will be verified by trending it against the RG 1.99, Rev. 2 predictions and the data for similar surveillance materials.

Due to the lead factors being greater than 5, all capsules will be removed early in life. Unit 2 will be able to store capsules in the spent fuel pool for future reinsertion to assure that the vessel continues to be monitored throughout the licensed life (and potential license extensions).

Requested Information

2. For those surveillance capsules with lead factors greater than 3, justify that the surveillance specimens in the capsules will provide metallurgically meaningful data, in terms of the expected design life and/or licensed life of the RV, including possible license renewal terms, based on the fluences these capsules are projected to receive.

Response: Two of the six Unit 2 surveillance capsules have lead factors greater than three, but less than five. The validity of the accelerated irradiation data will be assessed by comparing the Unit 2 surveillance data to the Regulatory Guide 1.99, Revision 2 (RG 1.99, Rev. 2) predictions. The validity of the data will also be assessed by comparing the Unit 2 surveillance data to results for similar forging and weld material to ascertain that the observed trends are consistent. The RG 1.99, Rev. 2 predictions and the available data on similar materials from other plants represent data irradiated under a wide range of lead factors, so the Unit 2 surveillance data will be verified by trending it against the RG 1.99, Rev. 2 predictions and the data for similar surveillance materials.

Due to the all six of the Unit 2 lead factors being greater than 3, all capsules will be removed early in life. Unit 2 will be able to store capsules in the spent fuel pool for future reinsertion to assure that the vessel continues to be monitored throughout the licensed life (and potential license extensions).

5.3.1-3 Background

In Amendment 97, in FSAR Section 5.2.4.2, the description of the orientation of the Charpy V-Notch specimens used to determine the initial USE of the RV beltline region was changed from “transverse” to “tangential and axial.” Section 5.4.3.6 of the FSAR was also modified in Amendment 97 to change the description of the specimen orientation from “longitudinal and transverse” to “tangential and axial.”

ASTM E 185-82 Section 6.2 states:

“The tension and Charpy specimens from base metal shall be oriented so that the major axis of the specimen in parallel to the surface and normal to the principal rolling direction for plates, or normal to the major working direction for forgings as described in Section III of the ASME Code. The axis of the notch of the Charpy specimen for base metal and weld metal shall be oriented perpendicular to the surface of the material; for the HAZ specimens, the axis of the notch shall be as close to perpendicular to the surface as possible so long as the entire length of the notch is located within the HAZ.”

The requirements of ASTM E 185-82 with respect to specimen orientation are consistent with the ASME Code, Section III, NB-2322.2. The Nuclear Regulatory Commission’s (NRC) Branch Technical Position 5-3, “Fracture Toughness Requirements,” uses the term “weak direction”, defined as “transverse to the direction of maximum working.” NRC BTP 5-3 generally uses the terms “transverse” and “longitudinal” to describe the weak and strong specimen orientations.

Note (b) to Table B-1 in the PTLR (Ref. 4) indicates that the tangential direction is the strong direction and the axial direction is the weak direction. The FSAR changes appear to be for consistency with the PTLR.

Requested Information

For the WBNP-2 RV beltline materials, clarify the orientation of the Charpy specimens for the base metal in terms of the language used in ASTM E 185-82.

Response: Weak direction can be described as axial orientation for forgings (transverse for plates). The major axis of the specimen is parallel to the surface and normal to the major working direction. The axis of the notch is oriented perpendicular to the surface of the material.

Strong direction can be described as tangential orientation for forgings (longitudinal for plates). The major axis of the specimen is normal to the surface and parallel to the major working direction. The axis of the notch is oriented perpendicular to the surface of the material.

Data for the beltline initial upper shelf energy values in Table B-1 was only available for specimens tested in the strong direction (hence the 65% reduction per NUREG-0800, Rev. 1). However, the surveillance program contains Charpy specimens oriented both in the axial (weak) and tangential (strong) orientations so that both orientations may be tested in the future.

5.3.1-4 Background

Per 10 CFR 50.61, the pressurized thermal shock reference temperature (RT_{PTS}) for the RV beltline materials must be calculated for the end-of-license (EOL) fluence. 10 CFR 50.61 defines the *EOL fluence as* the best-estimate neutron fluence projected for a specific vessel beltline material at the clad-base-metal interface on the inside surface of the vessel at the location where the material receives the highest fluence on the expiration date of the operating license.

In order to comply with GDC 14 and 31 related to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized, the RV materials must meet 10 CFR Part 50 Appendix G, which requires that the Upper Shelf Energy (USE) remain above 50 ft-lbs through the expiration of the plant license unless it can be demonstrated that that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Regulatory Guide (RG) 1.99 provides the guidance on evaluating the drop in USE due to neutron irradiation. The projected fluence at EOL must be used to evaluate the EOL USE.

Appendixes B and C to the PTLR (Ref. 4) provide the projected USE and RT_{PTS} values based on EOL fluence values for 32 Effective Full Power Years (EFPY). However, many nuclear plants are now operating to capacity factors of 90% or more, which could make the projected EOL fluence values nonconservative if WBNP-2 achieves similar efficiency.

Requested Information

Given that nuclear power plants are now typically operating at capacity factors of 90% or greater, justify that a best estimate EOL neutron fluence based on 32 EFPY is appropriate and conservative, given that 32 EFPY is based on a capacity factor of 80% over a 40-year license. If a greater EFPY value should be postulated, provide updated Appendixes B and C to the PTLR which recalculate the USE and RT_{PTS} values for the WBNP-2 RV based on new EOL neutron fluence values.

Response: For reactors with actual operating experience, a plant-specific calculation is performed for fuel cycles that have been completed to provide a best-estimate fluence, and fluence projections for future operation are generated on an assumed mode of operation. For reactors with no actual operating experience, such as Watts Bar Unit 2, design basis fluence calculations are completed based on the assumption of operation with a conservative out/in (non-low-leakage) fuel loading pattern for the entire licensed lifetime of the reactor. This design basis vessel fluence at EOL, whether based on the assumed 80% capacity factor or a higher capacity factor, will be conservative compared with best-estimate vessel fluence analyses. Therefore, the design basis EOL fluence values are appropriately conservative and do not need to be adjusted to account for a higher capacity factor.

5.3.1-5 Background

10CFR 50 Appendix G, requires the values of RT_{NDT} and Charpy USE for the RV beltline materials, including welds, plates and forgings to account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part.

In order to determine the changes in reactor vessel fracture toughness from neutron irradiation, the RV wall fluence must be determined. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes acceptable approaches for fluence determinations.

FSAR Section 5.4.3.6.2 describes the methodology of the calculation of the fast neutron flux received by the surveillance samples. In FSAR Section 5.4.3.6.2.1, "Reference Forward Calculation, " in Amendment 97, the applicant included new material on the neutron transport calculation.

Requested Information

The first paragraph of FSAR Section 5.4.3.6.2.1 refers to legacy references for Unit 2 that predate the development of ENDF-B/VI, and hence are unacceptable for current calculations, according to RG 1.190 recommendations. The second paragraph, however, refers to forward transport calculations performed using more up-to-date methods for analysis of Capsule W and subsequent capsules. Please clarify the difference between these two paragraphs.

Response: The first paragraph contains a discussion of the previous methodology used in the calculation of the reference forward design basis analysis. This methodology used the BUGLE-93 crosssection library, which was based on ENDF-B/VI data.

It should be noted that the reference forward design basis analysis was updated using the BUGLE-96 cross-section library, which is also based on ENDF-B/VI data. The calculated design basis fluxes for these two cross-section libraries are virtually identical.

The second paragraph is a discussion of the updated Westinghouse fluence methodology which was used in the analysis of Unit 1 Capsule W. This is the current Westinghouse methodology which will be used to calculate updated fluence values when surveillance capsules are pulled and analyzed in the future.

5.3.1-6 Background

With respect to special processes used to fabricate the RV (e.g. welding) SRP Section 5.3.1 essentially states that the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for fabrication of components, when the appropriate code symbols are affixed and appropriate certifications made by the manufacturer or installer.

With respect to special methods for nondestructive examination, SRP Section 5.3.1 further states that the requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the ASME Code, Section III, for fabrication nondestructive testing, and that the acceptance criteria for examination of the RV and its appurtenances by nondestructive examination are those specified in ASME Code Section III, NB-5000.

In FSAR Section 5.4.4.2, "Penetrant Examinations," in Amendment 97 the applicant changed the description of the liquid penetrant examinations of the core support block attachment welds. The description previously stated the core support block attachment welds were inspected by dye penetrant after first layer of weld metal and after each 2 inches of weld metal. The description now states the core support block attachment welds were inspected by dye penetrant after each ½ inch of weld metal.

The code of record for the WBNP-2 RV is the ASME Code, 1971 edition through 1973 addenda. ASME Code, Section III, Sub article NB-4433 of the code of record requires that structural attachment welds be full penetration welds except for temporary attachments and minor supports. ASME Section III, Subarticle NB-5260 states that structural attachment welds to pressure-retaining material shall be examined by either the magnetic particle or liquid penetrant method. The ASME Code does not provide any requirements for the increment of liquid penetrant examination, if performed, for structural attachment welds. However, inspecting the core support block attachment welds after each ½ inch of weld is consistent with the requirements of ASME Code, Section III, NB-5245 for fillet welded and partial penetration welded joints. ASME Code, Section III, NB-5245 requires such welds to be examined progressively using either the magnetic particle or liquid penetrant methods, with the increments of examination being the lesser of one half of the maximum welded joint dimension measured parallel to the center line of the connection or ½ in. (13 mm).

Provided that the core support block attachment welds are full penetration welds, all ASME code requirements are met and the specified nondestructive examination requirements do not conflict with the ASME code requirements.

<u>Requested Information</u>

1. Are the core support block attachment welds full penetration welds?

Response: The core support block welds are full penetration welds, but they are not welded through the full section of the blocks. There is a slot between the block and the vessel shell running the full width of the each block.

2. Provide the basis for the nondestructive examination requirements for the core support block attachment welds.

Response: The progressive liquid penetrant is performed as an alternative to ultrasonic examination (UT) because the configuration of the core support block attachment welds is not suitable for performing meaningful UT. In that respect, the situation is similar to a partial penetration weld, and progressive liquid penetrant examination is considered the acceptable alternative examination.

NOTE: One comment regarding Paragraph 4 of the RAI states that the WBNP-RV Code of Record is the ASME Code 1971 Edition through 1973 Addenda. This is incorrect; per the Reactor Vessel Design Specification, the Code of Record is 1971 Edition through Winter 1971 Addenda.

5.3.2-1 <u>Background</u>

10 CFR Part 50 Appendix G requires that pressure-temperature (P-T) limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code. The applicant provided proposed P-T limits in Reference 4. In the process of performing confirmatory calculations to check the WBNP-2 P-T curves, for the 100^{o}F/hour heatup, the allowable pressure calculated by the staff at lower temperatures is significantly less than the temperature calculated by the applicant using the methodology of Reference 6. The staff used the methodology of the ASME Code, Section XI, Appendix G to calculate the allowable pressures. The staff also observed that the thermal stress intensity factors K_{It} calculated by the applicant were generally significantly less than those calculated using the equations of the ASME Code, Section XI, Appendix G. The staff also observed that the metal temperatures given in the PTLR are higher than the corresponding metal temperatures calculated using ASME Code Section XI Appendix G Figure G-2214.2, up to a coolant temperature of 150^{o}F, above which the ASME metal temperatures are higher. The combination of the higher K_{It} and lower metal temperatures (resulting in a lower K_{Ic}) results in a lower allowable pressure being calculated using the ASME Code Appendix G methods, although the ASME values converge with the applicant's values as temperatures increase, and are within 1% of the applicant's values at 170^{o}F.

The staff used the simple equation from ASME Code Section XI, Appendix G, paragraph G-2214.3 to calculate the maximum K_{It} as a function of heatup rate:

$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$

Where:

HU is the heatup rate in ^{o}F/hr

t = wall thickness in inches

The staff used ASME Code Section XI, Appendix G, Figure G-2214-2 to determine the temperature difference from the coolant at a specific wall depth.

The PTLR provides the above ASME Code Section XI, Appendix G equation for K_{1t} from paragraph G-2214.3 as equation (6).

ASME Code Section XI Appendix G, paragraph G-2214.3 also provides the following alternative equation for the thermal stress intensity of an outside surface defect during heatup (reproduced as Equation (8) in the PTLR:

$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3)^* \sqrt{\pi a}$

The coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using:

$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3$

PTLR Section 3.2 notes that equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. Section 3.2 of the PTLR further states that the P-T curve methodology is the same as that described in Section 2.6 of Reference 6 (equations 2.6.2-4 and 2.6.3-1).

However, Equation 2.6.2-4 (for the steady state analysis) and 2.6.3-1 (for the finite heatup and cooldown rate analyses) of Reference 6 provide a different method of calculating K_{It} than described by the above as equations, as described by the following equations:

$K_{1p} = 1.1M_K\sigma_p \sqrt{\pi a/Q}$ (2.6.2-4)

$Q = \Phi^2 - 0.212(\sigma_p/\sigma_y)$

$K_{1t} = [\sigma_m 1.1M_k + \sigma_b M_b]\sqrt{\pi a/Q}$ (2.6.3-1)

$Q = \Phi^2 - 0.212(\sigma_m + \sigma_b/\sigma_y)$

Where:

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_k = correction factor for membrane stress

M_b = correction factor for bending stress, as a function of relative flaw depth (a/t)

Q = flaw shape factor modified for plastic zone size,

Φ = is the elliptical integral of the 2nd kind (Φ = 1.11376 for the fixed aspect ratio of 3 of the code reference flaw),

0.212 = plastic zone size correction factor,

σ_p = pressure stress,

σ_y = yield stress,

1.1 = correction factor for surface breaking flaws,

a = crack depth of ¼ t, and

K_{Ip} = pressure stress intensity factor.

The staff notes that when the applicant's K_{It} and metal temperature values were input to the ASME Code Section XI, Appendix G equations for allowable pressure, the values obtained are identical to the applicant's allowable pressures. However, the staff requires additional information on how the applicant's K_{It} and metal temperatures were determined.

Requested Information

In order to complete our review of the P-T limits, the staff requests the following information:

1. Clarify which set of equations was used to determine the K_{It} values used as input to the P-T curve calculation.

Response: The set of equations used to determine the KIt values are defined in Appendix G to Section XI of the ASME Code, paragraph G-2214.3, part (b).

For an inside surface defect during cooldown:

$$K_{It} = (1.00359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)^* \sqrt{\pi a}$$

For an outside surface defect during heatup:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3)^* \sqrt{\pi a}$$

The coefficients C0, C1, C2 and C3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3$$

Where x is a variable that represents the radial distance (in.) from the appropriate (i.e., inside or outside) surface and a is the maximum crack depth (in.).

These equations are incorporated in the OPERLIM computer code.

Requested Information

2. If equation (8) of the PTLR was used, elaborate on how the constants C_0, C_1, C_2 and C_3 were determined, and how the thermal stress distribution σ(x) was determined.

Response: Equation (8) of WCAP-17035-NP was used. The temperature distribution through the wall was calculated as a function of time and position for both heatup and cooldown. The one-dimensional transient heat conduction equation was used to determine the through-wall temperature distribution (see Section 2.6.1 of WCAP-14040-A, Revision 4) as a function of time during the heatup or cooldown.

Using the through-wall temperature distribution, the thermal stress distribution was then determined using the equations contained in Section 2.6.1 of WCAP-14040-A, Revision 4 for computing thermal stresses. Then, a polynomial fitting method was used to determine the values for C_0, C_1, C_2, and C_3, as defined in Equation (8) of WCAP-17035-NP.

The methods described above are incorporated into the OPERLIM computer code.

Requested Information

3. If Reference 2 equations 2.6.2-4 and 2.6.3-1 were used, describe how the membrane (σ_m) and bending (σ_b) stresses were determined. Specifically, how was the initial stress profile determined prior to the linearization procedure? Also, what value was used for the yield stress?

Response: Equations 2.6.2-4 and 2.6.3-1 were used in the calculations. The methods of Appendix A to Section XI of the ASME Code were used in determination of membrane and bending stresses. The initial stress profile was determined using the equation given by Timoshenko (Reference 14 of WCAP-14040-A, Revision 4), which is Equation 2.6.1-4 in WCAP-14040-A, Revision 4. The value used for yield stress is a constant 50 ksi.

These methods are incorporated into the OPERLIM computer code.

Requested Information

4. With respect to the determination of the crack tip metal temperatures:

   a. Describe the boundary conditions that were assumed, particularly at the vessel outer diameter, and,

   b. Describe the methodology for calculating the temperatures.

Response: The vessel inner surface is assumed to have a very high convection coefficient (7000 BTU / (hr * ft2 * °F). The very high convection coefficient essentially allows unimpeded transfer of heat from the coolant to the inside surface of the vessel. The outside surface is assumed to be adiabatic. Therefore, it is perfectly insulated so that heat does not transfer from the outside surface to the air.

The temperatures are calculated using the one-dimensional transient heat conduction equation that is contained in Section 2.6.1 of WCAP-14040-A, Revision 4. A through-wall temperature distribution was calculated for each time step during each cooldown or heatup ramp of interest.

These methods and convection coefficients are incorporated into the OPERLIM computer code.

5.3.2-2 <u>Background</u>

In order to comply with GDC 15 as it relates to the reactor coolant system (RCS) being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operations, including anticipated operational occurrences, and 10 CFR Part 50 Appendix G with respect to fracture toughness requirements for the RV, SRP Section 5.2.2 provides guidance for the design of the low-temperature overpressure protection (LTOP) system or the cold overpressure mitigation system (COMS). The LTOP system or COMS should be in accordance with the requirements of NRC Branch Technical Position (BTP) 5-2, “Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures,” and that the LTOP system or COMS should be operable during startup and shutdown conditions below the enable temperature defined in paragraph II.2 of BTP 5-2.

Technical Specification B 3.4.12, “Cold Overpressure Mitigation System (COMS),” refers to the COMS arming temperature specified in the PTLR. However, the COMS arming temperature is not specified in the PTLR. Technical Specification B 3.4.12 also states that “The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs).” However, the PORV setpoints are not provided in the PTLR.

<u>Requested Information</u>

1. Provide the COMS arming temperature for WBNP-2. The PTLR must be revised to incorporate this information.

   Response: COMS is armed when any RCS cold leg temperature is ≤ 225°F. The Unit 2 PTLR is included in the System Description for the Reactor Coolant System (WBN2-68-4001) which will be revised to incorporate this information by September 17, 2010.

Requested Information

2. Provide the PORV setpoints for WBNP-2. The PTLR must be revised to incorporate this information.

Response: The PORV for the A train Pressurizer is 340A, and the PORV for the B train Pressurizer is 334B.

The PORV setpoints for Unit 2 are given in the following table:

Watts Bar Unit 2 PORV Setpoints vs. Temperature

Temperature (°F)	PCV 334 Setpoint (psig)	PCV 340A Setpoint (psig)
70	455	425
100	455	425
125	455	425
145	510	460
150	510	460
175	510	460
200	775	720
225	775	720
350	2335	2335

The Unit 2 PTLR is included in the System Description for the Reactor Coolant System (WBN2-68-4001) which will be revised to incorporate this information by September 17, 2010.

6.1.1-1 Background

In order to comply with GDC 41 as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained, SRP Section 6.1.1 recommends that hydrogen generation resulting from the corrosion of metals by containment sprays during a design-basis accident should be controlled as described in RG 1.7, “Control of Combustible Gas Concentrations in Containment,” Regulatory Position C.6 (note, the SRP recommendation was based on Revision 2 to RG 1.7, Regulatory Position C.6 is now Regulatory Position C.4 in Revision 3 to RG 1.7). RG 1.7, Regulatory Position C.4 states:

Materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

10 CFR 50.44 requires that all water-cooled reactor construction permits or operating licenses under Part 50 issued after October 16, 2003 comply with the following:

All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

FSAR Section 6.2.5 states that the combustible gas control system of the containment air return system, the hydrogen analyzer system (HAS) and the hydrogen mitigation system (HMS) conform to 10 CFR 50.44 requirements. FSAR Section 6.2.5.1 further states that:

In an accident more severe than the design-basis loss-of-coolant accident (LOCA), combustible gas is predominantly generated within containment as a result of the following:

(1) Fuel clad-coolant reaction between the fuel cladding and the reactor coolant.

(2) Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel.

It appears that corrosion of materials has been removed from the design bases of the HMS in FSAR Section 6.2.5.

Additionally, FSAR Section 6.2.1.3.3 includes the following as an input assumption for the containment pressure analysis:

Hydrogen gas was added to the containment in the amount of 25,230.2 Standard Cubic Feet (SCF) over 24 hours. Sources accounted for were radiolysis in the core and sump post-LOCA, corrosion of plant materials (aluminum, zinc, and painted surfaces found in containment), reaction of 1% of the Zirconium fuel rod cladding in the core, and hydrogen gas assumed to be dissolved in the reactor coolant system water. (This bounds tritium producing core designs.)

If the potential for generation of hydrogen gas due to corrosion of reactive metals is insignificant compared to the generation of hydrogen due to the fuel clad-coolant reaction and the molten core-concrete interaction in a severe core melt sequence with a failed RV, it may be unnecessary to limit or quantify reactive metals in order to comply with 10 CFR 50.44.

However, SRP Section 6.1.1 still recommends that hydrogen generation due to corrosion of reactive metals be addressed. The information in FSAR Section 6.2.1.3.3 also implies a design basis assumption on the amount of aluminum, zinc and coatings containing reactive metals.

Requested Information

1. Was the potential for generation of hydrogen due to corrosion of reactive metals such as zinc or aluminum considered in the design of the combustible gas control system, or other analyzes such as containment pressure? If so, describe how this contribution was evaluated.

2. If the contribution of hydrogen from reactive metals was not evaluated, justify why this contribution was not evaluated.

3. If the contribution of hydrogen from reactive metals was evaluated, discuss the measures taken to ensure that the use of materials that could yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions is limited as much as practicable, and is maintained within design basis limits.

References

1. Overview Report on Zinc Addition in Pressurized Water Reactors-2004, 1009568 Final Report, December 2004, Electric Power Research Institute

2. Pressurized Water Reactor Primary Water Zinc Application Guidelines 1013420 Final Report, December 2006, Electric Power Research Institute

3. PWR Axial Offset Anomaly (AOA) Guidelines, Revision ,1008102, Final Report, June 2004, Electric Power Research Institute

4. WCAP-17035-NP, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," Revision 2, December 2009 (ADAMS Accession No. ML100550651)

5. ASTM E-185-82, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"

6. WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 4, May 2004 (ADAMS Accession No. ML0501202094)

Response to 1: Yes. Reactive metals were considered in the concentration of hydrogen generated during post-LOCA conditions calculations. It should be noted that Unit 2 is being licensed consistent with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment," Revision 3. As such, unlike Unit 1, Unit 2 will not have Hydrogen Recombiners (there will be Hydrogen Igniters). Reactive metals were also appropriately considered in the Westinghouse containment analyses. The quantity of reactive metals considered was conservatively assumed to be approximately 130% of the Unit 1 baseline inventory.

Response to 2: As discussed in the response to question 1, the contribution of reactive metals was considered.

Response to 3: Watts Bar procedures require evaluation and accounting of reactive metals (aluminum and zinc) in containment to minimize the production of post accident hydrogen. These procedures require that, “Materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practical.”

As indicated in Response 1, Unit 2 calculations are based on approximately 130% of the Unit 1 baseline inventory. Near the end of the Unit 2 construction completion project, a Unit 2 baseline inventory will be established using techniques developed for Unit 1. This will include a combination of containment walkdowns, and design basis document and work package reviews. Once the baseline inventory is established, it will be compared to Unit 2 calculations to assure that the assumptions used are conservative. After the baseline inventories are established, future additions and removals will be controlled by station procedures.

Preliminary RAIs for FSAR 6.2.4 and 6.2.6 (taken from e-mail from NRC dated 05/10/2010)

FSAR Sections 6.2.4 and 6.2.6

1. Figure 6.2.4-13, “Type XV, Personnel Access”, appears to show the interior door within the shield building wall with the airlock doors opening outward, away from primary containment. FSAR Section 6.2.4.2.3, “Penetration Design”, states that “A special hold-down device is provided to secure the inner door in a sealed position during leak rate testing of the space between the doors.” This would suggest that the doors in each primary containment airlock open inward. Clarify the orientation of the personnel access airlock doors

Response: Both personnel access airlock doors (interior and exterior) open inward towards containment (refer to Section A-A of Unit 2 FSAR Figure 6.2.4-13).

2. FSAR Section 6.2.4.2.3, “Penetration Design”, states that “The ice blowing line penetration has a blind flange with double O-rings installed on the outside of the containment as shown in Figure 6.2.4-16. Sealing between the outside and the annulus penetration through the shield is provided by a blind flange fitted with a gasket installed on the inside and outside of the Shield Building penetration. FSAR Section 6.2.4.3.1, “Possible Leakage Paths”, includes in the Type B leakage paths from containment to the annulus the ice blowing line O-ring and blind flange through line leak and refers to Figures 6.2.4-16 and 6.2.4-23. Figure 6.2.4-16, “Type XVIII, Ice Blowing Line”, shows a line having a flange with single O-ring gasket inside the containment building as well as in the annulus between the containment building and shield building. Figure 6.2.4-23, “Ice Blowing and Negative Return Lines – Blind Flange Details”, shows a single flange with double square cross section gaskets located in the annulus. Clarify the configuration of the ice blowing and negative return penetration barriers.

Response: Amendment 98 to the Unit 2 FSAR corrected Figure 6.2.4-16 to show that the ice blowing line penetration (X-79A) has a blind flange with double O-rings installed on the outside of containment. Figure 6.2.4-23 which provides the details for the ice blowing and negative return penetration barriers (X-79A and X-79B) correctly shows double O-rings that fit into a square groove in the flange. This figure also shows that these O-ring gaskets are compressed by the blind flange when it is installed, thereby providing a double barrier against containment leakage.

3. Table 6.2.4-1, "Watts Bar Nuclear Plant Containment Penetrations and Barriers", has several inconsistencies for which the staff needs clarification, including:

a) Table 6.2.4-1 contains numerous abbreviated notations for information as well as a notes column but does not appear to have a legend or notes table to support understanding of the data presented.

Response: Amendment 99 to the Unit 2 FSAR reattached the four pages of abbreviations and notes to the end of Unit 2 FSAR Table 6.2.4-1 as sheets 65 through 69 of 69.

b) Entry for penetration X-6 lists the valve numbers as 50 and 51 while the details sketch shows valves 51 and 58.

Response: Amendment 98 to the Unit 2 FSAR corrected the sketch for containment penetration X-6 of Table 6.2.4-1 to show the valve numbers as 50 and 51.

c) Entry for penetrations X-23, X-28, X-85A, X-86A, X-86B, X-86C, X-92C, X-105, and X-106 show them to be spare but also shows both App J Type C as well as Type A tests being applicable. Table 6.2.6-2, "Containment Isolation Valves Subjected to Type C Testing", lists isolation valves for penetrations X-23, X-28, X-85A, X-86A, X-86B, X-86C, X-92A, X-92B, X-92C, X-105, and X-106.

Response: Amendment 99 to the Unit 2 FSAR corrected Table 6.2.4-1 to remove the Appendix J Type C test identified for spare containment penetrations.

Amendment 98 to the Unit 2 FSAR corrected Table 6.2.6-2 to remove all valves listed for spare penetrations X-105 and X-106 and the penetrations.

Amendment 100 to the Unit 2 FSAR will revise Table 6.2.6-2 to remove spare penetrations X-23, X-28, X-85A, X-86A, X-86B, X-86C, X-92A, X-92B and X-92C and associated valves.

d) Entry for penetration X-25C shows both test connection valves (in series) 42B/1 and 42B/2 as being closed for ILRT testing but only 42B/1 as being tested by the ILRT.

Response: As noted in the response to question 3.e below, these test connection valves should not have been listed in Unit 2 FSAR Table 6.2.4-1 and are being removed from the table via Amendment 100.

Note: Containment penetration X-25C has been spared and the instrument line for PdT 30-42 now enters containment via penetration X-60B.

e) Entries for instrumentation penetrations (X-25C, X-26C, X-85C, X-86D, X-97) list the test connection valves while the entries for other penetrations do not specifically list valve data for test connection valves.

Response: The subject test connection valves are all less than 1-inch nominal diameter, administratively locked closed whenever containment integrity is required and form a double barrier in the associated test connection line. Section 3.3.1 of ANSI/ANS-56.8-2002 indicates valves that meet these criteria do not require Appendix J, Type B or C testing. These test connection valves, when installed in the mid 1990s, were added to the entries for the associated containment instrumentation penetrations on FSAR Table 6.2.4-1 via Amendment 66. However, as test connection valves that are not subject to Appendix J, Type B or C testing, they should not be listed in FSAR Table 6.2.4-1.

Amendment 100 to the Unit 2 FSAR will revise Table 6.2.4-1 to remove the listing of these test connection valves from containment instrumentation penetrations X-26C, X-57B, X-60B, X-97, X-98 and X-102. This will ensure the entries for these penetrations are consistent with the entries for other penetrations that do not specifically list valve data for test connection valves.

Additionally, since these test connection valves are not subject to Type B or C testing, Amendment 100 to the Unit 2 FSAR will revise Table 6.2.6-3 to remove the test connection valves for containment instrumentation penetrations X-26C, X-57B, X-60B, X-97, X-98 and X-102.

Note: Amendment 98 to Unit 2 FSAR Table 6.2.4-1 made the following changes in penetrations for the indicated instrument PdTs:

PdT 30-42 was moved from penetration X-25C to penetration X-60B.

PdTs 30-44 and 30-311 were moved from penetration X-85C to penetration X-57B.

PdT 30-45 was moved from penetration X-86D to X-102.

f) Entries for fire protection system penetrations X-31 and X-78 show a process fluid code of “A” and normal position codes of “O” for the inboard and outboard isolation valves.

Response: The reactor building, including the annulus, is provided with dry standpipe systems. Thus, process fluid code of “A” (air) is correct for fire protection line penetrations X-31 and X-78.

Amendment 100 to the Unit 2 FSAR will revise Table 6.2.4-1 to show the normal and shutdown positions of check valves 26-1296 and 26-1260 as closed for penetrations X-31 and X-78, respectively.

g) Entry for penetration X-107 shows no applicable App J test for relief valve 74-505 when the other inboard boundary valves are shown as tested by App J Type A test.

Response: The RHR system is in service during the ILRT and therefore, the valves associated with penetration X-107 are not Type A tested.

Amendment 100 to the Unit 2 FSAR will revise FSAR Table 6.2.4-1 to:

- show valve 74-2 is open during the ILRT; and
- remove the Type A test identified for valves 74-2, 74-8 and 63-185.

h) Entry for penetration X-118 indicates there are two blind flanges, one in the containment building and one in the shield building, although the detail sketch only shows the flange in the shield building. The detail listing also does not show a normal, shutdown, post-accident, or ILRT position for the flange in the shield building.

Response: Amendment 100 to the Unit 2 FSAR will revise Table 6.2.4-1 for penetration X-118 to remove the blind flange inside containment from the table since the flange is not a Code item, and thus cannot be credited for containment boundary. The revision will show that the flange inside the shield building is closed under normal conditions, can be open for shutdown conditions, is closed under post-accident conditions and can be open for ILRT. The blind flange installed on penetration X-118 inside the shield building is designed with double O-ring gaskets and a leak rate test connector. These O-ring gaskets are compressed by the blind flange when it is installed thereby, providing a double barrier against containment leakage.

i) Entry for penetration X-54 lists a blank flange in both the containment building and in the shield building but the detail sketch shows only the flange in the shield building and Figure 6.2.4-15, “Type XVII, Incore Instrumentation Thimble Assembly Renewal Line”, shows only one flange in the shield building. The valve (barrier) data listing for one flange indicates that it is closed post-accident but that the App J ILRT position is “O”.

Response: Unit 2 FSAR Table 6.2.4-1 incorrectly lists two blind flanges for penetration X-54, one in the shield building and one in the auxiliary building. Note 16 to Table 6.2.4-1 incorrectly indicates two blind flanges are provided in the annulus for penetration X-54, one as a primary containment isolation barrier and the other as a secondary containment isolation barrier.

Amendment 100 to the Unit 2 FSAR will revise Table 6.2.4-1 to remove the blind flange inside the auxiliary building from the listing for containment penetration X-54. Note 16 for this penetration will be revised also.

The blind flange installed on penetration X-54 inside the shield building is designed with double O-ring gaskets and a leak rate test connector.

These O-ring gaskets are compressed by the blind flange when it is installed, thereby, providing a double barrier against containment leakage. The blind flange in the shield building on containment penetration X-54 is closed post-accident. During the ILRT, the blind flange is opened, and a pressurization assembly is attached to the penetration to support the pressurization of the containment for the ILRT.

j) Entry for penetration X-39A shows isolation valves 63-64 and 63-868 while Table 6.2.6-2 shows isolation valves 63-64 and 77-868.

Response: Amendment 100 to the Unit 2 FSAR will revise Table 6.2.6-2 to identify valve 77-868 as 63-868.

k) Entry for penetration X-39B shows isolation valves 68-305 and 68-849 while Table 6.2.6-2 shows isolation valves 68-305 and 77-849.

Response: Amendment 100 to the Unit 2 FSAR will revise Table 6.2.6-2 to identify valve 77-849 as 68-849.

l) Entry for penetration X-41 details sketch shows relief valve 1-77-2875.

Response: Amendment 100 to the Unit 2 FSAR will revise Table 6.2.4-1 to identify the valve shown in the sketch for penetration X-41 as 77-2875.

m) Entry for penetration X-47A shows isolation valves 61-191, 61-192, and 61-533 while Table 6.2.6-2 lists valves 61-191, 61-192, and 61-788.

Response: Amendment 100 to the Unit 2 FSAR will revise Table 6.2.6-2 to replace valve 61-788 with valve 61-533.

n) Entry for penetration X-47B shows isolation valves 61-193, 61-194, and 61-680 while Table 6.2.6-2 lists valves 61-193, 61-194, and 61-935.

Response: Amendment 100 to the Unit 2 FSAR will revise Table 6.2.6-2 to replace valve 61-935 with valve 61-680.

o) Entry for penetration X-76 shows isolation valves 33-713 and 33-714 while Table 6.2.6-2 lists isolation valves 33-732 and 33-733

Response: Unit 2 FSAR Amendment 98 corrected Table 6.2.4-1 such that both the valve sketch and the valve data identify the valves as 33-732 and 33-733.

p) Entry for penetration X-97 shows no isolation valves and only test connection valves 30-133B/1 and 30-133B/2 while Table 6.2.6-2 lists isolation valves 30-134 and 30-135

Response: Table 6.2.4-1 is correct in that there are no containment isolation valves for penetration X-97. Amendment 98 to the Unit 2 FSAR removed penetration X-97 from Table 6.2.6-2 (Containment Isolation Valves Subjected to Type C Testing).

q) Entry for penetration 26C lists in-line valves 30-43A and 30-310A along with test connection valves 30-43C1, 30-43C2, 30-310C1, and 30-310C2 while Table 6.2.6, "Valves Exempted From Type C Leak Testing", lists only the test connection valves. Table 6.2.4-1 also shows valves 30-43A and 30-310A being open post-accident and for ILRT but also being tested by the ILRT

Response: Valves 30-43A and 30-310A are normally open manual isolation valves that are not required to close post-accident to provide containment isolation. As such, these valves should not have been listed in Unit 2 FSAR Table 6.2.4-1 for containment penetration X-26C. These valves are also not subject to any Appendix J Type B or C testing, and therefore, should not be included in Unit 2 FSAR Table 6.2.6-3.

Amendment 100 to the Unit 2 FSAR will revise Tables 6.2.4-1 and 6.2.6-3 to remove the test connection valves and valve numbering from the sketches for penetration 26C."

4. Section 6.2.4.1, "Design Bases", item (5) states that "Relief valves may be used as isolation valves, provided the relief valve setpoint is greater than 1.5 times the containment design internal pressure." The information in Table 6.2.4-1 shows all relief valves used as containment isolation valves, other than those on the four main steam lines, to be discharging to the primary containment, whether or not they are installed inside or outside of containment. Confirm if this is correct.

Response: Unit 2 FSAR Table 6.2.4-1 identifies five relief valves as containment isolation valves: 2-RV-62-662 (X-15), 2-RV-63-28 (X-30), 2-RV-70-703 (X-35), 2-RV-77-2874 (X-41) and 2-RV-74-505 (X-107).

The requested information for each of these valves is as follows:

1. 2-RV-62-662 (X-15): configuration control drawings (CCDs) 2-47W809-1 and 2-47813-1 show 2-RV-62-662 discharging inside containment to the pressurizer relief tank.

2. 2-RV-63-28 (X-30): EDCR 53580 installs 2-RV-63-028. Drawing Revision Authorization, DRA 53580-001 is the markup for CCD 2-47W811-1. On this drawing, the valve is located outside of containment. The flow diagram continues on CCD 2-47W813-1 which shows the relief valve discharge entering containment through X-24; it then goes to the pressurizer relief tank. The relief valve discharge line for thermal relief valve 2-RV-63-28 (X-30) is routed to a 4-inch relief valve discharge header located outside containment. The relief valve

discharge header is routed to the PRT and enters containment through penetration X-24. The containment isolation provisions for penetration X-24 consist of a closed system outside containment and check valve 68-559 inside containment.

3. 2-RV-70-703 (X-35): EDCR installs 2-RV-70-703. As shown on CCD 2-27W859-3, 2-RV-70-703 is located inside containment and discharges inside containment to the waste disposal system.

4. 2-RV-77-2874 (X-41): EDCR 53948 installs 2-RV-77-2874. DRA 53948-005 is the markup for CCD 2-47W851-1. On this drawing, the valve is located inside containment and discharges to the equipment drain sump.

5. 2-RV-74-505 (X-107): As shown on CCD 2-47W810-1, 2-RV-74-505 is located inside containment and discharges inside containment to the pressurizer relief tank.

5. Section 6.2.6.2, “Containment Penetration Leakage Rate Test”, description of exemptions (I)(2), (I)(3), and (I)(5) state, as does Note 3 to Table 6.2.6-3, that water testing of water sealed valves is “as specified in 10 CFR 50, Appendix J.” This section also indicates that Appendix J Option B is to be implemented as specified in the plant Technical Specifications. Appendix J Option B does not specifically describe water testing as does Option A. Appendix J Option B requires the plant technical specifications will include, by general reference, the regulatory guide or other implementation document used to develop a performance-based leakage-testing program. Clarify the source of the requirement regarding water testing of valves.

Response: The references to water testing in I(2), I(3) and Note 3 pertain to piping integrity and Section XI of the ASME Boiler and Pressure Vessel Code. System pressure tests are described in Article IWA-5000 of the code.

Exclusion from Type C testing is based on the provisions of a seal pressure greater than 1.1 P_a and a 30-day seal water inventory. This is as described in 10 CFR 50, Appendix J. Type C testing is performed with air or nitrogen.

6. Section 6.2.6.2, “Containment Penetration Leakage Rate Test”, item (1), “Method 1, Pressure Decay” indicates that either air or nitrogen can be used as the test medium and that the leakage rate be calculated using the specified formula. The formula does not appear to provide a conversion for the results when using nitrogen. Clarify if test results are converted to equivalent air leakage when using nitrogen. This section also uses the abbreviation “P_{ac}” while “P_a” is used elsewhere. “P_a” is defined in Appendix J while “P_{ac}” is not. Describe how both abbreviations are being used.

Response: **Nitrogen Conversion**

The Pressure Decay Method does not use partial vapor pressures nor does it use a conversion for pure nitrogen. It uses the Ideal Gas Law found in ANSI/ANS-56.8-1994:

$$L_i = \frac{TV}{t}\left[\frac{P_1}{T_1} - \frac{P_2}{T_2}\right]\frac{T_{stp}}{P_{stp}}$$

Where:

L_i = Local leak rate, cfm

TV = Test volume, ft^3

P_1 = Initial pressure, psia

P_2 = Final pressure, psia

T_1 = Initial temperature, °R

T_2 = Final temperature, °R

T_{stp} = Standard atmospheric temperature, °R

P_{stp} = Standard atmospheric pressure, psia

t = Test duration, min.

For Type B and C tests, the test volume is initially pressurized with air or nitrogen above the design accident pressure of 15 psig. The final test pressure also remains above the design accident pressure. Pressures and temperatures are recorded at the beginning and at the end of each test. This is a 'state point to state point' test that does not need to be adjusted for partial vapor pressures.

Watts Bar Unit 2 Technical Specifications Bases 3.6.1 (Containment) references Regulatory Guide 1.163 for Appendix J testing. The Regulatory Position in that document is that NEI 94-01 provides methods acceptable to the NRC staff for complying with the provisions of Option B in 10 CFR 50 Appendix J. NEI 94-01, in turn, references ANSI/ANS-56.8-1994 where this Pressure Decay Method is described (Section 6.4).

P_{ac} vs P_a

The abbreviation P_a is found in 10 CFR 50 Appendix J Option B and is defined as the calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the Technical Specifications. The abbreviation P_{ac} is found in ANSI/ANS-56.8-1994 and is defined as the calculated peak containment internal pressure related to the DBA. In later revisions of 56.8, P_{ac} was changed to P_a. Also, as seen in the NRC Safety Evaluation Report to revision 2-A of NEI 94-01, P_{ac} was changed to P_a. P_{ac} was a carry-over from the 1994 revision of 56.8. Amendment 100 to the Unit 2 FSAR will change P_{ac} to P_a.

7. Section 6.2.6.2, “Containment Penetration Leakage Rate Test”, item (1), “Method 3, “Waterflow”, has been deleted but the paragraph preceding the Method 1 description still refers to water as being a pressurizing medium. Clarify whether or not water could be used as the pressurizing medium.

Response: Watts Bar 2 does not perform Type B or C testing using water as the medium. These tests are performed using the pneumatic fluids nitrogen or air. Water seals are provided on some of the penetration lines and a seal water inventory leakage rate test may be performed in lieu of a Type C air leakage rate test. Amendment 100 to the Unit 2 FSAR will remove water from the paragraph that discusses it being a pressurizing medium.

8. Table 6.2.6-2, “Containment Isolation Valves Subjected to Type C Testing”, has several inconsistencies for which the staff needs clarification, including:

a) Penetration X-52 entry lists valves 1-70-100 and 1-70-790.

Response: Amendment 100 to the Unit 2 FSAR will remove the Unit 1 identifiers from isolation valves “1-70-100” and “1-70-790” in Table 6.2.6-2. These valve numbers were verified with CCD 2-47W859-3.

b) Penetration X-56A entry lists valves 1-67-113 and 1-67-1054D.

Response: Amendment 100 to the Unit 2 FSAR will remove the Unit 1 identifiers from isolation valves “1-67-113” and “1-67-1054D” in Table 6.2.6-2. These valve numbers were verified with CCD 1-47W859-3 and EDCR 52796.

c) Penetration X-65 entry lists valves 31-309, 31-308, and 31-3407 while Table 6.2.4-1 details sketch shows valves 31-309, 31C-308, and 31-3407.

Response: Amendment 100 to the Unit 2 FSAR will remove the “C” designator from “31C-308” in Table 6.2.4-1 for penetration X-65. It was also noted that penetration X-64 had an unnecessary “C” in “31C-306.” It will also be removed. These valve numbers were verified with CCD 2-47W865-5.

d) Penetration X-66 entry lists valves 31-326, 31-327, and 31-3392 while Table 6.2.4-1 detail sketch shows valves 31-326, 31C-327, and 31-3392.

Response: Amendment 100 to the Unit 2 FSAR will remove the “C” designator from “31C-327” in Table 6.2.4-1 for penetration X-66. This valve number was verified with CCD 2-47W865-5.

e) Penetration X67 entry lists valves 31-330, 31-329, and 31-3378 while Table 6.2.4-1 detail sketch shows valves 31-330, 31C-329, and 31-3378.

Response: Amendment 100 to the Unit 2 FSAR will remove the “C” designator from “31C-329” in Table 6.2.4-1 for penetration X-67. This valve number was verified with CCD 2-47W865-5.

9. Table 6.2.6-3, “Valves Exempted From Type C Testing”, has two inconsistencies, for which the staff needs clarification:

a) Penetration X-19A lists valves 63-072 and 72-044 while Table 6.2.4-1 shows valves 63-72 and 72-44.

Response: Amendment 100 to the Unit 2 FSAR will change the valve identifiers for penetration X-19A in Table 6.2.6-2 from “63-072” and “72-044” to “63-72” and “72-44,” respectively. These valve numbers were verified with CCD 2-47W812-1.

b) Penetration X-19B lists valves 63-073 and 72-045 while Table 6.2.4-1 shows valves 63-73 and 72-45.

Response: Amendment 100 to the Unit 2 FSAR will change the valve identifiers for penetration X-19B in Table 6.2.6-2 from “63-073” and “72-045” to “63-73” and “72-45,” respectively. These valve numbers were verified with CCD 2-47W812-1.

Preliminary RAIs for FSAR 6.2.5 (taken from e-mail from NRC dated 05/10/2010)

FSAR Section 6.2.5

6.2.5 - 1. Provide a description of how each of the criteria 1 through 8 of Regulatory Guide (RG) 1.7 Revision 3, Section C.2.1 for commercial grade hydrogen analyzer are met.

Response: Unit 2's Hydrogen Analyzer is manufactured by Meggitt Safety Systems, Inc. (MSSI).

1. Survivability

   The sampling system is comprised of a combination of components qualified by test for 1E safety related system and commercial components. Wetted components within the sample loop (components exposed to containment sample) are primarily qualified by test. Exceptions include the use of commercially available solenoid valves and a condensate trap. Materials of construction for these two items consist of stainless steel (SST) and ethylene propylene diene monomer (EPDM) for the seals. Both of these materials have been tested extensively for use within our 1E systems. Although the sample pump housing is smaller, all of the components that are exposed to the sample are identical to those used in the 1E sample pump. The pump motor is also smaller (1/2 hp) but continues to provide margin. Testing under the worse case combination of conditions has proved the maximum load the pump imposes is <1/3 hp.

   Mechanical integrity of the system was demonstrated by test (Seismic Test Report ER 11441) in 2009.

2. Power

   The operating code is stored on an Electrically Erasable Programmable Read-Only Memory (EEPROM) included with the CPU. Battery backup is also provided with the CPU. Power is provided from the 480V Reactor Vent Board 2A which is a diesel backed source.

3. Quality Assurance

   Meggitt Safety Systems operates under a documented Quality Assurance program that meets the requirements of 10 CFR 50, Appendix B and is structured around the guidelines provided by ANSI/ASME NQA-1, 2000. Amended Rule Continuous Air Monitor System (CAMS) was designed and manufactured within this QA program.

Meggitt is a major supplier of Safety related equipment in the US, Mexico, Europe and Korea. This equipment encompasses hydrogen monitoring systems, and silicon dioxide insulated instrument and control signal cables (including 10 CFR 50, Appendix R Replacement cables).

In the course of being an equipment and services supplier of safety related equipment, Meggitt has been audited by many US utilities with the most recent being a Nuclear Procurement Issues Committee (NUPIC) audit of the program. Meggitt is committed to maintaining this prominent position as a quality supplier of critical, post accident monitoring equipment.

Meggitt Safety Systems is an ISO 9001, 2001 certified supplier.

4. Display/Recording

   The system includes an internal trend function for the most recent 24 hour period and results of the most recent 5 automatic calibrations. Storage beyond that described herein is outside the scope of the system. Analog output signals are provided suitable for user's chart recorders.

5. Range

   The MSSI analyzer's internal operating range is 0 to 100% Hydrogen. The user specifies the desired operating range of the analog output signal to be as low as 5% full scale up to 100% full scale in minimum increments of 1%.

6. Servicing/Calibration

   Meggit specifies periodic testing of the system including an instrument channel function that is based on field experience. Access codes provide for administrative control. All of the maintenance activities may be performed during plant power operation.

7. Human Factors

   The human interface for the system is a touch screen display for which the screens and their organization have been well received by the industry. The sample station and system control electronics have been designed to provide good access for maintenance activity as well as convenient test points. Rapid resolution of issues is enhanced by diagnostic features included within the software. A key feature includes listings of current and historical alarms with recommended steps for their resolution. Logic and Digital to Analog Converter (DAC) Drive screens provide a convenient method for verifying control logic and analog signal functions.

8. Direct Measurement

   MSSI CAMS use electrochemical sensing for detecting hydrogen (or oxygen) concentration. The cells are specifically designed to respond only to hydrogen (or oxygen). This is achieved through selection of the electrode materials and the electrolyte. For example, hydrogen entering the hydrogen sensor will be ionized, releasing electrons. The number of electrons being exchanged is directly proportional to the number of hydrogen molecules entering the sensor. Since the number of molecules entering the sensor is directly related to hydrogen partial pressure, the exchange of electrons between electrodes provides a signal directly proportional to hydrogen partial pressure.

6.2.5 - 2. Describe the approach for demonstrating equipment survivability in the beyond design basis accident environment conditions inside the containment. RG 1.7 Revision 3, Section C.2.1, item (1) "Equipment Survivability identifies that the acceptable approaches for demonstrating equipment survivability are described in Chapter 19 of references 9 and 11 given in the RG.

Response: In Supplements 4 and 5 to the Sequoyah (SQN) SER (NUREG-0011), the NRC staff found the interim hydrogen ignition system to be an acceptable means for hydrogen control for degraded core accidents. The operating licenses of SQN 1 & 2 were conditioned, however, on the basis that TVA continue research programs on hydrogen control measures for containment integrity and equipment survivability.

TVA, in cooperation with Duke Power and American Electric Power, continued studies and eventually decided that deliberate ignition systems were the best option for controlling hydrogen. In the early 1980s, TVA developed, and submitted to the NRC, an extensive evaluation of equipment subjected to the hydrogen burn. This was documented in letters dated June 2, 1981; December 1, 1981; and December 9, 1982. TVA compared analytically and experimentally determined thermal responses of essential equipment with their qualification temperatures. The NRC concluded in SSER6 that the issues of hydrogen control and equipment survivability during postulated degraded-core accidents were satisfactorily resolved, subject to the addition of 4 more igniters in upper containment. SQN then added 4 igniters bringing the license condition to a close.

Watts Bar Unit 1 also added 4 igniters and performed analyses on the production and accumulation of hydrogen within containment. Watts Bar Unit 1 however based much of its analysis on its sister plant, SQN. All three units are 4-loop Westinghouse pressurized water reactors with ice condenser containment systems that are nearly identical in design. Two plant differences noted by the NRC were the higher spray system design flow rate and the non-existence of vacuum breakers at Watts Bar. Based on the similarity of the igniters, the air return fans and the hydrogen monitors, the NRC staff concluded that a plant specific analysis of degraded-core accidents was not necessary for Watts Bar Unit 1. The staff found that Watts Bar Unit 1 met the requirements of 10 CFR 50.44 in SSER8.

Since the Watts Bar Unit 2 design is nearly identical to the SQN units and Watts Bar Unit 1, the approach that SQN used to demonstrate equipment survivability is applicable to Watts Bar Unit 2 in lieu of a plant-specific analysis. An overview of Sequoyah's extensive survivability analysis follows.

<u>Essential Equipment</u>

The selection of equipment that must survive a hydrogen burn was based on that component's function during and after an accident. The four equipment categories were:

(1) Systems mitigating the consequences of the accident

(2) Systems needed for maintaining integrity of the containment pressure boundary

(3) Systems needed for maintaining the core in a safe condition

(4) Systems needed for monitoring the course of the accident

The list of equipment was then limited to equipment most sensitive to temperature change. Items with low heat capacity, items that contained heat sensitive components or items located in containment were determined to bound all the items originally on the list. These items were selected for an evaluation of their thermal response in a hydrogen burn environment:

(1) Mitigating Systems

- 1.1 Hydrogen Igniters
- 1.2 Air Return Fans (ARFS)
- 1.3 Associated Power and Control Cables
- 1.4 Hydrogen Recombiners

(2) Systems Maintaining Containment Pressure Boundary

- 2.1 Air Locks and Equipment Hatches
- 2.2 Containment Isolation Valves including Hydrogen Sample Valves
- 2.3 Electrical Penetrations
- 2.4 Gaskets and Seals for Flanges

2.5 Electrical Boxes

(3) Systems Maintaining Core Safety

3.1 Reactor Vessel Vent Valves (PORVs)

(4) Monitoring Systems

4.1 Steam Generator, Pressurizer and Sump Water Level Transmitters

4.2 Core Exit Thermocouples

4.3 Reactor Coolant System Pressure Transmitters

4.4 Hot Leg RTDs

4.5 Cold Leg RTDs

4.6 Reactor Vessel Level System

4.7 Associated Cables (in conduits and exposed)

4.8 Junction Boxes

4.9 Operators on Solenoid Valves

4.10 Hydrogen Analyzers

(1) Igniter Assemblies

(2) Barton Transmitters

(3) Igniter Power Cables

(4) Thermocouple Cables

(5) Resistance Temperature Detector (RTD) Cables

The NRC staff reviewed the criteria for selecting the equipment and the rationale for bounding the remaining equipment and found them to be acceptable.

<u>Thermal Environment Response Analysis</u>

The CLASIX computer code was used for modeling the thermal environment. Hydrogen accumulation was assumed as a result of a small break LOCA in conjunction with the loss of emergency core coolant injection but with both trains of sprays and air return fans operating. The hydrogen was assumed to reach 8 volume percent when ignition was initiated with each burn assumed to reach 85% completion. Flame propagation was assumed to have a velocity of 1 fps throughout containment with a constant adiabatic flame temperature of 1400°F. 6 burns were assumed in the lower compartment and 26 burns were assumed in the upper plenum. No burns were assumed in the upper compartment. The average time between burns in the lower compartment was 200 seconds and gas temperature reached 884°F. The average time between burns in the upper plenum was 90 seconds and gas temperature reached

1146°F. Acceptance criterion was based on the qualification temperature of the equipment and the duration for which the temperature was maintained.
The models used in the analysis were verified by comparing calculated results derived from other accepted computer codes. Heat transfer inside equipment was determined using the HEATING 5 computer code (ORNL). The COCO computer program was used for pressure transmitter response. Thermocouple response was compared to the studies performed by Fenwal in 1980. Sandia performed independent verification of TVA's analyses and concluded they yielded conservative results. The thermal response of igniter cable was determined experimentally at Singleton Laboratory.

Pressure Effects

The pressure profile inside containment during a hydrogen burn was obtained from a CLASIX analysis with a 12 fps flame speed. With deliberate ignition, the highest pressure did not exceed pressures used during qualification testing. It was also shown that the strain on the blades of the air return fans, because of the pressure differential between the upper and lower compartments, was precluded by the backdraft dampers.

Staff Conclusions

After reviewing TVA's analyses and experimental investigations, the NRC staff concluded that all the equipment required to ensure safe shutdown and containment integrity was able to survive the environment created by the burn of hydrogen in a degraded core accident. The staff acknowledged that the peak containment pressure, assuming a broad range of accident scenarios with conservative assumptions, would remain well below the containment pressure capacity. The staff's concern that the igniters might not initiate a lean mixture of hydrogen in a spray environment was resolved by the addition of 4 igniters.

6.2.5 - 3. RG 1.7, Section C.3 states that all containment types should have an analysis of the effectiveness of the method used for providing a mixed atmosphere. This analysis should demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity. In addition, the footnote 2 in the RG states "The NRC staff believes that current lumped parameter analytical codes may overestimate mixing processes (in particular, natural convection). Applicants should substantiate the applicability of these codes to their analyses through sensitivity studies, validation with data, or other means."

Describe the analysis performed and the results obtained which demonstrates the effectiveness of the proposed method for providing a mixed atmosphere during a beyond design basis accident. In addition, describe the approach taken to demonstrate that mixing is not overestimated by the code used for the analysis.

Response: In Supplement 3 to the Sequoyah (SQN) SER (NUREG-0011), analyses indicated that hydrogen released during a degraded core accident would be reasonably well mixed by the time it left the lower compartment. These analyses were cursory in nature and did not quantitatively characterize hydrogen mixing and distribution within the ice condenser containment. During this time, the Electric Power Research Institute (EPRI) was engaged in a hydrogen research program attempting to more effectively show that large hydrogen gradients would not occur during a degraded core accident. As part of this research, the Hanford Engineering Development Laboratory (HEDL) performed a series of large scale tests.

The mixing tests were conducted at HEDL's Containment Systems Test Facility (CSTF). This facility had a vessel that was 67 ft tall and 25 ft in diameter. The upper compartment was prone to better mixing because of containment sprays therefore the modeling emphasis was on the lower compartment. Atmospheric temperatures, velocities and gas concentrations were measured at several distribution points. The tests were designed to characterize hydrogen distribution for two release scenarios: (1) a 2 inch pipe break with a horizontal orientation, and (2) a 10 inch pressurizer relief tank rupture disc opening with a vertical upward orientation, both with no emergency core cooling injection. Two different release rates were investigated and the tests were performed with and without air return fans. Helium was used as the test medium in most cases for site safety.

The test results showed good mixing in the lower compartment when the air return fans operated throughout the accident. In all cases, with forced recirculation, the maximum hydrogen or helium concentration difference between all points was less than 3 volume percent and was generally on the order of 2%. The concentration differences were noted to stop increasing even before the release period was over and were less than 1 volume percent within 5 minutes after stopping the source gas.

The HEDL test results with no forced recirculation (air return fans inoperable) were inconclusive. During the gas-steam release, the maximum concentration difference between all measurement points was 2 volume percent. Following the release, however, the test compartment developed a vacuum as the steam condensed. This reverse migration coupled with the lack of forced recirculation created a concentration difference of as much as 7 volume percent. It was noted, however, that later in the test (~20 minutes after stopping the release), with or without air return fans, the test volume was well mixed with less than 1 volume percent concentration difference.

Based on the results of the HEDL tests, the NRC staff concluded that the formation of significant hydrogen concentration gradients in containment was unlikely if the air return fans operated during the accident. They concurred that operation of the igniters would maintain the hydrogen concentrations at or below the flammability limit for the duration of the accident. The staff agreed with the TVA position that detonation was not a credible phenomenon because (1) no rich hydrogen concentrations would be reached with the combination of mixing and deliberate ignition; (2) there were no high-energy sources to initiate a detonation; and (3) there were no geometrical channel obstructions that could cause flame acceleration yielding a deflagration to detonation transition (DDT). The staff found that the concurrent loss of air return fans and igniters was highly unlikely because of the redundancy of both systems. They also concluded that with the combination of air return fans and deliberate ignition, the DDT phenomenon was also highly unlikely. This was documented in SSER6.

Following these tests, TVA was asked to model broader accident scenarios to account for variable steam and hydrogen releases (i.e., steam inerting, bursts of steam or hydrogen, etc.). These scenarios included an intermediate line break with the loss of ECCS; a small line break with the loss of containment heat removal; a loss of main feedwater concurrent with a total loss of AC power; and a concurrent loss of main and auxiliary feedwater with the loss of ECCS. TVA effectively modeled hydrogen release rates of 6 lb per second with and without ice. These CLASIX computer code studies were then submitted to the NRC.

The NRC staff compared the results of the CLASIX studies with an independent study performed by Brookhaven National Laboratory (BNL). They also compared the release rates to those proposed in 46 FR 62281, “Interim Requirements Related to Hydrogen Control.” The staff found that the release rates and the accident sequences were an adequate representation of degraded core situations. They concluded that the scenarios used to develop steam/hydrogen source terms were acceptable. They also concluded that the CLASIX code was an adequate tool for modeling the ice condenser containment in a degraded core condition. This was also documented in SSER6.

The tests and studies documented in Supplement 6 of the Sequoyah SER demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause a loss of containment integrity. Additionally, the CLASIX code was found to be an acceptable tool with which to model hydrogen in containment. The Watts Bar Unit 2 containment structure is nearly identical to the Sequoyah containment structure as are the air return fans and deliberate ignition system. It is proposed, that since Watts Bar and Sequoyah are sister plants, the HEDL tests and CLASIX studies be directly applicable to Watts Bar Unit 2.

6.2.5 - 4. Refer to the first paragraph of Section 6.2.5.2.b. Describe the "Phase B isolation signal" referred to in this section.

Response: This is the containment "Phase B isolation signal" that is described in Unit 2 FSAR Sections 7.3.1.1.1 and 7.3.1.1.4. Amendment 100 to the Unit 2 FSAR will add a reference to 7.3.1.1.1.

6.2.5 - 5. RG 1.7 Revision 3, Section C.1 provides guidance for survivability of systems, structures and components (SSCs) installed to mitigate the hazards from the generation of combustible gas during a beyond-design-basis accident environment. FSAR Section 6.2.5.2, third paragraph states "Ductwork not protected by embedment is designed to withstand the LOCA environment" – i.e., does not state beyond-design-basis environment. Verify that the SSCs installed for the mitigation of the hazards of combustible gas are designed to operate in the beyond-designed basis environment or provide justification for not meeting the guidance.

Response: In the early 1980s, TVA developed, and submitted to the NRC, an extensive evaluation of equipment subjected to the hydrogen burn in a degraded core condition. This was documented in letters dated June 2, 1981; December 1, 1981; and December 9, 1982. TVA compared analytically and experimentally determined thermal responses of essential equipment with their qualification temperatures.

The selection of equipment that had to survive a hydrogen burn was based on that component's function during and after an accident. The four equipment categories were:

(1) systems mitigating the consequences of the accident;

(2) systems needed for maintaining integrity of the containment pressure boundary;

(3) systems needed for maintaining the core in a safe condition; and

(4) systems needed for monitoring the course of the accident.

The list of equipment was then limited to equipment most sensitive to temperature change. Items with low heat capacity, items that contained heat sensitive components or items located in containment were determined to bound all the items originally on the list. These items were selected for an evaluation of their thermal response in a hydrogen burn environment:

The NRC staff reviewed the criteria for selecting the equipment and the rationale for bounding the remaining equipment and found them to be acceptable. The NRC concluded in Supplement 6 to the Sequoyah SER that the issues of hydrogen control and equipment survivability during postulated degraded-core accidents was satisfactorily resolved. Since the Watts Bar Unit 2 design is nearly identical to the SQN units, the approach that SQN used to demonstrate equipment survivability is applicable to Watts Bar Unit 2 in lieu of a plant-specific analysis.

Unit 2 FSAR Sections 6.2.5 and 6.8 have been revised to remove references to the LOCA and to replace them with the beyond-design-basis accident.

Preliminary RAIs for FSAR 11 (taken from e-mail from NRC dated 03/23/2010)

Section 11

11 - 3. Table 11.2-7

a. Provide a basis for concluding that the doses to members of the public presented in the table for the year 2040, are bounding and conservative for current plant operation.

Response: Based on the 2000 Census, the population within 50 miles in 2000 is estimated to have been 1,064,513, indicating that the area around the site has been growing faster than projected. Based on this trend, the population in the year 2040 is projected to be 1,519,000 within 50 miles. Taking the ratio of the year 2040 population to the year 2000 population, results in a growth rate of 1.42. The Year 2040 doses for the water supplies are bounding and conservative since the population doses were multiplied by the population growth factor of 1.42.

b. Verify that the individual doses listed in the table are to the maximum exposed individual in each group.

Response: The individual doses in the table were verified using a Watts Bar computer software program, "Quarterly Water Dose Assessment" (QWATA) that calculates release doses. QWATA uses the methodology specified in Reg. Guide 1.109 to calculate radiation doses from potable water, aquatic food, and shoreline deposits to the maximum exposed individuals in each of four age groups.

11 - 4. Verify that the land-use census that is reflected in Table 11.3-9 is still valid or provide a basis for concluding that the analysis based on this information is bounding and conservative.

Response: When the NRC reviewed the FSAR, the table of interest was Table 11.3-9; a later amendment to the Unit 2 FSAR resulted in this table being re-numbered as Table 11.3-8.

The land-use data currently contained in Table 11.3-8 is not valid. Amendment 100 to the Unit 2 FSAR will correct the land-use census data in Table 11.3-8.

11 - 7. Describe the source term used to calculate the doses listed in Table 11.3-11. Amendment 95 resulted in lower values for the Total Body and Skin doses. Describe what factors changed with Unit 2 operation that resulted in these lower revised values.

Response: The source term used to calculate the doses listed in Table 11.3-11 is the same as that discussed in the response for **RAI 11 - 3**.

Factors that resulted in the differences in dose values in Table 11.3-10 (Amendment 99 version) versus the dose values in Table 11.3-11 (Amendment 95 version) are as follow:

For Amendment 95, the information contained in Table 11.3-11 was for two (2) units operating with one (1) unit containing TPBARs.

For Amendment 99, the information in Table 11.3-10 was for only 1 unit operating without TPBARs.

Another factor that resulted in different doses was the land use survey used in Amendment 95 contained higher feeding factors for the time cows were grazing on pasture.

Further, Amendment 99 used updated X/Q, D/Q, and joint frequency distribution tables for the period from January 1986 to December 2005, whereas Amendment 95 used X/Q, D/Q, and joint frequency distribution tables for the period from January 1974 to December 1993.

Finally, the 50-mile population was updated since Amendment 95 and was used in determining values for Table 11.3-10 in Amendment 99.

Preliminary RAIs for FSAR 11.3 and 11.4 (taken from e-mail from NRC dated 04/28/2010)

<u>Section 11.3</u>

11.3 - 02 The references in Section 11.3.2 to Tables 11.3-4/5 appear incorrect. A review of the tables indicate the reference to 11.3-4 should be 11.3-3 and 11.3-5 should be 11.3-4 and 11.3-5 (these two tables appear to be sheet 1 and sheet 2 of the same table or some of the sheets are missing)

Response: Amendment 98 to the Unit 2 FSAR corrected “Table 11.3-5” to “Table 11.3-4” and “Table 11.3-4” to “Table 11.3-3.” Since these were considered to be editorial changes, no change bars were provided, and the amendment level remained the same.

11.3-03: Section 11.3.3.1 under "Waste Gas Compressors" did not include a revision base for the deletion of: "Each unit is sized for 40 gpm."

Response: Amendment 95 to the Unit 2 FSAR deleted this information because the equivalent Unit 1 UFSAR portion did not include this information. This change was considered to be editorial. Additionally, this information is provided in Unit 2 FSAR Table 11.3-1 (Gaseous Waste Processing System Component Data).

11.3-05: Section 11.3.3.2, "Instrumentation Design." last paragraph. Confirm that this paragraph accurately describes the operation of this instrument. Specifically address the difference between how the Unit 2 instrument operates compared to how the Unit 1 instrument operates as described in the Unit 1 UFSAR.

Response: The last paragraph of Unit 2 FSAR 11.3.3.2, “Instrumentation Design,” accurately describes the operation of the automatic sequential gas analyzer. As stated in the “Auxiliary Services” portion of Unit 2 FSAR 11.3.2, the auxiliary services portion of the Gaseous Waste Processing System includes two automatic gas analyzers. The automatic sequential gas analyzer monitors several sample points and the second analyzer monitors the operating gas compressor. These two analyzers are common plant equipment for both Unit 1 and Unit 2 operation. Since these are common monitors, there is no difference between Unit 1 and Unit 2.

<u>Section 11.4</u>

11.4-01: Table 11.4-1, "Process and Effluent Radiation Monitors - Liquid Media," includes the "Steam Generator Blowdown Liquid Sample Monitor" and the "Boric Acid Evaporator Condensate Monitor." Both of these monitors have a footnote that states "Deleted by Amendment 95." Explain why these monitors are being deleted and why are they still included in the table, if not to be installed in the plant.

Response: The Steam Generator Blowdown Liquid Sample Monitor was isolated in Unit 1 by DCN 29903. Monitor RE-90-124 was to be used solely to determine which SG has a leak during an SGTR event. In place of the monitor, grab samples provide a quicker determination.

The Boric Acid Evaporator Condensate Monitor was already deleted in Unit 2 by Unit 1 DCNs 21582 and 51426. The monitors were abandoned in place by DCN 21582 and physically removed by DCN 51426. Abandoning of the monitors is part of the original Unit 1 licensing bases.

Both Unit 1 DCNs were used as design input to EDCR 52339. While no physical work was required, EDCR 52339 included a FSAR change package that removed the monitor entries from the Unit 2 FSAR.

Amendment 98 to the Unit 2 FSAR deleted the entries.

ENCLOSURE 1

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

Preliminary RAIs for FSAR 12 (taken from e-mail from NRC dated 03/25/2010)

Chapter 12

12 - 4. As required by 10 CFR 20.1406, describe the Watts Bar Unit 2 design features and operating procedures that will minimize, to the extent practicable, contamination of the facility and the environment to facilitate decommissioning.

Response: Design features and operating procedures that: 1) minimize contamination to the facility, 2) facilitate the eventual decommissioning of the plant, and 3) minimize radioactive waste are contained in Unit 2 FSAR Section 12.3, "Radiation Protection Design Features". Plant physical attributes and procedures that are described in this FSAR Section are consistent with 10 CFR 20.1406.

12 - 9. Provide a description of the radiation monitoring in areas where reactor fuel is handled or stored sufficient to demonstrate compliance with the requirements of 10 CFR 70.24 or 10 CFR 50.68

Response: The referenced CFRs are requirements related to criticality monitors for areas where reactor fuel is handled or stored. NRC issued an exemption from the requirements of 10 CFR 70.24 as part of the Unit 1 operating licensing. See the following excerpt from section 2.D.(2) of the Unit 1 operating license, which has been incorporated into the Unit 1 Technical Specifications:

> "2.D.(2) The facility was previously granted an exemption from the criticality monitoring requirements of 10 CFR 70.24 (see Special Nuclear Material License No. SNM-1861 dated September 5, 1979). The technical justification is contained in Section 9.1 of Supplement 5 to the Safety Evaluation Report, and the staff's environmental assessment was published on April 18, 1985 (50 FR 15516). The facility is hereby exempted from the criticality alarm system provisions of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license."

Since the new fuel and spent fuel storage areas are common to both units, it is concluded that based on the above, criticality monitors are not required for Watts Bar in areas where the fuel is handled or stored. This is also consistent with our application for Special Nuclear Material License dated November 12, 2009 (ADAMS Accession No: ML100120487).

ENCLOSURE 1

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

Preliminary RAI for FSAR 14 (taken from e-mail from NRC dated 03/25/2010)

<u>Chapter 14</u>

14 - 1. In NUREG-0847, SSER 16, dated September 1995, the staff stated in Section 14.2, "Preoperational Tests," Item 11, that …"Before issuance of an operating license for Unit 2, however, the applicant would have to demonstrate the capability of each common station service transformer to carry the load required to supply ESF loads of one unit under LOCA conditions, in addition to power required for shutting down the non-accident unit." However, Table 14.2-1 (Sheet 48 of 89) of Amendment 97 to the Watts Bar FSAR for the AC Power Distribution System Test Summary, does not incorporate this additional language in the Test Method section of the test description. Provide a discussion specifically addressing this SSER condition for Unit 2, given the scenario of having both units operational.

Response: Amendment 100 to the Unit 2 FSAR will correct the test description and Acceptance Criteria.

ENCLOSURE 1

Response to Preliminary RAIS and 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)]

Reactor Systems (SRXB)

SRXB 1. (5.2.2.4.2) Section 5.2.2.4.2 of FSAR Amendment 97, Pressure Transient Analyses, includes an evaluation of low temperature overpressure transients (Section 5.2.2.4.2.1), but no evaluation of at-power overpressure transients. Please provide an evaluation of at-power overpressure transients, consistent with the guidelines of Section 5.2.2 of NUREG-0800.

Response: Amendment 100 to the Unit 2 FSAR will add the following:

5.2.2.4.2.1 At-Power Overpressure Transients

For overpressure protection during power operation, the relief valves are provided with sufficient capacity to preclude actuation of the safety valves during normal operational transients, when assuming the following conditions:

a. Reactor is operating at the licensed core thermal power level.

b. RCS and core parameters are at values within normal operating range that produce the highest anticipated pressure.

c. All components, instrumentation and controls function normally.

The two PORVs are designed to limit the pressurizer pressure to a value below the high-pressure reactor trip setpoint for all design transients up to and including a 50% step load decrease with steam dump actuation. Isolated output signals from the pressurizer pressure protection channels are used to control pressurizer spray and the PORVs in the event of an increase in RCS pressure. The PORVs are pilot actuated valves which respond to pressure signals or to manual control. They provide a means for venting noncondensible gases or steam from the pressurizer which may impair stabilization of the RCS following a design basis event. They also provide a means to depressurize the RCS following a steam generator tube rupture event by reducing primary to secondary break flow as well as increasing safety injection flow to refill the pressurizer.

The pressurizer safety valves prevent RCS pressure from exceeding 110% of system design pressure, in compliance with ASME Nuclear Power Plant Components Code. These are totally enclosed pop-type, spring loaded valves and are self actuated by direct fluid pressure action and back-pressure compensation designed to ASME Boiler and Pressure Code, Section III. The combined capacity of two of the three safety valves is greater than or equal to the maximum surge rate resulting from the complete loss of load due to a turbine trip concurrent with the complete loss of main feedwater, all without a reactor trip or any other control.

A rise in coolant temperature can cause an insurge to the pressurizer. Pressurizer spray provides a method to decrease the rate of steam production in the pressurizer as spray injection condenses the steam at a faster rate than it is generated. The spray line enters the pressurizer at the top and terminates in the spray nozzle inside the unit. The spray rate is regulated by a PID controller which has remote overrides. In parallel with the spray valves are manual throttle valves. Temperature sensors in each spray line alert the operator of insufficient bypass flow. The spray rate is selected to prevent pressurizer pressure from reaching the PORV setpoint during a step load reduction of ten percent from full load.

The pressurizer is provided with heaters and their primary function is to heat and maintain water in the pressurizer at the saturation temperature corresponding to the operating pressure. The heaters limit the pressure decrease resulting from a drop in average coolant temperature which, during unloading, causes an outsurge from the pressurizer. The heaters are actuated automatically during insurges and outsurges and they also have manual overrides.

The function of the pressurizer relief tank (PRT) is to condense and cool the discharge from the pressurizer safety and relief valves. Steam is discharged into the PRT through a sparger pipe under the level of the water. The tank is designed to condense and cool a discharge of steam equal to 110% of the volume above the full-power pressurizer water level set.

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

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 - 2. Identify which control rods are being used, WBN Unit 2 FSAR makes multiple reference to both Ag-In-Cd control rods and B_4C with Ag-In-Cd tips. If WBN Unit 2 is transitioning from B_4C with Ag-In-Cd tips (as approved in WBN Unit 1) to solely Ag-In-Cd control rods, provide justification for such a transition.

Response: Westinghouse Field Change Notice FCN-WBTM-10794 will install Ag-In-Cd rod control cluster assemblies (RCCAs) at Watts Bar Unit 2. The previous design was the L-106A-HDR, heavy drive rod with B_4C RCCAs. The new design is the standard L-106A drive rod with a modified coupling to mate with Ag-In-Cd RCCAs. The new drive rods will be installed after RCCA installation but prior to the reactor vessel head installation. The associated safety analyses reported in the Unit 2 FSAR assume the use of the Ag-In-Cd control rod design.

Amendment 100 to the Unit 2 FSAR will revise applicable portions of the FSAR to reflect the change in control rods.

Chapter 4.2.2

SNPB 4.2.2 - 1. In WBN Unit 2 Amendment 95 Section 4.2.2.2 under the heading 'Upper Core Support Assembly' (p 4.2-25), do the support columns also contain the thermocouple supports?

Response: No. The core exit thermocouples are located at the top of the Incore Instrument Thimble Assemblies (IITA) which are inserted from the bottom of the core. The IITAs are supported by the fuel assembly into which they are inserted.

Chapter 4.2.3

SNPB 4.2.3 – 2. Does WBN Unit 2 have any part-length CRDMs?

Response: Watts Bar 2 will not use part-length CRDMs. Westinghouse has completed the modifications to Unit 2 to remove 8 part-length CRDMs drive rods and to install 8 guide tube covers at the top of the reactor internals under the part length CRDM housings.

SNPB 4.2.3 – 3. FSAR Amendment 95 for WBN Unit 2 Table 4.1-1 (page 4.1-6) indicates the control rods for WBN Unit 2 are Ag-In-Cd. FSAR Amendment 7 for WBN Unit 1 Table 4.1-1 indicates the control rods for WBN Unit 1 are B_4C with Ag-In-Cd tips. Amendment 95 for WBN Unit 2 Section 4.2.3.2.1 under the heading 'Rod Cluster Control Assembly' indicates the control rods are identical to WBN Unit 1, B4C pellets, which are stacked on top of the extruded Ag-In-Cd slugs. Are the control rods in WBN Unit 1 and WBN Unit 2 the same, B_4C with Ag-In-Cd tips, or does WBN Unit 2 have different control rods as indicated in Table 4.1-1 of its FSAR? Additionally, make the appropriate updates to WBN Unit 2's FSAR.

Response: See the response to RAI **SNPB 4.1 - 2**.

Additionally, Amendment 95 to the Unit 2 FSAR corrected the design of the control rods to reflect the Ag-In-Cd design.

Unit 1 is in the process of transitioning to this design such that both units will be using the Ag-In-Cd design.

<u>Plant Systems (SBPB)</u>

SBPB 10.4.1-1 In the 1982 Safety Evaluation (SE) for Watts Bar (NUREG-0847), the NRC staff observed that the three pressure zones for the main condenser are designed to produce a turbine back pressure of 1.5 (low pressure -LP), 2.15 (intermediate pressure -IP), and 3.065 (high pressure -HP) inches of mercury for Units 1 and 2. In Amendment 7 of the WBN Unit 1 FSAR, those values are changed to 1.63 LP, 2.38 IP, and 3.40 HP. For the proposed FSAR for WBN Unit 2, the values for the three pressure zones are 1.92, 2.70, and 3.75 respectively. The NRC staff requests that TVA explain why the values for WBN Unit 1 and the proposed WBN Unit 2 pressure zones for the main condenser are different from the values indicated in the original SE.

Response: The three zone pressures (1.5, 2.15, and 3.065 inches of mercury) in the 1982 SE were the original design pressures for the Unit 1 and Unit 2 main condensers. The pressures were provided by the condenser vendor based on 100% condenser cleanliness and the condenser heat duties associated with the original licensed thermal power (OLP). The design pressures of 1.63, 2.38, and 3.40 inches of mercury (identified in the Condensate System Design Description N3-2-4002) are based on 70 °F CCW inlet temperature, 95% clean tubes and HEI Design Mode, 1995 Revision.

In the period since 1982, several modifications have been made to Unit 1, which have affected main condenser performance: 1) re-tubing the condenser with Sea-Cure stainless steel tubes to remove copper from the secondary side for steam generator preservation; 2) rerouting of the condenser zone cascading scheme (Zone "A" bypass to Hotwell) to mitigate the impact of lower than expected cooling tower performance to maintain hotwell temperature to the condensate demineralizers below process limits); and 3) a 1.4% power uprate was approved which increased the heat duty on each condenser zone. In addition, with a 115-foot length, stainless steel tubing the condenser cleanliness cannot be maintained at 100% cleanliness (typically 80 – 85%). Note that the Amendment 6 heat balance (Figure 10.1-2) which is included in Unit 1 UFSAR Amendment 7 utilizes zone pressures of 1.92, 2.38, and 3.75 inches of mercury. As shown on the figure, these zone pressures are based on a condenser cleanliness of 80%.

The Unit 2 condenser has also been re-tubed with the same material (Sea-Cure) as Unit 1, but it will not receive the Zone “A” bypass modification. Following the bypass modification on Unit 1, supplemental condenser cooling water (SCCW) from the Watts Bar Lake was mixed with the Unit 1 cooling tower discharge. This reduced the condenser cooling water inlet temperature and thereby helped reduce the hotwell temperature. Unit 2 will receive significant benefit from SCCW; therefore, the Zone “A” bypass modification will not be made. In the proposed Unit 2 FSAR, the condenser zone backpressures were developed consistent with the secondary side thermal model generated for a future (post-license) 1.4% power uprate, rather than OLP. In addition, the Westinghouse supplied HP and LP turbines and the Moisture-Separator-Reheaters (MSRs) have been replaced with more efficient units supplied by Siemens. With no operational data for Unit 2, the zone pressures utilized for Unit 1 were provided to Siemens for the Unit 2 replacement turbine design. These same zone pressures were carried over to the initial heat balances for Unit 2. There are other minor design differences between the turbine systems that also contribute to the unit differences.

SBPB 10.4.1-2 In Amendment 7 to the WBN Unit 1 FSAR, TVA shows the Birmingham Wire Gauge (BWG) for the balance of tubes to be 22 BWG, whereas in the proposed WBN Unit 2 FSAR, this value is shown as 12 BWG. The NRC staff requests that TVA explain why WBN Unit 2 has a lower BWG value than WBN Unit 1.

Response: Amendment 98 to the Unit 2 FSAR corrected “12 BWG” to “22 BWG.” Since this was a correction of a typographical error (22 BWG is correct per EDCR 52320), the amendment level on the page did not change.

SBPB 10.4.1-3 In the original SE for WBN Unit 1 and 2, copper-nickel tubes were used to minimize corrosion and erosion of condenser tubes. In Amendment 7 to the WBN Unit 1 FSAR and the proposed WBN Unit 2 FSAR, the material is listed as SEACURE for tubes. The NRC staff requests that TVA confirm that the SEACURE material is the replacement for the copper-nickel tubing that was originally used for the condenser tubes.

Response: TVA confirms that the Unit 2 Main Condenser has, in fact, been retubed with SEACURE stainless steel material identical to that used on the Unit 1 Main Condenser. Unit 2 Engineering Document Construction Release (EDCR) 52320 effectively replicates actions taken for Unit 1 under TVA Design Change Notice (DCN) M-38974-A which retubed the Unit 1 Main Condenser.

SBPB 10.4.1-4 In the original SE, the NRC staff noted that the main condenser design allows for water storage capacity for approximately 3 ½ minutes of full-load operation. In the proposed WBN Unit 2 FSAR, this value is the same. However, in Amendment 7 of the WBN Unit 1 FSAR, the condenser is shown to handle water storage capacity of 3 minutes at full-load operation. The NRC staff requests that TVA explains the deviation of the WBN Unit 1 storage capacity from the proposed WBN Unit 2 storage capacity for the main condenser.

Response: The "3 minutes of full-load operation" information contained in Amendment 7 to the Unit 1 UFSAR was determined to be inconsistent with design documentation. Amendment 8 to the Unit 1 UFSAR corrected "3 minutes of full-load operation" to read "approximately 3 ½ minutes of full-load operation."

SBPB 10.4.2-1 In the original SE (NUREG-0847), there are listed three mechanical vacuum pumps, an electrical heating coil, a HEPA filter, and a carbon absorber to comprise of the main condenser evacuation system (MCES). In Amendment 7 of the WBN Unit 1 FSAR and proposed FSAR for WBN Unit 2, TVA does not provide the full description of the components of the MCES, nor clarifies if the MCES still comprises of the same components as listed in the original SE, with the exception of the vacuum pumps. The NRC staff requests that TVA provides a description of the components for the MCES for WBN Units 1 and 2.

Response: The Main Condenser Evacuation System is shown in Unit 2 FSAR Figures 10.4-7, 10.4-9, 10.4-10, and 10.4-12. These figures show the System Components, the flow diagrams, the electrical control diagrams, and the electrical logic diagram for the Condensate System including the Main Condenser Evacuation System. The description of these components for the Condensate System is as follows:

1. The Condensate system boundaries are defined to include the following equipment:

   a. The main condenser to the interface with the turbine exhaust, and to the interface with the Condenser Circulating Water (CCW) system connections at the waterboxes.

   b. The condenser vacuum pump including the suction piping from the main condenser and the MFPT condensers, and the vacuum discharge to the turbine building roof.

c. The MFPT condenser channel and tube side, and up to the MFPT exhaust and up to the shell drain connections.

d. The condensate storage tanks including all interconnecting piping and the condensate transfer pump.

e. The hotwell pumps.

f. The condensate demineralizer pumps.

g. The condensate booster pumps.

h. The channel and tube side of the GSC, SGB Heat Exchangers and low pressure and intermediate pressure feedwater heaters.

2. At Watts Bar Units 1 and 2, the Main Condenser Evacuation System is comprised of: Condenser Vacuum Pumps, Radiation Monitors for the exhaust of Condenser Vacuum Pumps and Vents to roof.

   DCN # M-03153-A (Dated 10/1992) removed the internals for HEPA Filter, Carbon Absorber Train in the MCES and unplugged the Duct Heater. This DCN allowed the removal of the HEPA and Charcoal Filter Absorber Cartridge from its housing in the MCES. ECDR # 53306 allowed the same modifications for WBNP-2 MCES. Flow Diagrams and Physical Drawings will be updated to reflect this change for Unit 2 as part of the EDCR implementation process.

SBPB 10.4.2-2 In Amendment 7 to the WBN Unit 1 FSAR, TVA indicates that there are two types of radiation monitors for the vacuum pump exhaust. However, TVA describes the vacuum pump exhaust in the proposed WBN Unit 2 FSAR as having three types of radiation monitors. The NRC staff requests that TVA explains the reasoning for WBN Unit 2 having three radiation monitors versus WBN Unit 1 only having two radiation monitors.

Response: Unit 1 originally had three physical monitors containing four detectors monitoring the Condenser Vacuum Pump Exhaust: RE-90-99, RE-90-119, and RE-90-404. Unit 1 deleted 2-RE-90-99 with DCN 35431. The basis for removal was:

> "Calculation WBNAPS3-048 Rev.8 concluded that the multiple noble gas detectors used to monitor the condenser vacuum pump exhaust stream provide sufficient range overlap without the RE-90-99 mid-range channel. This channel is therefore declared redundant and may be removed from the Radiation Monitoring System without loss of required monitoring capability."

The Unit 1 RE-90-404 is an accident monitor that is in actuality made up of two separate radiation detectors with overlapping ranges. This dual detector arrangement is what allowed Unit 1 midrange monitor RE-90-99 to be deleted. Because the monitor currently used for RE-90-404 in Unit 1 is obsolete, two new single-detector monitors are being installed in Unit 2. The two monitors are required to maintain the necessary range overlap. Installing two separate monitors requires that new equipment identifiers be assigned. This results in the appearance that Unit 2 has three detectors and Unit 1 has only two.

RAIs for FSAR Chapters 8 and 9 [from NRC letter dated 07/12/2010 (ADAMS Accession No. ML101530354)]

Section 8.1- Electric Power – Introduction

8.1 - 1. Final Safety Analysis Report (FSAR) Section 8.1.5.3, "Compliance to Regulatory Guides and Institute of Electrical and Electronics Engineers (IEEE Standards)," states that WBN Unit 2 complies with Regulatory Guide (RG) 1.32, Revision 0, "Criteria for Power Systems for Nuclear Power Plants;" RG 1.81, "Shared and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants," Revision 1, IEEE Std 308-1971," "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations," in meeting NRC regulations in Title 10, *Code of Federal Regulations* (10 CFR) Section 50, Appendix A General Design Criteria 5 and 17. FSAR Section 3.1.2 states that the preferred and emergency electric power systems are shared. Since NRC staff has not previously reviewed the capability of the preferred and emergency electric power systems for dual-unit operation, provide an executive summary of the analysis to support the following design requirements.

Response: As reflected on the key diagrams and the single line diagrams, the Watts Bar Auxiliary Power System (APS) was originally designed for two unit operation. However, due to indefinite deferral of Unit 2, detailed analysis of the adequacy of APS was performed for Unit 1 only during the Unit 1 licensing effort. Therefore, the current APS analysis evaluates the system to support Unit 1 only although it takes into account Unit 2 busses, boards and loads required for Unit 1 operation and safe shutdown.

The purpose of the AC APS analysis is to determine its adequacy to support two unit operation. The analysis was performed using Electrical Transient Analyzer Program (ETAP) Version 5.5.6N. The evaluation includes steady state and transient voltages, equipment short circuit currents, equipment capability and bus loading. The analysis was performed for various configurations and includes steady state conditions, dynamic motor starting, and short circuit current and degraded voltage analysis. The detailed analysis was limited to safety related equipment and components powered from the safety related boards. The scope of this analysis included the Common Station Service Transformers (CSST A, B, C and D), 6.9kV Shutdown Boards, 6.9kV Start Buses, 6.9kV Common Boards, 6.9kV Unit Boards, the downstream 6900V-480V transformers, 480V Distribution Systems loads, and all interconnections. The answers to specific NRC questions below is based on this analysis:

a. A dual-unit trip as a result of abnormal operational occurrence

Response: This analysis is enveloped by the analysis performed while postulating an accident in one unit and concurrent orderly shutdown of the second unit (See details in the response to **RAI 8.1 - 1.c**.).

b. Accident in one unit and concurrent shutdown of the second unit (with and without offsite power)

Response: Analysis with offsite power available:

Unit 1 and Unit 2 share the two independent offsite power sources for normal operation and safe shutdown of the plant. To review the capability of the auxiliary power system for two unit operation, the following analysis was performed:

- Normal plant configuration with CSSTs C and D supplying power to the 6.9kV shutdown boards and CSSTs A and B supplying power to non-Class 1E loads: In this scenario, each secondary winding of CSST C and D supplies power to one shutdown board only (one Train of one unit).

- One offsite circuit out of service resulting in outage of either CSSTs C&B or D&A: With an outage of one set of transformers, all loads powered by those transformers are automatically transferred to the remaining set of transformers. Thus, all shutdown boards are powered from one CSST C or D and each winding will supply power to two shutdown boards (one train of each unit)

- Substituting CSST B or A in case of an outage of CSST C or D respectively: In this case, CSST B or A will feed the manually transferred shutdown boards in addition to the other normally fed non-Class 1E loads (substitution of CSST B or A as offsite source is limited to the use either B or A at any given time, provided both CSSTs A and B are available).

- Normally when the units are operating, the unit boards are powered from the USSTs and transfer to CSSTs A & B on unit trip. Therefore, if a shutdown board is transferred to the corresponding unit board and an accident occurs, the ESF loads on the transferred shutdown board would start on the USST. Since the USSTs are unitized (there are two USSTs per unit), only one shutdown board can be transferred to one USST. Analysis has been performed to verify that the USSTs are capable to start all ESF loads on the transferred shutdown board in addition to the non-class 1E loads on the associated RCP and unit boards.

- The analysis was performed considering “block start” of all 6.9kV and 480V motors required to start on receipt of the accident signal for the accident unit. It also included those loads on both units that are associated with the normal operation and are not automatically tripped.

The above analysis is performed with a grid voltage drop of 11kV (164-153kV) when the shutdown boards are powered from CSSTs C&D and 9kV with a minimum voltage of 153kV when powered from CSST A or B. In case of ESF loads starting on USSTs, the analysis is performed considering the generator as swing bus at 23kV since the generator remains connected to the grid and working as a synchronous condenser.

The auxiliary power system was determined to be adequate to support the above scenarios for two unit operation. The voltage recovery times were within the time limits so that 6.9kV shutdown board degraded voltage relays reset and do not separate 6.9kV shutdown boards from the offsite power source.

Analysis with onsite power (Diesel Generators)

There are four diesel generators (DGs), one each dedicated DG for 6.9kV shutdown board feeding one train of Unit 1 or Unit 2 ESF loads. The ESF loads are not shared on the diesels. The analysis with onsite power system was limited to the verification of each DG loading under accident conditions considering sequential loading of 6.9kV motors. Pre-operational testing performed on Unit 1 DG with a safety injection signal and starting of random loads demonstrated the capability of the DGs to provide adequate voltage to all required loads. Due to similarity of Units 1 and 2 diesel generators, it is reasonable to assume that the dynamic voltage response of Unit 2 pre-operational testing will be the same as previously determined during Unit 1 preoperational testing. Unit 2 pre-operational testing will validate the diesel response to sequencing of loads on the Unit 2 diesels.

c. Accident in one unit and spurious Engineered Safety Features actuation in the other unit (with and without offsite power)

Response: The ESF loads for each Unit's Train A and B are powered from a dedicated DG and battery (load group) with exception of the Turbine Driven Auxiliary Feedwater (TDAFW) Pumps and vital power (See Attachment 6.). In the case of the TDAFW pump, the controls cross to the opposite unit's battery in order to create a "Special Train." In the case of the 120VAC vital power, a division of power is supplied from each of the four batteries in order to create the required four divisions. Each DG and Battery can support the ESF loads on a safety injection signal.

The auxiliary power system and supporting analysis complies with the requirements of position C.2.b of Regulatory Guide 1.81. The design of the onsite power system is unitized such that ESF loads that would be applied due to a spurious accident signal on one unit would not adversely affect the capability of the other unit to supply accident loads. The electrical loads for shared systems such as emergency raw cooling water and component cooling are fully applied in the loading analysis for both units. Board feeder alignments for power distribution boards are required to be in the normal position for operability such that cross-unit power connections would not be an operable condition. There is no design requirement per the FSAR to be able to support an accident in one unit and a spurious ESF actuation in the other unit when supplied from off-site power. Therefore, analysis with one unit in accident and the spurious actuation of ESF loads in the second unit has not been performed.

8.1 - 2. Explain how the industry and WBN Unit 1 operating experience and the NRC generic communications have been reviewed and incorporated in the electrical design, maintenance, surveillance testing, and operations for WBN Unit 2.

Response: The electrical power system is under the operating unit's control. The operating unit reviews operating experience (OE) and NRC generic communications under the Operating Experience Program, SPP-3.9. Under this program, external and internal OE as well as generic communication is reviewed and actions developed.

<u>Section 8.2- Offsite Power System</u>

8.2 - 1. The common station service transformers (CSSTs) are described in Section 8.2.1.2 of the FSAR. It is stated that their calculated loading is well below their winding ratings for all conditions. FSAR Page 8.1-13 Position C2 states, "The shared safety systems are designed so that one load group (Train 1A & 2A or Train 1B & 2B) can mitigate a design basis accident in one unit and accomplish an orderly shutdown of the other unit." These CSSTs are shared between the WBN Unit 1 and WBN Unit 2. In view of the WBN Unit 2 loads being applied to the CSSTs along with WBN Unit 1 loads, the NRC staff requests the following information:

Note: For ease of providing a response, the NRC question is split into three parts as described under each bullet below:

- Provide an executive summary of the calculations and analyses which detail the loading for both units (or added loads of WBN Unit 2 to the existing loads of WBN Unit 1).

Response: TVA performed the APS loading analysis with the offsite power system. CSSTs C and D provide two shared sources of offsite power to the 6.9kV boards to provide power to both trains of Unit 1 and Unit 2 ESF loads. Train A of both units is powered from CSST C and train B is powered from CSST D. When both CSSTs are available, each transformer secondary winding feeds only one 6.9kV shutdown board (one train of a unit).

In addition to CSST C and D, CSST A and B, which currently provide a maintenance feed for the 6.9kV shutdown boards via 6.9kV unit boards, are retrofitted with automatic on load tap changers by DCN 52336 and may be used during transformer maintenance to provide a qualified offsite source to one train of safety related ESF loads (EDC 55945 - under issue). CSST A will be used as a substitute for CSST D when CSST D is out of service, and similarly CSST B will be used to substitute for CSST C. Thus, two CSSTs will always be available to provide two independent sources of offsite power to the 6.9kV shutdown boards. However, since CSSTs A and B also feed the plant non-safety related loads, the use of CSST A and B to provide a qualified offsite source will be limited to only one train of ESF loads with both CSSTs A and B available. CSST A and B will be used either as an immediate or as a delayed second offsite source. A delayed second source is permissible and meets the requirements of RG 1.32. The calculation also analyzes when one train of ESF loads is supplied from CSSTs A and B.

The worst case loading on any one secondary winding of CSSTs C and D with Unit 1 in accident condition (Unit 2 defueled) is calculated as 6.02MVA. With only Unit 1 operating, Unit 2's 6.9kV shutdown boards are lightly loaded (2.84MVA). With Unit 2 operating, the worst case loading on the secondary winding feeding the accident unit and non-accident unit loads is calculated as 6.12MVA and 5.07MVA respectively.

Analysis has also been performed to verify that sufficient power is available to safety loads to mitigate an accident in one unit and to safely shutdown the other unit for loss of one offsite power source, or a CSST or a Common Switchgear. In this, case one secondary winding of the remaining CSST will feed two 6.9kV shutdown boards (i.e., Train A of both units) and the second secondary winding will feed two 6.9kV shutdown boards for Train B of both units. The worst case loading on each secondary winding with one unit in accident and the other unit in concurrent controlled shutdown is calculated as 11.30MVA with primary winding load of 22.44MVA. Both primary and secondary winding loads are well within the CSST self-cooled rating of 22MVA and 33MVA respectively.

Note: ERCW pumps are being replaced. Upon final testing, slight adjustment may be required for ERCW pump loading which is less than 0.013MVA.

With one train of ESF loads powered from CSST A or B, the worst case steady state loading on CSST A or B is calculated as 67MVA which is within the FA rating of 76MVA.

The worst case loading on the (Y) secondary and primary winding (Y-winding feeding shutdown boards via unit boards) when one Train of ESF loads for each unit is simultaneously powered from CSST A or B (substituting for CSST D or C, respectively) is 39.79MVA and 67.61MVA which is within the FA rating of 48MVA and 76MVA respectively.

Refer to Attachment 6 for CSSTs loading details for various configurations.

The 6.9kV unit boards, which provide alternate supply to the 6.9kV shutdown boards, are normally powered from the USSTs. The USSTs remain connected to the 500kV system for 30 seconds after a unit trip. Therefore, it is possible that the ESF loads for the accident unit will be connected to the USSTs if CSST A or B is being used as the offsite source. In such a case, the ESF loads could "block start" on the USSTs. Therefore, an analysis has also been performed to verify that the USSTs are adequate to support the safe shutdown while powering the shutdown boards. It is determined that in this case also, the degraded voltage relays get reset in less than 5 seconds without tripping the offsite power. The loading of the USSTs is also calculated to be well within its FA rating of 20MVA.

The rating of all 6900V/480V transformers, buses and boards feeding safety related loads are adequate and no overloading was identified.

- Define the bounding conditions for maximum loading that demonstrates that the winding ratings are not exceeded.

Response: Bounding conditions evaluated in the AC APS analysis is one unit in accident and the other unit in controlled shutdown with loss of one offsite power source or a CSST (C or D) or a common station switchgear (C or D). It is concluded from the analysis that the winding ratings are not exceeded.

- Provide a summary of the calculation that demonstrates the design margin in the CSSTs with a design-basis accident (DBA) in one unit and a concurrent shut down of the other unit. The summary of the design calculations must include inputs, assumptions, and a summary of output results (with acceptance criteria) including any load creep for both units.

Response: As stated in response to the question under the first bullet, adequate design margins are available in the CSSTs under the worst case configuration. Both CSSTs C and D operate within their OA ratings even when all shutdown boards are powered from one transformer due to outage of one offsite power source. As stated earlier, both CSST A and B operate within their FA ratings when used as alternate to CSST D and C respectively. See Attachment 7 for inputs, assumptions and summary of the output results with acceptance criteria.

8.2 - 2. FSAR Section 8.2.1 states that to provide a stable voltage, CSSTs C and D have automatic high-speed load tap changers (LTCs) on each secondary, which adjust voltage based on the normally connected shutdown boards.

Provide an executive summary of the calculations/analyses that details the plant loadflow/voltage studies and operations of the load tap-changing units including a detailed discussion of the control voltage setting, the voltage control band, the time-delays for LTC operation, etc. The summary of the calculations must include inputs, assumptions, and summary of the output results (with acceptance criteria).

Response: The load flow/voltage analysis of the APS system to support two unit operation is performed under various configurations for steady state and transient conditions (See the matrix in Attachment 8 for the configurations analyzed.). This transient analysis (motor starting) is performed considering one unit in accident and the other unit in controlled shutdown mode. The short circuit analysis is performed using upper band of the LTC and considering the highest grid voltage of 169kV.

For the CSST C and D automatic LTCs are provided on each low voltage winding of the CSSTs. The optimum setting for the LTCs was previously established prior to Unit 1 start. The LTC settings are set to regulate the 6.9kV shutdown board bus voltage to 7071V (102.5%). The lower and upper setpoints of the dead band are 7132V (103.4%) and 7010V (101.6%) respectively. The analysis is performed to evaluate the minimum voltage requirements with LTC setting of 7010V and a dead band of ± 82.2V. The initial time delay for the LTC is 2 seconds and a step time delay of 1 second with each tap step of 1.25%.

CSSTs A and B have been retrofitted with an LTC provided on the primary (high voltage) winding of the CSSTs. The LTC settings are set to regulate the 6.9kV shutdown board bus voltage to 7071V (102.5%). The lower and upper setpoints of the dead band are 7132V (103.4%) and 7010V (101.6%) respectively. The analysis is performed to evaluate the minimum voltage requirements with LTC setting of 7010V and a dead band of ± 82.2V. The initial time delay for the LTCs is 1 second and step time delay of 2 seconds with each tap step of 1.05%.

Based on the analysis, adequate voltages are available at the 6.9kV and 480V buses/boards. The transient voltage on the 6.9kV shutdown boards falls below the degraded voltage due to “block start” of ESF loads on the accident unit 6.9kV shutdown boards, but the degraded voltage relays reset in 5 seconds or less without tripping the offsite power. See Attachment 8 for inputs, assumptions and summary of the output results with acceptance criteria and Attachment 8 for summary of the board voltages and degraded voltage relay reset time. Loading of all 6900V-480V transformers, safety related boards and MCCs was determined to be within the equipment rating and no overloading was identified.

<u>Section 8.3.1 – Alternating Current (AC) Power Systems (Onsite)</u>

8.3.1 - 1. FSAR Sections 8.2.2 and 8.3.1 describe the degraded voltage (27DAT, DBT, DCT) and loss of voltage relays (27LVA, LVB, LVC). The degraded-voltage relays as described in the FSAR have a voltage setpoint of 96 percent of 6.9 kV and a time delay of 10 seconds. FSAR Section 8.2.1 states that to provide a stable voltage, CSSTs C and D have automatic high-speed LTCs on each secondary, which adjust voltage based on the normally connected shutdown boards. The recent NRC Component Design Basis Inspections (CDBI) indicated issues associated with calculations to support the degraded voltage setpoints. In view of the CDBI findings, provide the below listed information with regard to WBN Unit 2.

a. Provide an executive summary of the results of calculations/analyses which detail plant load flow and voltage drop studies, and operations of the load tap-changing units including a detailed discussion of the control voltage setting, the voltage control band, time-delays associated with LTC operation, etc. The summary of the calculations must include inputs, assumptions, and summary of the output results (with acceptance criteria).

Response: See the response to **RAI 8.2 - 2**.

b. Provide a summary of the analyses (steady state and transient) that demonstrates that the above degraded voltage trip set points are adequate to protect all safety-related equipment required for design basis events and also to provide the required minimum voltage at the equipment terminal to start and run all loads consistent with the accident analysis assumptions without crediting the LTCs of the CSSTs.

Response: As a result of NRC Electrical Distribution System Functional Inspections (EDSFI) at various plants, concerns were raised about the adequacy of degraded undervoltage relays and time delay setpoints. TVA was instrumental in forming an industry working group with other utility members and developing a set of guidelines for the required analysis to determine proper degraded voltage setpoints and time delays. These recommendations were used to develop the TVA degraded voltage analysis methodology. TVA analyzed the degraded voltage protection scheme for Unit 1 based on the aforementioned methodology. The same approach was adopted to perform the analysis for Unit 2. This approach is consistent with IEEE 741-1997, “IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations” Annex A, “Illustration of concepts associated with degraded voltage protection”.

Capability to start Class 1E motors for Unit 2 has been evaluated in two ways (similar to Unit 1) as follows:

- To evaluate the plant response to a DBE, maximum (block-start) loading is applied in conjunction with the maximum grid drop down to the minimum expected grids voltage of 153kV. This dynamic motor starting analysis demonstrates that even though the shutdown board voltage drops into the degraded voltage relay operating range during the momentary voltage dip, the voltage recovers above the reset value within the degraded voltage relay time delay. This analysis also demonstrates that all equipment required to mitigate an accident receive sufficient voltage to start and accelerate within the required time.

- The analysis is performed with a maximum grid drop of 11kV in case of normal alignment from CSST C &D and with grid drop of 9kV with alternate alignment from CSSTs A & B.

- Degraded voltage analysis evaluates the capability to individually start and run class 1E motors at steady state conditions. This analysis ensures that all motors have adequate starting voltage at the upper boundary of the degraded voltage relay setpoint setting (6672V) and adequate running voltage at the lower boundary of the degraded voltage relay setpoint setting (6555V). The upper boundary was chosen because this is the lowest voltage that guarantees offsite power supply recovery from a DBE transient. The upper boundary setpoint has been revised to 6681V; therefore, the analysis performed with 6672V is conservative.

c. Provide executive summary of calculations/analyses for settings of the loss-of-voltage Relays (27LVA, LVB, LVC) provided at 6.9 kV shutdown boards. The summary of the calculations should include criteria, assumptions, and output results

Response: The Loss of Voltage (LOV) relay voltage setpoint upper limit is established to be less than, with margin, the safety bus voltage equivalent to the design calculated worst case transient voltage dip during accident loading sequence. The setpoint should be less than the TPS calculated security boundary voltage which is based on the grid voltage that is one failure away from collapse.

The LOV relay voltage setpoint lower limit is selected by evaluating operation of the APS under steady state (running) conditions, with the 6.9kV shutdown board voltage as low as possible while keeping all connected safety related motor loads above their stall voltage (greater than 70.7% of rated motor voltage for NEMA Design B motors). The lower limit must also be greater than the 6.9kV shutdown board voltage equivalent to having the lowest switchyard voltage that could be sustained without instability or collapse.

The lowest boundary of the LOV voltage relay time delay should be long enough to ride through short circuits and other short time system transients (lightning strikes, switching transients, etc.) taking into account the total sensing and clearing times for the above type of events. The time delay upper boundary should be less than the safety analysis time allowed for loss of voltage detection.

Based on the above criteria, the LOV relay lower and upper limits were calculated to be 5968V and 6060V respectively. The associated time delay relays lower and upper limits are at 0.4 seconds and 1.14 seconds, respectively.

The lowest 6.9kV shutdown board voltage which will prevent safety related motors from stalling was determined to be 5600V.

The settings of LOV relays are not impacted by Unit 2 since the design calculated worst case transient voltage dip during accident loading sequence used to determine the upper boundary setpoint is bounding based on the Unit 1 and Unit 2 analysis. Also the stall voltage of 5600V was determined to be acceptable for Unit 2 motors.

8.3.1 - 2. According to the FSAR, Figure 8.1-2A, the CSSTs, C and D normally supply Train 'A' (1A and 2A) and Train 'B' (1B and 2B) shutdown loads, respectively relating to the two units. In case of loss of either CSST, the loads fed from the corresponding CSST shutdown buses are automatically transferred to other CSST via the automatic transfer scheme.

Provide description of the automatic transfer scheme from normal to alternate source (whether fast – how many cycles etc.) as it relates to WBN Unit 2. Also, explain the transient behavior of loads that were already running on the shutdown boards.

Response: The 6.9kV shutdown boards have been provided with an automatic fast bus transfer scheme. Since all Unit 1 and Unit 2 shutdown boards are required to support Unit 1 operation, the existing bus transfer scheme is operational and is not affected by Unit 2.

The existing automatic bus transfer scheme has been evaluated. The fast transfer scheme for the 6.9kV shutdown boards from normal to alternate source is a trip signal given to the normally closed breaker with a "b" contact on the closed breaker initiating the close command for the alternate breaker. The transfer scheme uses an early "b" contact. An early "b" contact operates at the time the arcing contacts separate. Since the closed breaker gives a close command to the second breaker upon opening of the arcing contacts, the "dead bus time" is the time required for second breaker to close. The fast bus transfer can be either operator initiated (manual) or automatic (fault initiated).

The 6.9kV shutdown boards have AM-7.2-500 breakers. Per the vendor data sheet, the "dead bus time" for operator initiated fast transfer (manual) using early "b" contact is 3.1 cycles. The times given in the vendor data sheet are nominal times with a tolerance of ± 1 cycle.

For a fault initiated transfer (automatic), the sequence of events is the fault followed by the protective relay actuation, lockout relay (LOR) operation and breaker operations. The overcurrent relay operating time is approximately 4 ms and the LOR relay operates in 13 – 14 ms. The minimum time for the breaker contacts to open is approximately 40 ms resulting in a fault isolation time of 57 ms to 58 ms. The longer the time period that a board is connected to a fault the less energy the operating motors will have when transferred. The decrease of kinetic energy results in a decrease in magnitude of the motor's generated ac voltage and the resultant volts/hertz that the motors would experience on transfer. Industry standards issued recommend the resultant volts/hertz (calculated in per unit on the motor base) should not exceed 1.33 and this criterion was used to evaluate adequacy of the fast bus transfer scheme at Watts Bar.

The Fast Bus Transfer Analysis is a dynamic fast bus transfer analysis using ETAP computer software version 6.5N1. The analysis considers two fast bus transfers, operator initiated (manual) and automatic (fault initiated) with three different loading conditions: (1) normal operation, (2) accident SI-phase B, and (3) light loaded (one motor per bus). Based on the analysis, it is concluded that the that the automatic and manual 6.9kV fast bus transfers between the normal and alternate breakers are acceptable for all board loading conditions.

8.3.1 - 3. FSAR, Table 8.3-3 shows sequence of loads applied following a loss of preferred (offsite) power (from the time of closing of the generator breaker connecting the diesel generator to the shutdown board). However, in Section 8.3.1.1 (under subheading 'System Operation', it is stated that "The standby (onsite) power system's automatic sequencing logic is designed to automatically connect the required loads in proper sequence should the logic receive an accident signal prior to, concurrent with, or following a loss of all nuclear units and preferred (offsite) power." Regarding the above statement with respect to WBN Unit 2, explain the design of the automatic sequencing logic.

Response: The design of standby (onsite) power system's automatic sequencing logic for Unit 2 is identical to the logic for Unit 1. The load sequencer consists of discrete relays which are part of the individual load's breaker control circuit. In the event of a loss of offsite power, the block starting breaker closure circuit path is opened and the sequence starting path is closed using the BOX or BOY relay. Once voltage is restored to the board from the diesel generator, the sequencing timer for that load is initated by the UVX or UVY relay. Once the timer is timed out, the breaker receives a close signal.

8.3.1 - 4. In Section 8.3.1.1 of the FSAR (under the subheading "Equipment Capacities"), it is stated that "Tables 8.3-4 through 8.3-7 present the bus rating, connected load, and maximum demand load for each electrical distribution board in the standby (onsite) power system."

Because of anticipated two unit operation, the NRC staff requires the information on connected and maximum demand loads to assess the capacity and capability of the onsite distribution system for WBN Unit 2.

Response: Unit 2 FSAR Tables 8.3-4 through 8.3.7 describe the board/bus rating in kVA and do not represent the connected load or the maximum demand. This kVA rating is calculated by kVA = $\sqrt{3}$ VI, where V and I are the rated voltage and the rated current, respectively. Unit 2 6.9kV/480 boards have the same rating as the Unit 1 boards. It is verified in the APS analysis that loading on all safety related boards is within their rating, and no overloading has been identified. The loading on all boards, when powered from the standby onsite power system (diesel generators) is enveloped by the loading in the calculation referred above.

8.3.1 - 5. In Section 8.3.1.1 of the FSAR (under the subheading "Standby Diesel Generator Operation"), it is stated that "For test and exercise purposes, a diesel generator may be manually paralleled with a normal or alternate (offsite) power source. A loss of offsite power will automatically override the manual controls and establish the appropriate alignment."

Regarding the above statement with respect to WBN Unit 2, please explain what is meant by "appropriate alignment."

Response: The term "appropriate alignment" is explained as follows:

During testing of the diesel generator, should a loss of offsite power and accident occur, an accident signal will trip the DG feeder breaker. Tripping the DG breaker will automatically place the DG in asynchronous mode of operation. As soon as the offsite power supply breaker to the 6.9kV shutdown board is tripped and the undervoltage load stripping relays operate, the DG feeder breaker to the board will close and load sequencing logic will be initiated to load the accident loads.

8.3.1 - 6. In order to verify the adequacy of the diesel generator capacity stated in Section 8.3.1.1 of the FSAR for WBN Unit 2 loading, provide the following information:

(a) Worst case expected diesel generator load profile (considering both auto and manual loads) during first 24 hours of accident occurring in one unit and shutdown of other unit and assuming single failure of one diesel generator or assuming single failure of “A” train or “B” train (two diesel generators of same train out).

Response: There are four diesel generators which supply onsite power to four 6.9kV shutdown boards 1A-A, 1B-B, 2A-A and 2B-B with one diesel generator dedicated to one shutdown board (Unit 1 and Unit 2; Train A and B). The ESF loads for Unit 1 and Unit 2 are separated on different shutdown boards. Therefore, the worst load on any diesel generator will be that of Train A or Train B of one unit only. Since there is a dedicated diesel generator for each train and one train of ESF loads is considered to be adequate to safety shutdown an accident unit, an outage of one train (two diesel generators) will not affect the ability to safely shutdown the unit.

A calculation was performed to verify adequacy of the DGs to power all the required ESF loads for an event. The calculation was performed using the same ETAP database which was used to perform APS analysis. Each diesel generator was evaluated for the worst case accident loading under LOOP and LOOP+SI Phase A or B. The calculated loading for the automatically sequenced loads and required manual loads is determined to be within the DG ratings for the analyzed DBEs.

Table below depicts the worst case loading of all scenarios and the margin available.

Maximum Steady-State Loading, 0 hrs to 2 hrs

	1A-A	1B-B	2A-A	2B-B	Rating	Minimum Margin (%)
kVA Event/Time	4847.06 SIB 1810s	4721.16 SIB 1810s	4688.42 SIB 1810s	4831.87 SIB 1810s	6050	19.9

Maximum Steady-State Loading, 2 hrs to End)

	1A-A	1B-B	2A-A	2B-B	Rating	Minimum Margin (%)
kW Event/Time	4207.45 SIB 7200s	4077.12 SIB 7200s	4079.96 SIB 7200s	4198.79 SIB 7200s	4400	4.97*
kVA Event/Time	4847.06 SIB 7200s	4721.16 SIB 7200s	4688.42 SIB 7200s	4831.87 SIB 7200s	5500	11.8

* Excludes load of 125V DC spare charger. Also all pumps are considered operating for the entire period.

Maximum Transient Loading, 0 to 180 sec

	1A-A	1B-B	2A-A	2B-B	Rating	Minimum Margin (%)
kW Event/Time	3937.65 SIA 35s	3520.30 SIA 35s	3459.14 SIA 35s	3878.36 SIA 35s	4785	17.7

Maximum Transient Loading, 180 sec to End

	1A-A	1B-B	2A-A	2B-B	Rating	Minimum Margin (%)
kW Event/Time	4736.24 SIB 184s	4498.05 SIB 184s	4481.74 SIB 184s	4755.44 SIB 184s	5073	6.2

Maximum Step Load Increase (Excitation), 0 sec to End

	1A-A	1B-B	2A-A	2B-B	Rating	Minimum Margin (%)
kVA Event/Time	4111.29 SIB 0s	4397.35 SIB 0s	3885.44 SIB 0s	3944.21 SIB 0s	8000	45.0

(b) Confirm factors such as cable losses, pump run-out conditions, power factor, off-nominal frequency and off-nominal voltage, motor efficiency have been accounted in the diesel generator load profile calculations.

Response: The ETAP DG loading analysis considers the cable losses, power factor, and efficiency as documented in the load data editors of ETAP. The off-normal frequency and voltage (for diesel loading) is not considered in the diesel generator loading analysis. TVA's position on this is that TVA has added administrative limits to the plant operating procedures for both DG voltage and speed range. These administrative limits are so tight that there would be negligible impact on the DG loading due to off-normal frequency and voltage.

8.3.1 - 7. In Section 8.3.1.2.3 of the FSAR (under the subheading "Underground Cable Installation"), it is stated that "Cables are designed to operate in wet conditions. The Class 1E cables required to operate the plant in the flooded condition are continuous or provided with a waterproof splice in a manhole. Cables have been tested at the factory by the manufacturer according to TVA specifications, which invoke Insulated Cables Engineers Association (ICEA, formerly IPCEA) standards for cables installed in wet environments." Clarify whether the WBN Unit 2 underground cables are designed for submerged or flooded conditions.

Response: Unit 2 underground cables are designed for submerged or flooded condition. All Unit 2 safety related underground cables that were routed in duct banks to the remote facilities like intake pumping station and diesel generator buildings were turned over to Unit 1 when Unit 1 was licensed in 1996 and have since been in service supporting Unit 1.

Section 8.3.2 - DC Power Systems (Onsite)

8.3.2 - 1. On page 8.3-60 of the FSAR, a description is given on load assignments with respect to divisional requirements. The staff requests additional information on assignment of loads for maintaining separation between loads of different divisions and channels as follows:

a. Provide a detailed discussion on the divisional requirements (i.e., the requirements for two and four divisions of separation).

Response: The divisional requirements for Class 1E electrical systems are delineated in a Watts Bar Design Criteria. The Reactor Protection System (RPS), Engineered Safeguard Features Actuation System (ESF), and Essential Supporting Auxiliary Systems (ESAS) and Electrical Power Systems must function to initiate shutdown of the reactor and initiate engineered safety features, if required, under the conditions produced by design basis event, occurring before, during, or after the abnormality requiring protective actions. The number of divisions (channels or trains) is determined by the number of independent sources of power required for a given function. Attachment 6 pictorially depicts this design feature for Unit 1 and Unit 2. Loads are assigned to the systems according to the load's divisional requirements. The Auxiliary Power System consists of two trains, A & B. 125VDC vital dc and 120VAC instrument power loads are assigned to the four redundant channels (I, II, III, and IV). The 125 VDC loads primarily associated with Unit 1 are assigned to Channels I and II while the loads primarily associated with Unit 2 are assigned to Channels III and IV.

b. Describe the methodology that you used for distributing the nondivisional loads among the four channels.

Response: As stated in the FSAR (page 8.3-56), non-divisional loads associated with Unit 1 are assigned to Channels I or II and Unit 2 non-divisional loads are assigned to Channels III or IV. This distribution of loads is done to minimize the impact of a loss of a battery to more than one unit.

FSAR page 8.3-64 states the following:

"A battery service test, conducted in accordance with the procedures of Section 6.6 of IEEE Standard 450-1980 or modified performance test based on Section 5.4 of IEEE 450-1995, is also used to test the batteries under conditions as close to design as practical."

a. In order to credit the modified performance test as a replacement for the service test it must completely envelope the service test. Provide the duty cycle load profile for both the service and modified performance tests (in graphic form) to show that the modified performance test completely envelopes the service test for each of the vital and diesel generator (DG) system batteries' design duty cycles (i.e., DBAs, station blackout (SBO), and Appendix R).

Response: Due to duty cycle limitation imposed by IEEE Standard 450-1995, Section 5.4, Watts Bar has not developed a modified performance test duty cycle or an implementing procedure.

b. The latest version of IEEE Standard 450 that is endorsed by the NRC is IEEE Standard 450-2002. The NRC has not endorsed IEEE Standard 450-1995. Provide the technical basis for selecting the IEEE Standard 450-1995 instead of the IEEE Standard 450-2002.

Response: IEEE Standard 450-1995 is cited in the FSAR and Technical Specifications Bases for Unit 1 relative to requirements for modified performance tests tests and is applicable to Unit 2.

c. Clarify whether the battery service test(s) verify the design duty cycles for DBAs, SBO, and Appendix R scenarios.

Response: The service test duty cycle is determined in a Unit 1/ Unit 2 vital battery sizing calculation from a review of battery load duty cycles associated with Battery I, II, III, IV & V for the SBO, LOCA/LOOP and Appendix R cases. A composite load duty cycle for the batteries was established manually by identifying the maximum positive plates required plus the maximum current required during the first minute. Design margin was included in the load values obtained from the calculations for the battery duty cycle used for testing.

The duty cycle chosen represents the worst-case duty cycle for an SBO event wherein worst-case means that battery load case controls the battery size (more positive plates) and voltage (minimum voltage). The LOCA/LOOP cases were considered for first minute only because the total duty cycle is much smaller than the SBO duty cycle. The "Appendix R" cases do not have an impact because the duty cycles are bounded by SBO.

Based on review of the load study case results, Load Study Cases for Battery I LOCA and Battery III SBO represent the worst case loading for the first minute and complete duty cycle 1 - 240 minutes respectively for batteries I, II, III, and IV & V.

8.3.2 - 2. FSAR page 8.3-66 states the following:

The 125V dc Class 1E electrical systems were designed, components fabricated, and are or will be installed meeting the requirements of the NRC 10 CFR Part 50 Appendix A General Design Criteria, IEEE Standard 308-1971, NRC Regulatory Guides 1.6 (Revision 0) and 1.32 (Revision 0), and other applicable criteria as enumerated herein.

a. Explain how the system design meets the guidance provided in RG 1.32 and IEEE Standard 308-1971 with regard to sharing DC power sources at a multi-unit nuclear power plant site.

Response: The vital batteries and battery chargers are demonstrated through analysis to have the capability to supply shared loads including those that would result from an accident in one unit and safe shutdown of the second unit. Sharing of dc power at a multi-unit power plant site is addressed in Regulatory Guide 1.81. This is addressed in Unit 2 FSAR section 8.3.2.2 under Analysis of Vital 125-volt dc system for compliance to Regulatory Guide 1.81, Position C2 and Section 8.3.2.4, Independence of Redundant DC Power Systems. The safety loads are assigned to the vital 125-V batteries so that sharing will not significantly impair their ability to perform their safety functions including in the event of an accident in one unit and orderly shutdown and cool-down of the remaining unit while considering the effects of a single failure. ESF loads are assigned to batteries I and II for Unit 1 and to III and IV for Unit 2 such that an accident in one unit will not adversely impact battery sources for ESF loads of the other unit. The TDAFW controls are supplied by battery III (normal) or battery IV (alternate) for Unit 1 and battery I (normal) or battery II (alternate) for Unit 2. The loading impact of these for both units is considered in the analysis.

The Watts Bar 125V DC power system meets the requirements of Section C.b of Regulator Guide 1.32 (Battery Charger Supply). The battery chargers have the capacity to support the steady-state operation of the connected loads required during normal operation while maintaining its battery in a fully charged state and have sufficient capacity to restore the battery from the design minimum discharged state to its fully charged state within 12 hours in Loss of Coolant Accident (LOCA) and 36 hours in Station Blackout (SBO) events while supplying applicable steady-state loads.

b. Provide an executive summary of the results of calculations used for determining the size of the inverters, chargers, batteries, and fuses (the scope of this request includes the normal and 125 Volt (V) DG battery systems). Include key inputs and assumptions and a discussion of margins in your response.

Response: Vital Battery and Battery Chargers

Analysis of the Watts Bar 125V DC vital battery system has been performed for determining the size of battery and battery chargers for the design basis conditions. The analysis developed load profiles for each battery for two unit operations under the following design basis conditions:

a) Station Blackout (SBO): loss of both offsite and onsite ac sources without accident (four hours)

b) Accident: accident with loss of offsite power and chargers plus a single failure (30 minutes, LOCA/LOOP in one unit & orderly shutdown of the second unit)

c) Normal Operation: charger loading during normal operation

d) Appendix-R: Loss of ac power to chargers and inverters; maintain reactor at hot shutdown for a period of two hours. Appendix R event loading is bounded by SBO.

This calculation establishes minimum available voltage and short circuit current at the battery boards and downstream 125V

DC control buses at switchgear and control panels to verify their adequacy for continuous operation within their voltage and current ratings. Design inputs and assumptions for this calculation are included in Attachment 9.

The analysis and methodology is in accordance with IEEE 485, "IEEE Recommended Practice for Sizing Lead Storage Batteries for Generating Stations and Substations" and IEEE 946, "IEEE Recommended Practice for Design of Safety DC Auxiliary Power Systems for Generating Stations." The analysis is performed using ETAP-DC Version 5.5.6N.

<u>Loading</u>

The capacity of the existing 125V Vital Batteries I, II, III, IV and V with 16 positive plates is adequate to support two unit operation under all design conditions and the size of the existing battery chargers is adequate for recharging the batteries while simultaneously feeding the applicable continuous loads. Both batteries and battery chargers meet the sizing requirements as stated in the Unit 2 FSAR Section 8.3.2. The results from the most recent revision (R8) of the sizing calculation are summarized as follows:

Battery Size Positive Plates (PP)

Battery No.	Available PP	Calc. PP SBO	Calc. PP LOCA+LOOP	Calc. PP APP. R
I	16	15.90	5.40	11.21
II	16	15.23	5.30	10.78
III	16	15.90	4.75	10.12
IV	16	14.97	4.87	9.59
V	16	15.90	5.40	11.21

Load shedding on vital inverters 1-I, 2-I, 1-II and 2-II is required to be completed within 30 minutes in accordance with Unit 2 FSAR Section 8.3.2.1.1, Load Time of Application, to conclude that the selected battery is adequate for the four (4) hour SBO coping duration. The list of specific loads which may be turned off to shed load is documented in the inverter loading calculation and design output drawings.

Margins

Margins for aging and minimum temperature have been applied in accordance with IEEE 485 methodology. Design margin has been identified on each battery board for accommodation of future loads. An additional margin included in the calculation is that load shedding effects are credited at 45 minutes rather than 30 minutes. The worst case loading for all batteries, considering conservative application of connected loads and design margin allowances, requires less than the rated 16 positive plates per cell. Provision is made in the calculation for evaluating and incorporating the effects of future design changes on design margins.

Voltages

Available voltage at the battery terminals and downstream control buses exceeds the required minimum voltage when they are fed from normal or alternate batteries with all battery cells are in service. The available maximum voltages at busses and devices was evaluated and determined to be acceptable. The batteries are capable of supporting a minimum terminal voltage of 114V (117.8V, Battery V) for the first minute and 105V (108.5V, Battery V) at the end of the discharge period.

Short Circuit

Available short circuit currents at battery board I, II, III, IV, V and associated downstream safety related control buses were determined and evaluated with respect to interrupting rating capacity of the protective devices. The evaluation determined that the interrupting ratings of the breakers and fuses for the existing design on the battery boards and downstream control buses are adequate.

Service Test Duty Cycle

A service test duty cycle which bounds worst-case load conditions for all batteries for operation of Units 1 and 2 is established by this calculation. Refer to Unit 2 FSAR Table 8.3-12.

Vital DC fuses

Fuses for the 125V dc system include battery board fuses for battery main feed, vital battery charger and vital inverter feeds, non-safety related load bus groups, and control fuse assemblies consisting of 250 sets of 5-amp signal fuses used to supply miscellaneous control circuits for solenoid valves and control relays. Additional fuses analyzed includes fuses for medium and low voltage switchgear control circuits, 5-amp signal fuses used for power distribution in relay racks for relay control circuits and fuses used for supplemental circuit protection or isolation in various control circuits. This analysis verifies fuses are applied within their ratings for continuous operation, have adequate dc voltage ratings and interrupting capacity, provide cable protection where required and that they are appropriately coordinated with upstream protective devices.

A supplemental analysis specifically for Unit 2 circuits not already in service for Unit 1 has been performed.

Vital Inverters

120V AC Vital Inverter loading analysis considering the effects of Unit 2 loads has been performed. The purpose of this analysis is to verify the adequacy of the 120VAC Vital Instrument Power System to supply loads powered from 120VAC Vital Instrument Power Boards 2-I, 2-II, 2-III and 2-IV.

Each power board is powered by a static inverter rated at 20kVA @ 0.8 to 1.0 power factor. The boards each contained 48 branch circuit breakers supplied from the main bus in groups of 12 through fused sub-distribution buses. The analysis performs the following listed objectives:

- Determine and document branch circuit loads and verify the adequacy of the breaker trip ratings.

- Calculate sub-distribution bus loads and verify adequacy of fuse ratings.

- Calculate overall power board steady-state load and power factor and verify capability of inverter and power board main bus to service the load.

Load Limits

Limitations are placed on the Unit 1 and Unit 2 120VAC Vital Instrument Power Boards 1-I, 1-II, 1-III, 1-IV, 2-I, 2-II, 2-III and 2-IV due to restraints imposed by battery capability for two unit operation, and these limits are tabulated below. Unit 2 FSAR Table 8.3-11 has been revised to show the following load limits for the Unit 1 and Unit 2 120VAC Vital Instrument Power Boards.

Load Limits (kVA)

Channel	1-I	1-II	1-III	1-IV
Rating	20	20	20	20
Load Limit	14	14	10.5	10.5
SBO Load Limit *	10	10	10.5	10.5

Channel	2-I	2-II	2-III	2-IV
Rating	20	20	20	20
Load Limit	14	14	10.5	10.5
SBO Load Limit *	10	10	10.5	10.5

* Shedding of non-required loads within 30 minutes of the onset of a SBO event may be necessary to reduce actual loading on distribution boards 1-I, 1-II, 2-I and 2-I. Loads which may be shed to achieve the SBO load limit are identified on load summary table with the designation “LS” adjacent to the breaker number and are listed as design output on drawings.

Branch circuit breakers are properly sized and the adequacy of the breaker trip ratings is verified for continuous full load and inrush currents.

70A sub-distribution bus fuses are properly sized to supply all bus loads.

Loading for all of the 120VAC Vital Instrument Power Boards is within the inverter ratings and allowable load limits. The Vital Inverters are properly sized to supply power for power board loads. The results of the most recent revision (R131) of the inverter loading calculation are as follows:

Inverter Ratings and Calculated Loading (KVA)

Channel	2-I	2-II	2-III	2-IV
Rating	20	20	20	20
Load Limit	14	14	10.5	10.5
Calculated Load	8.9	9.5	8.7	8.9
SBO Load Limit	10	10	10.5	10.5
Calculated SBO Load	6.1	7.1	8.7	8.9

Design margins are identified and the documentation provides a means for evaluating and incorporating the effects of future design changes.

<u>Vital AC fuses</u>

Fuses for the 120V ac system include vital inverter output fuses, distribution board load group fuses, fuses in series with circuit breakers that provide back-up penetration protection and fuses used for supplemental circuit protection or isolation in various control circuits. This analysis verifies fuses are applied within their ratings for continuous operation, have adequate voltage ratings and interrupting capacity, provide cable protection where required and that they are appropriately coordinated with upstream protective devices.

The analysis was revised to incorporate evaluation for Unit 2 circuits not previously analyzed.

<u>Diesel Generator (DG) Battery and Battery Charger System</u>

The 125V dc DG control power system is required to provide control power for the control and field flashing of the DG sets under normal conditions and a loss of all ac power. Analysis for the DG control power system has evaluated the adequacy of the size of battery and battery chargers. The analysis establishes a duty cycle for a worst case SBO scenario considering diesel circuit loads based on sequence of application for an assumed three start attempts: (1) a failed start attempt at initial diesel emergency start signal, (2) a second failed attempt at 29 minutes, and (3) a third attempt at the end of the four-hour SBO coping period. The analysis conservatively assumes that fuel oil pumps and lube oil pumps for both engines each start simultaneously such that starting currents for two pump motors are applied at the same time. No special operator actions are credited to reduce circuit load between start attempts. The calculation demonstrates that the batteries and battery chargers are adequately sized.

The diesel generator battery system configuration is shown on FSAR Figure 8.3-24.

Margins for Diesel Generator Battery Systems

The cell sizing calculation utilizes IEEE-485 methodology and considers margins for aging and minimum temperature. Design margin remains based on a required 2.75 positive plates versus the provided 3 for the C&D KCR-7 battery cells.

Battery Size Positive Plates (PP)

Battery No.	Available PP	Calc. PP SBO
1A-A	3	2.75
1B-B	3	2.75
2A-A	3	2.75
2B-B	3	2.75

The battery chargers are rated 20 amps and can carry normal load currents while recharging the battery within 8 hours for the analyzed battery duty cycle.

The calculation provides a means for evaluating and incorporating the effects of future design changes.

Fuses for Diesel Generator Battery Systems

Fuses for the 125V Diesel Generator battery system include distribution panel fuses for the battery main feed, and panel instruments. Additional fuses analyzed include fuses for engine control and lube oil pump control circuits and alarm circuit isolation fuses. The calculation verifies that the fuses are applied within their ratings for continuous operation, have adequate dc voltage ratings and interrupting capacity and provide selective coordination with other protective devices.

All of the analyzed circuits are already in service for Unit 1. The analysis was reviewed for applicability to Unit 2 and it was determined that no additional analysis was required. Design inputs and assumptions are included in Attachment 9.

8.3.2 - 3. FSAR page 8.3-67 states the following:

The normal or preferred power source to each distribution board is from the battery charger, which is supplied from either one of two 480V ac shutdown distribution boards. The battery serves as an emergency source in the event the battery charger source is lost or is inadequate for the load required. Table 8.3-12 provides maximum loading for each board for normal, loss of all ac power, and accident conditions.

After reviewing Table 8.3-12, the staff noticed that it does not include the maximum loading values for normal conditions. Provide the maximum loading values under normal conditions.

Response: The bounding value for maximum charger load other than battery charging, which would include normal operation, is 120 amps as specified in Note 2 of Table 8.3-12. The vital battery system analysis determines the loads for each battery including the normal load current to be supplied by the chargers. Based on the latest revision of the analysis, the maximum normal continuous load for any channel is less than 95 amps.

8.3.2 - 4. FSAR page 8.3-69 states the following:

"Seismic Category I(L) battery charger V is intended solely to maintain vital battery V in its fully charged state and to recharge it following its use or testing. At no time will battery charger V be used to supply vital battery system loads. The fifth battery charger does not supply dc system loads; therefore, the overvoltage and failure alarm relays do not serve any safety or protective function and consequently are not required for alarms."

a. Describe how Vital Battery V and its associated components will be protected against potential overvoltage conditions when being used as a temporary replacement for Vital Battery I, II, III, or IV.

Response: When Vital Battery V (62 cells) is used to replace Vital Batteries I, II, III, or IV (60 cells), Battery Charger V is disconnected from Vital Battery Board V and Vital Battery Board V is aligned through transfer switches and disconnects to connect Vital Battery V to the main distribution bus of the replaced battery. The normal or spare charger for the replaced channel can be used to maintain charge on Vital Battery V and supply normal loads. The typical charger alignment for this configuration is that the normal charger is isolated from the main distribution bus and used to support the disconnected, replaced battery. The spare charger is aligned to the battery board main distribution bus and output voltage verified per procedure, Standard Operating Instruction SOI-236.5 (125V DC VITAL BATTERY BOARD V) to be within the range of 137 to 140 volts, the float voltage for Battery V. This setting is within the allowed maximum voltage of 140V for the affected channel. Equalizing

Vital Battery V when connected to Vital Battery Boards I, II, III, or IV is not permitted since the equalize voltage would exceed the allowed maximum voltage for the channel.

b. Explain how the fifth battery is maintained in a fully charged state and its associated equipment is supplied power when used as a temporary replacement for Vital Battery I, II, III, or IV. In your response, include a discussion on the capability of the battery charger to recharge the battery and supply expected loads.

Response: See the response to **item 4a.**above. The battery charger sizing calculation is based on restoring the discharged ampere-hours plus carrying the normal steady-state loads. The Vital battery sizing analysis considers both the normally aligned battery and Vital Battery V for each channel and design basis condition. The results of the analysis demonstrates that the ampere hours discharged from Vital Battery V is essentially the same as Vital Batteries I, II, III and IV; therefore, the battery charger sizing calculation is applicable and valid when Battery V is aligned to the board.

8.3.2 - 5. Provide the title for Section 8.3.2.5 of the FSAR (located on page 8.3-71).

Response: The RAI was the result of a review of the red-line version of Amendment 95 to the Unit 2 FSAR. The red-line shows the deletion of header 8.3.2.5 and the first paragraph under this header. The software used to generate the markup leaves the header number (i.e., 8.3.2.5) until the changes are incorporated.

The issued version of page 8.3-66 contained in Amendment 95 to the Unit 2 FSAR properly reflects the deletion of both the header title and the associated header number (i.e., “8.3.2.5”).

8.3.2 - 6. FSAR page 8.3-72 states the following:

“The limiting conditions studies was the loss of offsite power concurrent with the failure of one battery. Table 8.3-13 shows the results of this study.”

After reviewing the FSAR, the staff could not locate this Table. Provide the Table 8.3-13 (or the results of this study) for staff review.

Response: Amendment 100 to the Unit 2 FSAR will add Table 8.3-13 to the FSAR.

8.3.2 - 7. Provide the performance characteristic curves that illustrate the capability of the Class 1E Batteries to respond to and supply the most severe loading conditions at the plant. In your response, include the performance characteristic curves such as voltage profile curves, discharge rate curves, and temperature effect curves.

Response: Load study cases for each vital battery have been performed for all design basis conditions. Battery performance curves for voltage profile and discharge rate for the most severe loading condition are included as Attachment 10. Temperature effects are applied in accordance with Table 1, Cell Size Correction Factors for Temperature, of IEEE Standard 485.

8.3.2 - 8. FSAR page 8.3-19 states the following:

"The diesel generator 125V dc battery system's chargers have the capacity to continuously supply all steady-state loads and maintain the batteries in the design maximum charged state or to fully recharge the batteries from the design minimum discharge state within an acceptable time interval, irrespective of the status of the plant during which these demands occur."

a. Define the term 'acceptable time interval'.

Response: Adequacy of the size of the Diesel Generator (DG) Battery Chargers has been analyzed. DG battery chargers are sized to supply all steady state loads and to fully recharge the batteries from design minimum discharge state to fully charged state within less than eight (8) hours. This DG battery recharge time compares favorably to the recharge time of 36 hours for 125 VDC Vital battery following an SBO and 12 hours following a LOCA. The period of eight (8) hours is therefore judged to be an "acceptable time interval."

8.3.2 - 9. FSAR page 8.3-19 states the following:

"Each of the diesel generator battery system has sufficient capacity to supply required loads for the four-hour station blackout (SBO) period."

Provide the technical basis for the 4-hour period and discuss in detail the required loads the DG battery system will be supplying during the four-hour SBO period

Response: Watts Bar SBO event analysis requires that the DG battery supply DG loads without the benefit of a charger for a 4-hour period. This includes three diesel start attempts: first attempt at t = 0 (initial emergency start), second attempt at t = 29 minutes, and the third attempt at the end of four-hour period. Because Watts Bar is a 4-hour coping plant, this is the technical basis for DG battery sizing. The following lists the DG battery duty cycle loads:

1. Diesel generator control circuit
2. Diesel fuel oil pumps
3. Diesel lube oil pumps
4. Diesel generator field flash circuit

Section 8.4 – Station Blackout

The staff review guidance on Station Blackout (SBO) (10 CFR 50.63) is given in NUREG 800, Chapter 8, Section 8.4. The staff's review of FSAR Amendments No. 95 and 97 finds that they do not contain information on Section 8.4 for addressing an SBO event in WBN Unit 2. The NRC staff issued a safety evaluation (SE) report dated March 18, 1993, (TAC Nos. M68624 and M68625) and a supplemental SE dated September 9, 1993, on WBN compliance with 10 CFR 50.63. The NRC staff believes that the original review of WBN Unit 2 compliance with an SBO was performed under TAC No. M68625. Since WBN Unit 2 is now seeking an OL 17 years after the initial review for conformance to the SBO rule, the NRC staff requests that TVA update and/or validate the original information, or provide a new submittal on how WBN Unit 2 meets the SBO rule. TVA should also update FSAR Section 8.4 to include the relevant information on SBO. The information to be submitted to the staff on SBO for WBN Unit 2 should include the following:

8.4 - 1. The specified coping duration to withstand and recover from a SBO based on the factors listed in 10 CFR 50.63 and the expected frequency of grid-related loss of offsite power in the last 20 years.

Response: Watts Bar's SBO design basis is defined in Watts Bar (WBN) Design Criteria WB-DC-40-64, "Design Basis Events Design Criteria," Section 4.41, "Loss of All AC Power (Station Blackout (SBO))." Compliance with the SBO design basis is documented in WBN calculation EPMMA041592, "Station Blackout Coping Evaluation," and referenced information. The SBO coping time, determined in accordance with NUMARC 87-00 guidelines, is four (4) hours. Areas evaluated include the following:

Condensate Inventory for Decay Heat Removal

Using the methodology described in NUMARC 87-00, Revision 1, Section 7.2.1, calculate the Condensate Storage Tank (CST) inventory required to maintain the RCS at Hot Standby without cooldown for four (4) hours and assess the adequacy of CST inventory (200,000 gallons).

Auxiliary Control Air (ACA)

Assess adequacy of ACA system supply to the TDAFW pump level control valves and SG-PORVs since these are the only SBO shutdown components which uses ACA air.

Reactor System (RCS) Inventory

Evaluate RCS inventory loss (seals, letdown) during four (4) hour SBO coping period.

Class 1E Battery Capacity

Evaluate the Class 1E 125 volt batteries capacity to provide DC power to SBO shutdown components during the coping period.

Other Battery Systems

Evaluate the EDG 125 volt DC batteries for their ability to successfully start EDGs at end of SBO to ensure restoration of AC power, assuming two (2) attempted EDG starts at the beginning of the SBO event.

Appropriate Containment Integrity

Evaluate the mechanical penetration/fluid system containment isolation valves against the exclusion criteria of NUMARC 87-00, Revision 1, Section 7.2.5. Those valves which did not meet those exclusion criteria are analyzed further. These valves are associated with penetrations X-19A and B, X-44, and X-107.

8.4 - 2. Provide a summary of the strategies and analysis for coping with SBO for the specified duration. This discussion should provide sufficient information, including baseline assumptions, on the systems and equipment required for coping with an SBO for the specified duration without ac power for the following:

a. The core and reactor system conditions and the ability to maintain adequate reactor coolant system (RCS) inventory to ensure that the core is covered and cooled. Discuss and provide information on RCS inventory taking into consideration shrinkage, leakage from pump seals, and inventory loss from letdown or other normally open lines.

Response: The success criterion for RCS inventory is that the core remains covered throughout the SBO coping period (4 hours), accounting for: 1) shrinkage; 2) letdown; 3) normal system leakage; and, 4) reactor coolant pump (RCP) seal leakage.

1) Watts Bar is a “Hot Standby” plant, so shrinkage is not significant. Even if a cool-down to 350°F occurred within the SBO coping period, the core would remain covered.

2) The letdown containment isolation valve closes on loss-of-AC power.

3) The normal system leakage is limited to 10 gpm (identified leakage) by Technical Specifications 3.4.13.

4) RCP seal leakage is assumed to be 25 gpm per RCP. This is conservative with respect to the 21 gpm provided in WCAP 10541, “Westinghouse Owner’s Group Report, ‘RCP Seal Performance Following a Loss of All AC Power.’” This value has also been assessed with respect to seal leak-off line failure as reported by Revision 2 of WCAP 10541. The seal leak-off line will not fail at WBN.

The total RCS inventory at the end of the SBO coping period is approximately 8,600 ft^3 (assuming no shrinkage) versus a reactor vessel volume of approximately 5,000 ft^3. Therefore, the core remains covered.

b. Discuss and provide information on the capacity of the condensate storage tank to ensure that there will be sufficient water inventory to remove decay heat during the specified SBO duration.

Response: Per Technical Specification 3.7.6, the CST shall have at least 200,000 gallons reserved for the Auxiliary Feedwater System. It will take approximately 75,500 gallons of CST inventory to remove decay heat (without cooldown) during the SBO coping period. In the unlikely event that plant cooldown is required, then the required inventory from the CST is 197,200 gallons.

c. Discuss and provide information on the compressed air capacity to ensure that air operated valves required for decay heat removal have sufficient reserve air and appropriate containment integrity will be maintained for the specified duration.

Response: The only Air Operated Valves (AOVs) required during the SBO coping period are the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) Level Control Valves (LCVs). These valves are normally supplied by the Auxiliary Control Air System (ACAS). This system will not be available during an SBO event. Modifications to install bottled nitrogen to supply these valves were made to the plant. The bottled nitrogen has sufficient capacity to supply the TDAFWP LCVs during the SBO coping period. The nitrogen bottles are normally installed. They are sized for five (5) LCV cycles. The Appendix R event is more limiting than the SBO, so the nitrogen bottle sizing is based on Appendix R.

d. Discuss and provide information on the adequacy of the battery capacity to support loads required for decay heat removal for the specified SBO duration and emergency diesel generator field flashing for recovering onsite power sources.

Response: The SBO design basis for Watts Bar is one unit in an SBO condition and the other unit with one operable diesel generator.

All battery sizing calculations consider dual unit operation.

1) 125 VDC Vital Power System

   As stated in Unit 2 FSAR 8.3.2.1.1, the vital 125V DC control power system batteries are designed to support an SBO event for four (4) hours. The coping duration for Watts Bar cannot be met with all the normal loads. Loads that are not required to mitigate an SBO will be removed within 30 minutes into the event to increase the discharge time of the battery. The battery capacity analysis demonstrates the capacity to provide the remaining loads for up to four (4) hours. The evaluation includes allowances for aging, design margin and temperature derating.

2) 250 VDC Battery System

   As stated in Unit 2 FSAR 8.2.1.4, the 250 VDC batteries are designed to support an SBO event for four (4) hours. The coping duration for Watts Bar cannot be met with all the normal loads. Loads that are not required to mitigate an SBO will be removed between one (1) and three (3) hours into the event to increase the discharge time of the battery. The battery capacity analysis and demonstrates the capacity to provide the remaining loads for up to four (4) hours. The evaluation includes allowances for aging, design margin and temperature derating.

3) 125 VDC Emergency Diesel Generator (EDG) Power System

As stated in Unit 2 FSAR 8.3.1.1, the EDG batteries are designed to support an SBO event for four (4) hours. The batteries are required to provide power and indication to allow starting of the EDG to recover from the event. The coping duration can be achieved with the battery and all loads connected, provided that a maximum of three (3) start sequences are attempted. The batteries will have sufficient capacity remaining to "flash the generator field" with the third and final start occurring at the end of the coping period. The battery capacity analysis demonstrates the capacity to provide the remaining loads for up to four (4) hours. The evaluation includes allowances for aging, design margin and temperature derating.

e. Discuss the integrity of electrical cabinets and provide information on the effects of the loss of ventilation to other equipment, such as the turbine driven emergency feed water pump, valves, the battery room and other equipment credited for mitigating an SBO event. Discuss and provide the information on the effects of loss of ventilation in all dominant areas of concern and on the equipment credited during an SBO event.

Response: Detailed room temperature evaluations consistent with RG 1.55 and NUMARC 87-00, Revision 1, guidelines have been performed in areas containing equipment required to cope with an SBO event. The following areas were evaluated:

1) 250 V Battery and Board Rooms
2) Control Room Complex
3) Cable Spreading Room
4) 125 V Battery and Board Rooms
5) 480 V Board Rooms
6) Pipe Chase Area
7) North and South Main Steam Valve Rooms
8) Turbine Driven Auxiliary Feedwater Pump room
9) 6.9 kV & 480 V Shutdown Board Room A

In general, and consistent with Sequoyah and Watts Bar Unit 1, electrical heat loads in these areas are assumed to be reduced by 50% of their normal values. An assumption of 50% reduction is considered to be very conservative as all AC power has been lost with only battery power remaining. In the rooms where actual heat loads have been calculated, the results have been less than 50% of normal. If the corresponding evaluation predicted excessive temperatures, more detailed evaluations of actual electrical heat loads were performed. Most HVAC systems are AC powered and are therefore lost during an SBO event. The Turbine Driven Auxiliary Feedwater Pump room is serviced by a DC powered ventilation system. Resulting room temperatures are within equipment qualification limits and support human habitability requirements (if any).

8.4 - 3. Provide information on site specific procedures and training on the following:

a. Coping with an SBO for the specified duration;

Response: The Unit 1 procedure for loss of shutdown board power is ECA-0.0 (Loss of Shutdown Power, currently Rev. 20). This procedure provides actions for responding to a loss of shutdown power. It also directs the restoration of shutdown board power based on the cause for the loss of shutdown board power. Step 6.a, Response Not Obtained (RNO), directs the operator to utilize AOI-35 (Loss of Offsite Power) or AOI-40 (Station Blackout), or AOI-43 (Loss of Shutdown Boards). If the loss of shutdown power is due to a Station Blackout, the operator will refer to AOI-040.

ECA-0.0 (Loss of Shutdown Power) is being drafted for Unit 2 and will be issued to support Unit 2 Startup.

The PURPOSE section of the Unit 1 and common Procedure AOI-40 (Station Blackout, currently Rev. 013, states the following:

“This Instruction provides guidance for restoration of shutdown AC power via the D/Gs or backfeed from the 500Kv system. Provides operator actions to reduce load on the 125V vital, and 250V station batteries for complete loss of all AC power to extend the useful life of the DC backup power system(s).”

AOI-40 also provides coping strategies for station 125 Vdc and 250 Vdc batteries.

Section 3.2 of this instruction includes restoration steps to return plant components and station service to normal configuration.

AOI-40 has been drafted for the Unit 2 procedure; it will be issued will be issued to support Unit 2 Startup.

b. Restoration of ac power following an SBO event of specified duration; and

Response: The PURPOSE section of the Unit 1 and common Procedure AOI-40 (Station Blackout, currently Rev. 013, states the following:

"This Instruction provides guidance for restoration of shutdown AC power via the D/Gs or backfeed from the 500Kv system. Provides operator actions to reduce load on the 125V vital, and 250V station batteries for complete loss of all AC power to extend the useful life of the DC backup power system(s)."

Section 3.2 of this instruction includes restoration steps to return plant components and station service to normal configuration.

AOI-40 has been drafted for the Unit 2 procedure; it will be issued to support Unit 2 Startup.

c. Preparation for severe weather conditions to reduce the likelihood and consequences of loss of offsite power and to reduce the overall risk of an SBO event.

Response: Procedure AOI-8 (Tornado Watch or Warning, currently Rev. 051) provides Operations' response to this event or conditions at the site.

The PURPOSE of AOI-8 states the following:

"This Instruction provides actions to be taken in the event a Tornado Watch or a Tornado Warning is issued."

AOI-8 also provides guidance to prevent damage to the facility from the potential effects of tornadoes at or near the site. The procedure includes actions such as anchoring cranes, verifying / establishing required damper / door / hatch positions, stabilization of fuel movements, and securing loose items outside of buildings.

AOI-8 has been drafted for the Unit 2 procedure; it will be issued to support Unit 2 Startup.

Section 9.5.3 – Lighting System

9.5.3 - 1.a. Provide a summary discussion of the typical luminance ranges for normal and emergency lighting in all areas/rooms of the plant to ensure that the functional capability of the lighting system design provides illumination level in accordance with the IESNA [Illuminating Engineering Society of North America] Lighting Handbook for Central Stations or NUREG 700. Discuss the technical basis if the design illumination levels are not in conformance with the guidelines of IESNA Lighting Handbook for Central Stations and NUREG 700.

Response: Luminance for normal and emergency lighting in all areas/rooms of the plant is delineated in TVA design Standard for Lighting Standards and Practices. This document is based on the following references:

- Illuminating Engineering Society of North America (IESNA) Lighting Handbook-1993
- USNRC NUREG 0700, “Guidelines for Control Design Reviews,” Appendix E-2

The design illumination levels are in conformance with the above references. Average illumination levels in footcandles for various plant areas/rooms are based on the above documents and are given in this standard as follows:

Area/Room	Illuminance (fc)
Normal Lighting	
Aisles, Corridors and Stairways	10 - 20
Auxiliaries, pumps, tanks, compressors	20
Battery and battery board room	20
Communication s room	40
Conference room	30-50
Equipment rooms, mechanical and miscellaneous	75-100

9.5.3 - 1.b. Section 9.5.3 of the WBN Unit 2 FSAR does not describe the illumination levels for the work areas or type of tasks in the Main Control Room (MCR), safety-related panels in the MCR and remote shutdown consoles. Provide a description of the illumination levels for normal lighting in these areas. Discuss the technical basis if the design illumination levels do not conform to the guidelines of IESNA Lighting Handbook for Central Stations or NUREG 700.

Response: The Watts Bar Units 1 and 2 MCR and Auxiliary Control Room (ACR) lighting system was modified in the 1989 to 1991 time frame to comply with the requirements of the following documents:

- Illuminating Engineering Society of North America (IESNA) Lighting Handbook-1981 application volume and 1981 reference volume.
- USNRC NUREG 0700, "Human Factors Engineering," Sections 6.1.5.3 and 6.1.5.4 and Appendix E-2.
- TVA Design Standard DS-E17.1.1, Rev. 2, "Lighting Design Standards and Practices."

Acceptance Criteria (minimum average illumination levels in footcandles) for the MCR and the ACR were based on the above documents.

MCR AND ACR	Illuminance (fc)	
Normal Lighting	Minimum	Maximum
Vertical Face of Switchboard (66 inches above the floor)	20	50
Benchboard (Horizontal Level)	20	50
Rear of the switchboard (vertical 60 inches above floor)	10	--
Unit Operators' Desk	50	100

After the modifications were implemented, a survey of both the MCR and the ACR was conducted to determine the actual illumination levels and to ascertain that the acceptance criteria was met. The results of this survey are documented in Watts Bar analysis for Main and Auxiliary Control Rooms; the analysis concludes that the illumination levels meet the acceptance criteria.

9.5.3 - 1.c. Discuss if the emergency lighting in the MCR, safety-related panels in the MCR and remote shutdown consoles provides illumination levels in these areas equal to greater than those recommended by the IESNA Lighting Handbook for Central Stations or NUREG 700 for at least 8 hours.

Response: Emergency and Standby lighting in the MCR and remote shutdown consoles meets the requirements of IESNA Lighting Handbook for Central Stations or NUREG 700 and is designed to provide the following illumination levels:

Standby Lighting: All Tasks, Each Train - 10 fc

Emergency Lighting: 3 fc

<u>Acceptance Criteria for the task area luminance ratio:</u>

Area	Luminance Ratio
Task area versus adjacent darker surroundings	3:1
Task area versus adjacent lighter surroundings	1:3

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

RAIs for FSAR SECTIONS 3.9.1, 3.9.2, 3.9.3, and 5.5.1 [taken from NRC letter dated 07/02/2010 (ADAMS Accession No. ML101530474)]

EMCB 3.9-1 The NRC staff noted a number of instances in the review of Sections 3.9.1, 3.9.2, 3.9.3 and their corresponding tables and figures of Amendment No. 97 to the 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) On page 3.9-18 of Reference 1, continuing to page 3.9-19, the first two paragraphs of Section 3.9.2.5.6, "Results and Acceptance Criteria," are duplicates of the first two paragraphs of the following section (3.9.2.5.7), also titled "Results and Acceptance Criteria."

Response: Amendment 98 to the Unit 2 FSAR corrected the noted discrepancy by deleting the repeated information. Since this was an editorial change, the amendment level remains the same.

2) On page 3.9-36 of Reference 1, superfluous spaces exist between the word "Table" and "3.9-17."

Response: The page of concern was page 3.9-30 in the issued version of Amendment 97 to the Unit 2 FSAR. Amendment 98 corrected the noted discrepancy by deleting superfluous spaces. Since this was an editorial change, the amendment level remains the same.

3) On page 3.9-44 of Reference 1, the primary membrane plus primary bending stress limit should be "1.1 S" versus the current "1.1.S."

Response: The page of concern was page 3.9-37 in the issued version of Amendment 97 to the Unit 2 FSAR. The second line on the version of page 3.9-37 in the issued version of Amendment 98 still contains "1.1.1.S." Amendment 100 to the Unit 2 FSAR will replace "1.1.1.S" with "1.1 S."

4) On page 3.9-63 of Reference 1, the title of Table 3.9-5 should be revised to state that the limits are "Maximum Deflections" versus the current wording of "Maximum Defections."

Response: Amendment 98 to the Unit 2 FSAR corrected the title of Table 3.9-5 to read "Maximum Deflections" instead of "Maximum Defections." Since this was an editorial change, the amendment level remains the same.

5) On page 3.9-63 of Reference 1, Note 1 references Westinghouse Commercial Atomic Power (WCAP)-5890 with a corresponding superscript of number 21, indicating that this refers to Reference 21. Page 3.9-58 of Reference 1 indicates that this WCAP report is Reference 22, not Reference 21. If this is not erroneous, please provide additional justification in conjunction with RAI 3.9.2-3 below.

Response: The NRC's comment was based on a review of the red-lined version for Amendment 97 to the Unit 2 FSAR. The noted apparent discrepancy in the reference number is because Reference (13) is marked for deletion on page 3.9-57.

The issued version of Amendment 97 to the Unit 2 FSAR correctly shows Reference (21) on page 3.9-50 as being Westinghouse Commercial Atomic Power (WCAP)-5890 which agrees with Note 1 [refers to Reference (21)] which is on page 3.9-55.

6) On page 3.9-77 of Reference 1, the third note corresponding to Table 3.9-16 should be revised to correct the misspelling of "Non-pressure" and "other justifiable" versus the current wording of "Non-pressur" and "othe justifiable."

Response: The page of concern was page 3.9-69 in the issued version of Amendment 97 to the Unit 2 FSAR. Amendment 98 (page 3.9-69) replaced "Non-pressur" with "Non-pressure."

The third line of Note 3 still contains "othe" instead of "other." Amendment 100 to the Unit 2 FSAR will replace "othe" with "other."

EMCB 3.9.1-1 In Supplemental Safety Evaluation Report (SSER) 6 (Reference 3), the NRC staff noted that the licensee's piping evaluation for a postulated main feedwater header rupture transient, which results in a water hammer event due to a rapid check valve closure, included an assumption that certain feedwater piping system supports failed when the loads exceeded their calculated capacities; this was listed as an open item in SSER 6 (tracked as Outstanding Issue 20(a)). In SSER 13 (Reference 6), the staff noted that the analyses performed, which postulated pipe support failures, was acceptable based on the difficulty involved with making subsequent pipe support modifications and the low probabilistic nature involved with the water hammer transient. Additionally, as part of the closure of this open item, SSER 13 also included a copy of a report performed by Brookhaven National Laboratories (BNL) regarding this issue. BNL was contracted by the NRC to evaluate the licensee's piping analyses performed to demonstrate compliance with the criteria of Appendix F of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code. BNL concluded that the licensee's piping analyses performed for the feedwater loops inside containment were sufficient and demonstrated that the piping system would maintain its structural integrity when subjected to the dynamic loading associated with the water hammer event.

Please describe the applicability of the conclusions made by the NRC staff and the contractor (BNL) regarding the piping analyses described above as they relate to the current WBN Unit 2 refurbishment efforts. Please indicate whether the same issues exist with the inability to modify certain piping supports within containment and whether the piping analyses for the WBN Unit 2 feedwater loops are the same as those analyses performed in support of WBN Unit 1. If these analyses are dissimilar, please summarize and provide justification for any portions of the analyses that are not exactly the same and whether the results of these dissimilar analyses demonstrate that the feedwater piping loops meet the acceptance criteria of the code of record for this piping system.

Response: Analysis methodology, piping geometry and pipe support locations including pipe whip restraint locations are similar to Unit 1. Also, Unit 1 pipe support designs and stiffnesses were used as input into the Unit 2 pipe support designs.

Each Unit 2 Main Feedwater Loop has a separate pipe stress calculation. As-designed Whip Restraint cold gaps and Steam Generator nozzle displacements were used in the analysis as design input. Six snubbers (from three of the four loops) that exist in the Unit 1 analysis are being deleted in Unit 2; however, these snubbers were assumed to fail in the Unit 1 Check Valve Slam analysis. The Unit 2 analysis has accounted for the removal of these snubbers.

EDCR 52430 (Modification of pipe supports on Main Feedwater System -003) has been issued to perform modifications on pipe supports as required to meet the acceptance criteria of the code of record and attain similarity with Unit 1 pipe support designs.

EMCB 3.9.2-1 In Section 3.9.2.3 of Reference 1, it is indicated that Sequoyah Nuclear Plant Unit 1 and Trojan Nuclear Power Plant (Trojan) "…have been instrumented to provide prototype data applicable to Watts Bar" for the purposes of evaluating the flow induced oscillatory pressure effects on the reactor vessel internals. Additionally, it is concluded, based on scale model test results and "…preliminary results from Trojan…," that plants with neutron shielding pads exhibit less core barrel vibration than plants with thermal shields. Based on the fact that Trojan ceased operations in the year 1992, please discuss the applicability of the statements above, which are currently included in Reference 1. If these data was captured during Trojan's operational state, please describe how this operating experience has been applied to the design or operational characteristics of any of the reactor vessel internals. Additionally, please indicate whether additional results, other than the "preliminary results" mentioned in Reference 1, were utilized to provide additional information regarding the comparison between plants with neutron shielding pads and plants with thermal shields as they relate to core barrel excitation.

Response: The testing on Trojan was done initially and was completed without the need to follow plant operation through the design life of the plant. Since no operational data was collected, none was applied to the Watts Bar internals design. Therefore, Trojan ceasing operations in the year 1992 has no affect on the results obtained from Trojan, and there are no additional test results required from Trojan's operational state through 1992. Also, there are no additional results, other than the "preliminary results" mentioned in Reference 1, that were utilized to provide additional information regarding the comparison between plants with neutron shielding pads and plants with thermal shields as they relate to core barrel excitation.

EMCB 3.9.2-2 The analyses methods described in Section 3.9.2.5 of Reference 1, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions," were approved for use by a previous license amendment request submitted for WBN Unit 1. These methods incorporate the use of the MULTIFLEX, LATFORCE, FORCE-2 and WECAN computer codes to model the complex, non-linear thermal-hydraulic loadings induced on the reactor vessel internals under upset loading conditions.

a. Please confirm that the inputs used to analyze these conditions for WBN Unit 2 are the same inputs as those used to analyze the loadings induced on the WBN Unit 1 reactor vessel internals.

Response: The inputs used to analyze these conditions for Watts Bar Unit 2 are the same inputs as those used to analyze the loadings induced on the Watts Bar Unit 1 reactor vessel internals.

b. If any variances exist between the WBN Unit 1 and WBN Unit 2 inputs for these codes, including primary and secondary loadings, flow parameters, mass models, finite element formulations, or other input parameters, provide justification for the variation and its effects on the ability of the WBN Unit 2 reactor vessel internals to meet the acceptance criteria provided in Table 3.9-5.

Response: No variations exist between the Watts Bar Unit 1 and Watts Bar Unit 2 inputs for these codes, including primary and secondary loadings, flow parameters, mass models, finite element formulations, or other input parameters.

c. Additionally, please clarify whether the references to “Watts Bar Unit 1” on pages 3.9-15, 3.9-19, and 3.9-20 (2) are correctly referring to WBN Unit 1 for purposes of comparing analyses or whether these instances are incorrect (i.e., these references should state WBN Unit 2 and not WBN Unit 1).

Response: The references to Watts Bar Unit 1 are for purposes of Watts Bar Unit 1 analyses which are also applicable to Watts Bar Unit 2.

EMCB 3.9.2-3 Table 3.9-5 of Reference 1, “Maximum Def[l]lections Under Design Basis Event (in),” provides the maximum allowable and no loss-of-function limits for the reactor vessel internals under design basis loading conditions. Note 1 to Table 3.9-5 indicates that WCAP-5890 provides limiting criteria for internals deflection based on stress levels induced in the internals structures.

a. Please discuss whether the acceptance criteria provided in Table 3.9-5 are based on WCAP-5890. If these criteria are based on this WCAP report, please provide the bases for the regulatory acceptance of this report.

Response: Amendment 100 to the Unit 2 FSAR will replace the values in Unit 2 FSAR Table 3.9-5 with the values from Unit 1 FSAR Table 3.9-5, as required.

In replacing this table, it was observed that the no-loss-of-function limit for the upper package axial deflection is 1.5 inches. This value should be changed to 0.15.

As noted in Note 1. of Table 3.9-5, “The allowable limit deflection … correspond to stress levels for internals structures well below the limiting criteria given by the collapse curves in WCAP-5890 …”.

This WCAP was developed to document the basis for ultimate strength criteria to ensure no loss of function. Westinghouse has no record that WCAP-5890 has been formally submitted to the NRC for review. However, WCAP-5890 is referenced in the Unit 1 FSAR which was reviewed and approved by the NRC. The regulatory acceptance for Table 3.9-5 from Unit 1 is applicable to Unit 2.

b. If these criteria are based on a methodology other than the WCAP report, please provide additional information regarding the development of these deflection limits and the bases for the regulatory acceptance of this alternate methodology.

Response: As noted in the response to **RAI EMCB 3.9.2-3 – a.**, the values in Unit 2 FSAR Table 3.9-5 will be replaced with the values from Unit 1 FSAR Table 3.9-5, as required. As described in the response to **RAI EMCB 3.9.2-3 – a.**, the methodology is based on the WCAP as a strength based limit and incorporates functional considerations which may be more limiting. This approach is the same as used and accepted for Unit 1. Since Amendment 100 to Unit 2 FSAR Table 3.9-5 will make its values the same as Unit 1 FSAR Table 3.9-5, there is no additional development information to be provided regarding Unit 2.

EMCB 3.9.2-4

a. Please provide justification for the variance between the WBN Unit 1 and WBN Unit 2 allowable and no loss-of-function deflection limits as this variance relates to the upper barrel expansion and compression limits and the no loss-of-function limit for the upper package axial deflection.

This justification should include information regarding whether there are variations in the analyses methodologies for determining the WBN Units 1 and 2 reactor vessel internals faulted loads (as requested in EMCB 3.9.2-2).

Response: As noted in the response to **RAI EMCB 3.9.2-3 – a.**, the values in Unit 2 FSAR Table 3.9-5 will be replaced with the values from Unit 1 FSAR Table 3.9-5, as required.

b. Additionally, this justification should indicate whether there are variations in the acceptance criteria for the WBN Units 1 and 2 deflection limits.

Response: As noted in the response to **RAI EMCB 3.9.2-3 – a.**, the values in Unit 2 FSAR Table 3.9-5 will be replaced with the values from Unit 1 FSAR Table 3.9-5, as required, and therefore, there is no change to the acceptance criteria.

EMCB 3.9.3-1 In SSER 4 (Reference 2), the NRC staff noted that a sampling program was initiated by TVA to determine whether the compressive stresses imposed on short column pipe supports exceeded the buckling criteria margin established by the NRC. The NRC staff accepted the sampling program and determined that TVA had adequately addressed the NRC design criteria for Class 2 and 3 pipe supports; this resolved Outstanding Issue 2.

a. Please confirm the applicability of the sampling program discussed in Reference 3 as it relates to Class 2 and 3 pipe supports at WBN Unit 2. (Note: Reference 3 in the transmittal of EMCB RAI 3.9.3-1 is SSER 6 which does not discuss a sampling program. It was discussed with Licensing and concluded that the sampling program referred to is the sampling program discussed in SSER 4).

Response: SSER 4 references TVA's letter to the NRC dated May 14, 1984, and a review of the sampling program described in that letter as the basis for concluding that Class 2 and 3 pipe supports at Watts Bar comply with the applicable NRC design criteria. The May 14, 1984, response was not defined as being unit specific and was written in response to a request from the NRC (letter dated April 17, 1984) to provide information for both Watts Bar Unit 1 and Unit 2. A list of the pipe supports included in the sample program described in TVA's May 14, 1984, letter was attached to a TVA letter to the NRC dated November 10, 1982. This listing of supports included several Unit 2 supports.

Watts Bar Design Criteria WB-DC-40-31.9, *Criteria for Design of Piping Supports and Supplemental Steel in Category I Structures*, Section 3.8 (Appendix "B" Table B-2) describes the allowable stress limits used in the design of pipe supports. This section of the design criteria incorporates the requirements of the TVA letter to the NRC dated May 14, 1984, as indicated by source note 14 on page 50 of WB-DC-40-31.9, R21. Design Criteria WB-DC-40-31.9 is used for the design of pipe supports for both Unit 1 and Unit 2 at Watts Bar.

Therefore, the sampling program and the information provided in TVA's letter to the NRC dated May 14, 1984, which supports the conclusion provided in SSER 4 is used to support the design and qualification of Class 2 and 3 pipe supports in both Watts Bar Unit 1 and Unit 2.

b. If this sampling program was not used in support of the WBN Unit 2 refurbishment effort, please discuss the current criteria used for demonstrating that these pipe supports maintain sufficient margin against critical buckling of short column pipe supports.

Response: The response to **RAI EMCB 3.9.3-1a.** notes that the program was used. Thus, this question is not applicable.

EMCB 3.9.3-2 In SSER 6 (Reference 3), the NRC staff noted its concerns regarding the licensee's use of earthquake experience data to seismically qualify Category I(L) piping and identified this concern as Outstanding Issue 19(h). In SSER 8 (Reference 5), the NRC staff noted that the licensee had developed a screening criteria to identify items in Category I(L) piping systems that may require further evaluation based on this earthquake experience data. Additionally, the licensee indicated that bounding stress cases would be performed to demonstrate the conservatism of these screening criteria. The NRC staff found this screening criteria adequate for demonstrating the seismic ruggedness of Category I(L) piping.

a. Please confirm that this screening has been performed for the WBN Unit 2 refurbishment efforts.

Response: The I(L) piping seismic evaluation program is currently being performed. The screening criterion used for the Unit 2 completion is WB-DC-20-32 (Integrated Interaction Program Screening and Acceptance Criteria) which is the same screening criteria that was used for the Unit 1 and common.

b. If this screening method was not utilized in the seismic qualification of the WBN Unit 2 Category I(L) piping, please discuss the criteria that has been used to seismically qualify these piping systems and discuss the regulatory acceptance bases for this alternate criteria.

Response: As stated in the response to **portion a.** of this request, there has been no change in the screening method.

EMCB 3.9.3-3 In addition to the screening methods used for Category I(L) piping systems described in RAI 3.9.3-2, SSER 8 also describes TVA's criteria used for the evaluation of Category I(L) piping supports. The NRC staff noted in SSER 8 that TVA had indicated it would utilize a factor of safety of three in their evaluation of concrete expansion anchor bolts for these pipe supports. The NRC staff accepted the use of this safety factor value for validating the existing design of concrete expansion anchors used in this piping system based on TVA's implementation of recommendations including additional concrete inspection, anchor spacing, and concrete edge distance in conjunction with the existing anchor bolts. The NRC staff also noted in SSER 8 that for future Category I(L) piping, the required safety factors for these piping systems found in the former Office of Inspection and Enforcement (IE) Bulletin 79-02, should be utilized.

a. Please discuss whether the existing, applicable Category I(L) piping supports at WBN Unit 2 have been evaluated in the manner described in SSER 8.

Response: The category I(L) piping supports for Unit 2 have been evaluated in the same manner as the category I(L) piping supports for Unit 1. That is, the existing supports were evaluated with the use of a factor of safety of three, which included additional concrete inspection, anchor spacing, and concrete edge distance attributes. The design of new category I(L) pipe supports is performed in accordance with Watts Bar design criteria WB-DC-20-32 (Integrated Interaction Program Screening and Acceptance Criteria) which is consistent with the required safety factors found in Bulletin 79-02.

b. If these supports have been evaluated in a dissimilar manner, please provide justification for the departure from the methods described in Reference 4.

Response: As stated in the response to **portion a.** of this request, there has been no change in the manner of evaluation.

EMCB 5.5.1-1 – a. Please discuss whether TVA has committed to perform an augmented inservice inspection of the reactor coolant pump (RCP) flywheel.

Response: TVA to NRC letter dated March 4, 2009 (ADAMS Accession No. ML090700378) submitted Development Revision A of Unit 2's Technical Specifications (TS) and Technical Requirements Manual.

TS 5.7.2.10 (Reactor Coolant Pump Flywheel Inspection Program) states, "This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulation Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975."

TRM Surveillance requirement TSR 3.4.5.1 states, "Inspect each reactor coolant pump flywheel according to the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975." Its frequency is "According to the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1."

EMCB 5.5.1-1 – b. If no commitment has been made, please provide justification that the potential for excessive vibration on the reactor coolant pump flywheels will be adequately addressed to minimize the possibility of RCP shaft or flywheel failure.

Response: The response to **RAI EMCB 5.5.1-1 – a.** documents the commitment; no response is needed to this RAI.

References

1) Letter from M. D. Jesse, Exelon Generation Company, LLC, to NRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) – Unit 2 – Final Safety Analysis Report (FSAR), Amendment 97," dated January 11, 2010. (ADAMS Accession Nos.: ML100191421 (letter), ML100191684 (Section 3.8.5-3.11))

2) NUREG-0847, Supplement 4, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated March 31, 1985. (ADAMS Accession No.: ML072060524)

3) NUREG-0847, Supplement 6, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated April 30, 1991. (ADAMS Accession No.: ML072060464)

4) NUREG-0847, Supplement 7, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated September 30, 1991. (ADAMS Accession No.: ML072060471)

5) NUREG-0847, Supplement 8, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated January 31, 1992. (ADAMS Accession No.: ML072060478)

6) NUREG-0847, Supplement 13, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," dated April 30 , 1994. (ADAMS Accession No.: ML072060484)

ENCLOSURE 1

Response to Preliminary RAIS and 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)]

Plant Systems (SBPB)

SBPB 3.6-01 Section 3.6A.2.2.2 "Blowdown Thrust Loads" contains the following equation:

$V_E = [2g_c(P_0 - P_A)/\rho_E]^{1/2}$

The equivalent equation in the WBN Unit 1 FSAR, Amendment No. 7 is:

$V_E = [2g_c(P_0 - P_A)]^{1/2}/\rho_E$ (see page 3.6A-17)

TVA is requested to clarify the differences in the two equations.

Response: The Unit 2 FSAR is correct. Amendment 8 corrected the Unit 1 UFSAR such that it now agrees with the Unit 2 FSAR.

SBPB 5.2.5-01 Previously there existed an intersystem leakage path "upper head injection system (UHI)." This system is no longer described in the FSAR. TVA is requested to confirm this system is no longer included in the WBN plant and provide the basis for deleting the system.

Response: *Deletion of UHI*

The UHI system was never utilized at Watts Bar. Both the Unit 1 and Unit 2 UHI Systems have been removed. “Watts Bar Nuclear Plant (WBN) – Assessment Report NA-WB-94-0020” was a determination of whether the Upper Head Injection Deletion (UHID) – Capital Program (CP) was adequately implemented. Attachments 1, 2 and 3 of the report list the DCNs, WPs and WOs that were executed during the removal of the upper head injection system. In SSER 7, the staff reviewed the request for the design change and found it acceptable to delete the UHI system from both units.

Licensing Basis for the Deletion of UHI

The UHI was eliminated to increase operational flexibility. The system benefits were overshadowed by frequent operational problems such as:

- Rupture or leakage of membrane in the gas crossover line separating the water and nitrogen accumulators.
- Level switch/transmitter problems for volume delivery within tolerances (specifically accuracy of Barton level switches, installation error of sensing lines and calibration procedures).
- Violation of chemistry requirements (for water accumulator nitrogen entrainment and boron concentrations).
- Violation of system gas pressure requirements.

The approach for justifying removal of UHI, however, relied on computer code technology to show that the acceptance criteria specified in 10 CFR 50.46 for the LOCA analysis was met.

Westinghouse performed the small-break LOCA analysis using NRC-approved methods. The NOTRUMP code (Westinghouse Topical Reports WCAP-10080 and WCAP-10081) was used for the calculation of transient depressurization of the reactor system, core power, water-steam mixture height and steam flow past the uncovered portion of the core. The LOCTA code (Westinghouse Topical Report WCAP-8305) was used for the peak cladding temperature analysis. The staff concluded that the Watts Bar small-break LOCA analysis results were within the acceptance criteria specified in 10 CFR 50.46.

Westinghouse also performed a large-break LOCA analysis supporting its request for removal for the UHI system. In the analysis, only the double-ended, cold-leg, guillotine (DECLG) breaks were analyzed because they resulted in the highest peak cladding temperatures. The analysis was performed using a modified revision of the 1981 Westinghouse ECCS evaluation model (WCAP-9220-P-A, Rev. 1). This evaluation model used the revised PAD fuel thermal safety model (WCAP-8720) for calculating the initial fuel conditions; the SATAN-VI code (WCAP-8302) for the thermal hydraulic calculation during the blowdown period; the transient WREFLOOD (WCAP-8170) and BASH (WCAP-10266 and addendum) codes for calculating the refill and reflood transient periods; the LOCBART code (WCAP-8305) for calculating the peak cladding temperature; and the LOTIC code (WCAP-8355) for calculating the ice condenser containment pressure transient. The staff found that the results showed that peak cladding temperature, metal-water reaction and cladding oxidation were within the acceptance criteria specified in 10 CFR 50.46 for LOCA analysis.

WBT-D-1460, “Final Small Break LOCA Summary Report,” January 22, 2010 and WCAP-17093-P, Revision 0, “Best Estimate Analysis of the Large-Break Loss-of-Coolant Accident for Watts Bar Unit 2 Nuclear Power Plant using the ASTRUM Methodology,” December 2009, are the current analyses generated for Unit 2. These analyses both show considerable margin to 10 CFR 50.46 peak clad temperature limits without the UHI system.

SBPB 9.2.6-1 In the Safety Evaluation Report (SER) related to the operation of WBN Units 1 and 2, NUREG-0847, Supplement 12, dated October 1993, the NRC staff wrote in Section 9.2.6, Condensate Storage Facilities:

> In Section 9.2.6 of the SER, the NRC staff indicated that the two condensate storage tanks reserved 200,000 gallons of condensate for each unit's auxiliary feedwater (AFW) system. <u>In FSAR Amendment No. 72, TVA revised this reserved amount to 210,000 gallons.</u> The basis for the storage capacity is not affected and this correction is made for clarification purposes only. This does not change any of the NRC staffs conclusions reached in the SER or supplements related to the condensate storage facilities or the AFW system. The NRC staffs effort was tracked by TAC M85037 and M85038.

In the proposed FSAR for WBN Unit 2, Section 9.2.6.2 System Description, TVA proposal states:

> The condensate facility, shown in Figure 10.4-7, consists of one condensate transfer pump and two condensate storage tanks connected in parallel (one tank for each unit) and associated piping, controls, and instrumentation. The tanks are located in the plant yard adjacent to the east wall of the Turbine Building. The auxiliary feedwater pumps take suction directly from the condensate storage tanks to supply treated water for cooldown of the reactor coolant system. <u>A minimum of 200,000 gallons in each tank is reserved for the auxiliary feedwater system</u>. This quantity is assured by means of standpipes through which other systems are supplied.

The NRC staff requests TVA to justify why the change to 210,000 gallons was not incorporated.

Response: Preliminary RAI 9.2.6 provided by the NRC in an e-mail of 04/23/2010 asked a similar question. TVA provided the following answer on page E1-17 of a TVA letter dated 06/03/2010 (ADAMS Accession No. ML101600477):

> "Amendment 89 revised the value from “210,000 gallons” to “200,000 gallons.” At that time, the FSAR was for both Unit 1 and Unit 2. Thus, the revision applied to both units.
>
> The amendment resulted from FSAR change number 0889. The reason for the change was to correct the condensate storage tank minimum reserve volume for auxiliary feedwater use, based on Calculation HCG-LCS-043085, Rev. 4."

SBPB 9.3.1-1 TVA provided a document titled, "FSAR Cross Referenced to SER sorted by SER, then by FSAR." In this document under the line item SER Section 9.3.1, "Compressed Air System," the scope identified new essential air compressors were installed. The compressed air system is a shared system between WBN Units 1 and 2. During a review of the proposed FSAR for WBN Unit 2, the NRC staff did not detect any changes.

The NRC staff requests TVA to explain whether there were any changes needed to be made to the proposed FSAR for WBN Units 1 and 2, based upon the installation of new essential air compressors.

Response: Evaluations of dual unit essential air system (Auxiliary Compressed Air System (ACAS)), demands have determined that the currently installed ACAS compressors have sufficient capacity to support dual unit operation. There are no plans to either replace the existing ACAS air compressors or to add additional compressors.

SBPB 10.3.0 Main Steam System TVA provided a document titled, "FSAR Cross Referenced to SER sorted by SER, then by FSAR." In this document under SER Section 10.3.0, "Main Steam Supply System," TVA identifies that this section includes a review of the following FSAR sections:

- 10.3 MAIN STEAM SUPPLY SYSTEM
- 10.3.0 Main Steam Supply System 10.3.1 Design Bases
- 10.3.0 Main Steam Supply System 10.3.4 Inspection and Testing Requirements
- 10.3.0 Main Steam Supply System 10.4.11 Steam Generator Wet Layup System

During a review of the FSAR, the NRC staff noted that Section 10.4.11, "Steam Generator Wet Layup System," was not included.

The NRC staff requests TVA to justify the omission of the FSAR Section 10.4.11, to include disposition of safety-related components that were a part of this system (e.g., containment isolation valves, piping and components).

Response: The Steam Generator Wet Layup System (SGWLS) is no longer used on Watts Bar Unit 1, and it will not be used on Unit 2. The system was designed to help protect the steam generator (SG) internals from corrosion during periods of cold shutdown. This protection was previously provided by the thorough mixing of the ammonium hydroxide and hydrazine layup solutions in each SG by utilizing the system. The previous “alternate” method of corrosion control was accomplished by injecting the chemicals into the SGs during cold shutdown; sampling using existing SG sample lines; and performing the mixing of the solutions by bubbling nitrogen (N_2) through the bottom of the SGs. An engineering analysis was performed which determined that N_2 sparging for chemical mixing in the SGs was sufficient for the wet layup process. The alternate method was adopted as the primary and only means for SG wet layup and a Unit 1 design change (DCN 51724) was performed to delete the SGWLS since it was no longer required.

That wet layup method will also be utilized on Unit 2 and a design change is being performed on Unit 2 (DCN 53864 and EDCR 54263). All Unit 2 associated piping, components and valves will be removed, modified or abandoned in place. The associated Unit 2 containment isolation valves will be removed and the piping capped. The discussion of the SGWLS was previously removed from the Unit 1 UFSAR and has been removed from the Unit 2 FSAR since the system will not be utilized. Neither RG 1.70 nor NUREG-0847 requires a discussion of the system. The abandonment of the SGWLS on Unit 1 was not associated with the replacement of the SGs.

SBPB 10.4.7 Condensate and Feedwater Systems

TVA provided a document titled, "FSAR Cross Referenced to SER sorted by SER, then by FSAR." In this document under SER Section 10.4.7, "Condensate and Feedwater System," TVA identifies that this section includes a review of the following FSAR sections:

- FSAR 5.5.9 Main Steam Line and Feedwater Piping
- FSAR 10.4.7 Condensate and Feedwater Systems
- FSAR 10.4.10 Heater Drains and Vents

During a review of the FSAR, the NRC staff noted that Section 10.4.10, "Heater Drains and Vents," shows up in the table of contents, but the text section is not included.

The NRC staff requests TVA to justify the omission of the FSAR Section 10.4.10, to include disposition of any safety-related components that were a part of this system.

Response: Heater Drains and Vents (HDV) are designed to remove all condensate from the feedwater heaters, moisture separators, 1st stage (low pressure) reheaters, 2nd stage (high pressure) reheaters, main feed pump turbine condensers and gland steam condensers. These drains and vents are not required for the safe shutdown of the plant or to mitigate the consequences of an accident.

During the re-constitution of the Unit 2 FSAR, one of the underlying tenets was to have the Unit 2 FSAR correlate as closely as possible to the Unit 1 UFSAR. Section 10.4.10 was removed from the Unit 1 UFSAR by Amendment 1; documented in Change Package 1569 and the Safety Evaluation for

EDC E-50038-A.

RG 1.70, Rev. 3, "Standard Format and Content of Safety Analysis for Nuclear Power Plants (LWR Edition)" and NUREG-0847 (series) "Safety Evaluation Reports Related to Watts Bar Nuclear Plant" do not require a discussion about HDVs.

Amendment 94 to the Unit 2 FSAR removed 10.4.10 from the Table of Contents.

Component Performance And Testing (CPTB)

CPTB Inservice Testing – 1.

Title 10 of the *Code* of *Federal Regulations,* Section 50.55a (10 CFR 50.55a), requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves be performed in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) and applicable addenda.

Paragraph 10 CFR 50.55a(f)(4)(i) requires:

"Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license under this part, or 12 months before the date scheduled for initial loading fuel under a combined license under part 52 of this chapter (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192, that is incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section."

In Amendment No. 97 to the Watts Bar Unit 2 FSAR, TVA states that IST of ASME Code Class 1, 2, and 3 pumps and valves will be conducted to the extent practical in accordance with 2001 Edition of ASME OM Code with Addenda through 2003. Justify how 10 CFR 50.55a(f)(4)(i) is met.

Response: This portion of Amendment 97 to the Unit 2 FSAR was / is in error. The Code of Record (COR) listed in the affected section is the COR for Unit 1. The COR for the initial Unit 2 Inservice Test Ten-Year Interval will be in accordance with 10 CFR 50.55a(f)(4)(i). Based on the current Unit 2 schedule and the state of current rulemaking, TVA anticipates that the Unit 2 COR will be the 2004 Edition through 2006 addenda of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). However, since the required COR is based on an unknown date for the Unit 2 operating license, TVA cannot be certain of the actual COR at this time.

Accordingly, Amendment 100 to the Unit 2 FSAR will revise the sentence regarding the COR to read as follows until the date for an operating license becomes more certain: "Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be conducted to the extent practical in accordance with the latest edition and addenda of the ASME OM Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date of issuance of the operating license for Unit 2, as required by 10 CFR 50.55a(f)."

Once the date for issuance of an operating license becomes more certain, TVA will update this sentence in accordance with 10 CFR 50.55a(f)(4)(i).

CPTB Inservice Testing – 2. TVA also indicates in Amendment No. 97 that exceptions to the OM Code requirements are noted in the IST program submittal made to NRC. Exceptions to the Code requirements are allowed by NRC regulations, but they must be identified in the IST program specifically for WBN Unit 2 along with proposed alternatives and relief requests. In proposing alternatives or requesting relief, TVA must demonstrate that: (1) the alternatives will provide an acceptable level of quality and safety, (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or (3) conformance would be impractical for its facility. The regulations in 10 CFR 50.55a authorize the Commission to approve alternatives and to grant relief from OM Code requirements upon making the necessary findings. NRC guidance contained in Generic Letter (GL) 89-04, "A Guidance on Developing Acceptable Inservice Testing Programs," provides alternatives to Code requirements that are acceptable to the NRC staff. Further guidance for developing an IST program is given in NUREG-1482, Revision 1, "A Guidance for Inservice Testing at Nuclear Power Plants," GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 95-07, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves."

Response: TVA understands that any exceptions or alternatives to the ASME OM Code requirements require NRC approval and will submit any such exceptions or alternatives for approval in accordance with NRC guidance and regulations.

Preliminary RAIs (taken from e-mail from NRC dated 04/28/2010):

5.5.11-01 Section 5.5.11 does not provide any information describing the design codes/standards/etc. for the pressurizer relief tank. A review of Table 3.2-2, "Summary of Criteria - Mechanical System Components," finds the pressurizer relief tank listed on Page 1 of 18, which would be an acceptable place to provide the information. However, there is no information provided on the pressurizer relief tank line in the columns. It appears that the table columns are out of alignment with the listing of components in the first column.

The staff also notes that there is no information provided on the Pressurizer Safety Valves line. Again, this appears to be because of the table column misalignment.

Response: Amendment 98 to the Unit 2 FSAR corrected the alignment issues in Table 3.2-2.

Unit 2 FSAR 3.2.1 (Seismic Classifications) presents the applicable Codes and Standards for the Structures, Systems and Components. Unit 2 FSAR Tables 3.2-2 (Summary of Criteria - Mechanical System Components) and 3.2-2a (Classification of Systems Having Major Design Concerns Related to a Primary Safety Function) provide the Codes and Standards for the Pressurizer Relief Tank /Piping and Pressurizer Safety Valves and Piping. Sheet 1 of Unit 2 FSAR Figure 5.1-1-1 (Powerhouse Unit 1 - Flow Diagram - Reactor Coolant System) shows the code class breaks for the Pressurizer Safety Valves and for the Pressurizer Relief Tank.

5.5.11-02 Section 5.5.11.3, Design Evaluation, states "The rupture disc on the relief tank have a…" This is not clear in that "…rupture disc.." indicates there is one rupture disc while "…relief tank have…" implies there is more than one rupture disc. It should be noted that the Unit 1 UFSAR contains the same words. NUREG-0847, Watts Bar Unit 1 and Unit 2 SER, Section 5.4.4 states "Tank overpressurization protection is provided by two rupture discs."

Clarify the number of rupture discs that are provided with the pressurizer relief tank.

Response: There are two parallel rupture discs on the relief tank in agreement with NUREG-0847, Watts Bar Unit 1 and Unit 2 SER, Section 5.4.4 (Ref. FSAR Figure 5.1-1, sheet 1 and Drawing 2-47W813-1).

Amendment 100 to the Unit 2 FSAR will revise the first sentence of the second paragraph of 5.5.11.3 (Design Evaluation) to state: "The two rupture discs on the relief tank have a total relief capacity equal to or greater than the combined capacity of the three pressurizer safety valves."

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 - 1. The NRC issued Amendment Nos. 40, 48, 67, and 77 to WBN Unit 1 operating license on September 23, 2002 (ML022540925), October 8, 2003 (ML032880062), January 18, 2008 (ML073520546), and May 4, 2009 (ML090920506). These amendments authorized irradiation of tritium production burnable absorber rods (TPBARs) within WBN Unit 1 core and transfer of these irradiated TPBARs through the shared WBN spent fuel pool. In granting these amendments, the NRC staff considered evaluations of the effect of storage of these TPBARs within the shared WBN spent fuel pool on heat generation and criticality prevention. However, the WBN Unit 2 FSAR through Amendment No. 97 does not address the presence of the TPBARs within the shared spent fuel pit and the effect of these TPBARs on safety functions related to fuel storage. Update the FSAR to provide appropriate information demonstrating that safety functions would be accomplished considering the effects of TPBAR storage in the shared spent fuel pool

Response: The effects of TPBARS have been properly considered in the calculation of Spent Fuel Pool (SFP) decay heat loads. Although there are no current plans to irradiate TPBARs in Unit 2, SFP decay heat calculations conservatively consider their use. See the response to **RAI SBPB 9.1 - 2.** for additional SFP decay heat information. Other aspects of TPBAR storage in the common SFP, e.g., criticality, have been previously addressed in Unit 1 analyses and licensing activities and are not impacted by Unit 2 operation. As indicated in the response to **RAI SBPB 9.1 - 2**, the Unit 2 FSAR will be appropriately updated.

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: 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 SFP decay heat loads are calculated in accordance with ANS Standard 5.1, "Decay Heat Power in Light Water Reactors," and USNRC Regulatory Guide 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).

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

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

All references to WBN Unit 1 are from the approved FSAR Amendment No. 7. All references to WBN Unit 2 are from Amendment No. 95.

Chapter 4.3.2

SNPB 4.3.2 - 1. Discuss the initial core loading strategy for WBN Unit 2.

Response: The initial core design for WBN Unit 2 was developed to meet a number of criteria. These included:

- Achieve a minimum of 10% margin to peaking limits.
- Limit the most positive moderator temperature coefficient during the cycle to less than -1.0 pcm/°F.
- Allow hot, full power axial flux difference limits of +7% and -12%.
- Meet all safety analysis limits with margin.
- Provide 405 days of full power capability to support the planned operation of cycle 1.
- Support a cycle 2 design with 500 days of full power capability if cycle 1 is shut down after 240 or 435 effective full power days of operation.
- Keep the power of the fuel at the edge of the core high enough to reduce the potential that hot leg temperature streaming will adversely impact the measured core flow in cycle 1.
- Minimize fuel cost over cycles 1 through 3.

Reload designs are typically comprised of approximately 84 fresh fuel assemblies with a large number of fresh Integral Fuel Burnable Absorber (IFBA) rods and relatively few discrete burnable absorbers. The remainder of the reload design is comprised of fuel that has been operated in one or more previous cycles.

With all fresh fuel, the initial cycle design characteristics differ from those of a reload design. A large number of Wet Annular Burnable Absorber (WABA) rods and fewer IFBA rods will be used in the initial cycle design because the WABA absorber is more effective than IFBA for maintaining a nonpositive moderator temperature coefficient.

The moderator temperature coefficient of an initial cycle design is more positive at a given boron concentration than the moderator temperature coefficient of a reload design due to lack of fission products and the decreased competition for neutrons in the design with all fresh fuel. The increased worth of the soluble boron in the all fresh fuel design increases the reactivity effects that arise from a change in moderator density.

The Unit 2 initial cycle design is very similar to the Unit 1 initial cycle design. Both designs use feed fuel enrichments of 2.1, 2.6, and 3.1 w/o in similar numbers. While the RFA-2 fuel assembly design that will be loaded for the initial Unit 2 cycle differs from the Vantage 5H fuel assembly design that was loaded in the initial Unit 1 cycle, the two fuel assembly designs have very similar neutronic characteristics. This similarity increases confidence that Unit 2 design will operate as designed.

The Unit 2 cycle design is preliminary pending completion of the safety analysis review. The review is scheduled to be completed by the end of 2010.

SNPB 4.3.2 - 2 In Table 4.3-1 (p 4.3-40) define the two numbers given for the following in the Fuel Assemblies section:

a. Diameter of Guide Thimbles (upper part)

Response: These numbers are the inside and outside diameters of the guide thimbles. The “ID” and “OD” labels were inadvertently removed.

Amendment 100 to the Unit 2 FSAR will reinstate the “ID” and “OD” labels.

b. Diameter of Guide Thimbles (lower part)

Response: These numbers are the inside and outside diameters of the guide thimbles. The “ID” and “OD” labels were inadvertently removed.

Amendment 100 to the Unit 2 FSAR will reinstate the “ID” and “OD” labels.

c. Diameter of Instrument Guide Thimbles

Response: These numbers are the inside and outside diameters of the guide thimbles. The “ID” and “OD” labels were inadvertently removed.

Amendment 100 to the Unit 2 FSAR will reinstate the “ID” and “OD” labels.

SNPB 4.3.2 - 10. In WBN Unit 2 Amendment No. 95 section 4.3.2.4.2, what is 4EF (p 4.3-20)?

Response: Amendment 98 to the Unit 2 FSAR corrected "4EF" to read "4°F." Since this was an editorial change, a change bar was not provided, and the amendment level was not revised.

SNPB 4.3.2 - 11. In WBN Unit 2 Amendment No. 95 section 4.3.2.8.5 and section 4.3.3.2, LEOPARD is referenced as reference 17, however, according to the references section, Reference 17 was deleted by Amendment 92. Why was the amendment deleted?

Response: Amendment 98 to the Unit 2 FSAR added the required reference for LEOPARD.

Chapter 4.3.3

SNPB 4.3.3 - 2. In WBN Unit 2 Amendment No. 95 section 4.3.3.3, should the reference to Section 4.3.2.2.7 be to Section 4.3.2.2.6 instead?

Response: Amendment 99 to the Unit 2 FSAR replaced "4.3.2.2.7" with "4.3.2.2.6."

Chapter 4.4.1

SNPB 4.4.1 - 1. In WBN Unit 2 Amendment No. 95 section 4.4.1.1 under the heading 'Discussion' (p 4.4-1), change 'DBN' to 'DNB'.

Response: Amendment 100 to the Unit 2 FSAR will replace "DBN" with "DNB."

Chapter 4.4.2

SNPB 4.4.2 - 2. In WBN Unit 2 Amendment 95 page 4.4-11, there are multiple locations on the right hand side of the page where equation numbers are pasted in the middle of paragraphs blocking the view of the words. Correct these errors.

Response: Amendment 98 to the Unit 2 FSAR corrected the noted discrepancies. TVA reviewed the pages currently numbered as 4.4-9 through 4.4-23 and ensured they were corrected.

SNPB 4.4.2 - 3. Confirm that WBN Unit 2 is limited to cores with only RFA-2 fuel and will not use any other type of fuel until an approved transition core methodology is submitted.

Response: The fuel fabrication contract for WBN specifies that the RFA-2 fuel design will be supplied. There are no provisions for the supply of a different fuel design. There is no plan to change the fuel design. If TVA decides to change the fuel design, TVA will comply with applicable fuel transition requirements.

SNPB 4.4.2 - 5. In WBN Unit 2 Amendment No. 95 page 4.4-18, there are multiple locations on the right hand side of the page where equation numbers are pasted in the middle of paragraphs blocking the view of the words. Correct these errors.

Response: Amendment 98 to the Unit 2 FSAR corrected the noted discrepancies. TVA reviewed pages currently numbered as 4.4-9 through 4.4-23 and ensured they were corrected.

Chapter 4.4.3

SNPB 4.4.3 - 3. In WBN Unit 2 Amendment No. 95 page 4.4-25, there is one location on the right hand side of the page where equation numbers are pasted in the middle of paragraphs blocking the view of the words. Correct this error.

Response: Amendment 98 to the Unit 2 FSAR corrected the noted discrepancies. TVA reviewed the pages currently numbered as 4.4-9 through 4.4-23 and ensured they were corrected.

Chapter 4.4.5

SNPB 4.4.5 - 1. In WBN Unit 2 Amendment No. 95 section 4.4.5.1, the figure referred to is Figure 4.4-5. In WBN Unit 2 Amendment No. 98 section 4.4.5.1, the figure referred to is Figure 4.4.6. While there is a change in the figure number between the two amendments, Amendment No. 98 has the page marked as ‘WBNP-95’ which means there have been no changes since Amendment No. 95. What are the criteria that would signify a change and cause the page to be marked ‘WBNP-98’?

Response: A portion of the changes incorporated per Amendment 98 to the Unit 2 FSAR was the addition of a new Figure 4.4-4. This resulted in the renumbering of old Figure 4.4-5 to 4.4-6. As a result, the reference to “Figure 4.4-5” was changed to “Figure 4.4-6.” The change to the correct figure number in 4.4.5.1 was an editorial change only; thus, a change in amendment number was not required for the applicable page (i.e., 4.4-32 in the Amendment 98 version).

RAIs for FSAR 15.5 [from NRC letter dated 07/12/2010 (ADAMS Accession No. ML101600278)]

<u>Accident Dose Branch – FSAR 15.5</u>

15.5 - ADB-1.a. For calculations of atmospheric dispersion factors (χ/Q values) using the ARCON96 methodology, please provide the input files (electronic files for data input into computer codes) and a discussion of the assumptions used to generate the χ/Q values.

Response: Attachment 11 provides the ARCON96 meteorological input data in electronic format. Other inputs are not available in electronic format, but are provided in hard copy format in Attachment 11. Also included in Attachment 11 is a list of assumptions and a discussion of the assumptions and methodology employed. The atmospheric dispersion factors (χ/Q values) calculated using the above inputs are listed in Unit 2 FSAR Table 15.5-14 (Atmospheric Dilution Factors At The Control Building).

15.5 - ADB-1.b. Include one or more scaled figures with true north clearly shown, when appropriate, from which distance, height, and direction inputs can be reasonably approximated. Provide the scale of each figure. Highlight all postulated sources and receptors, including the location of the control room envelop with respect to the postulated release locations.

Response: The requested figures are provided in Attachment 12.

15.5 - ADB-1.c. Please explain how distance inputs into the ARCON96 calculations were estimated (e.g., horizontal straight line distances). Please explain how the procedure used to estimate the distances properly factored in differences in heights between each source and receptor pair

Response: Distances between the sources and receptors used in the dose analysis were horizontal straight line distances calculated by triangulation using the dimensions shown on the drawings in Attachment 12. Use of the horizontal distances is conservative since the hypotenuse distance between the source and receptor would be a greater distance, resulting in smaller χ/Q values.

15.5 - ADB-1.d. Were any sources modeled as diffuse or high energy releases? If so, what is the basis for determination of the inputs specific to those cases?

Response: No sources were modeled as diffuse or high energy releases.

15.5 - ADB-2. Which χ/Q values were used in the dose assessments to model unfiltered inleakage into the control room envelope and why is use of these χ/Q values appropriate?

Response: The same χ/Q values used in the dose analysis are applied to the unfiltered inleakage. Use of these χ/Q values is appropriate since the intake air vent is a direct path into the MCR and other specific unfiltered leakage paths into the MCR have not been identified. In addition, the maximum χ/Q values used in the dose analysis represent leakage path locations relatively close to the release point, where unfiltered inleakage could originate over the total habitability envelope surface area, the majority of which is further from the release point than the air intake vents used in the dose analysis. The more remote inleakage locations would produce smaller χ/Q values as compared to locations near the release point. Thus, use of the maximum calculated χ/Q values are conservative relative to unfiltered inleakage. A total of 51 cfm unfiltered inleakage into the MCR is assumed in the dose analysis. Tracer gas testing of the MCR indicated the actual unfiltered inleakage is less than 6 cfm [TVA letter to NRC dated 08/04/2004 (ADAMS Accession No. ML042230173)]. Consequently, the dose analysis is very conservative relative to unfiltered inleakage.

15.5 - ADB-3.a. Please explain if any source/receptor pairs other than those resulting in the χ/Q values listed in Table 15.5-14 were considered. If so, which source/receptor pairs and χ/Q values were compared to determine the limiting control room χ/Q values for each design basis accident?

Response: Unit 2 FSAR Table 15.5-14 lists χ/Q values used in the FSAR chapter 15 Main Control Room (MCR) dose analysis for: 1) LOCA/FHA, 2) SGTR/MSLB/LOSS of A/C POWER, and 3) WGDT Rupture. The Loss of Coolant (LOCA) and Fuel Handling Accident (FHA) result in releases from the Shield Building Stack. The Steam Generator Tube Rupture (SGTR), Main Steam Line Break (MSLB), and Loss of A/C Power accidents results in releases from the steam generator/main steam system relief valves located near the roof of the valve vaults. The Waste Gas Decay Tank (WGDT) rupture accident results in releases from the Auxiliary Building Vent Stack.

The MCR is isolated during the above accidents, but makeup/pressurization air is supplied to the MCR through one of two intakes. The above sources and receptors are located on drawing 47W200-1 provided in Attachment 12. The χ/Q values were determined from each release point to each MCR air intake for both units, considering various building configurations. The worst case χ/Q values were selected for use in the MCR dose analysis and were included in FSAR Table 15.5-14. However, the selection of χ/Q values did consider that the Operating procedures require Operations to switch the MCR intake to the less contaminated (smaller χ/Q) air supply location 8 hours post-accident.

15.5 - ADB-3.b. Please explain how limiting releases were determined (quantitatively or subjectively).

Response: The limiting accident releases were determined based on the applicable NRC guidance documentation. For example, the LOCA analysis followed the guidelines in NRC Regulatory Guide 1.4. The FHA followed the guidance in Regulatory Guide 1.25. The SGTR and MSLB analysis followed guidance in NRC NUREG-0800, SRP 15.1.5. The Waste Gas Decay Tank rupture analysis followed guidelines in Regulatory Guide 1.24 for the design case and in NUREG-0800, Section 11.3 for the realistic analysis. No specific NRC guidance has been issued for Loss of A/C Power dose analysis. However, this analysis employed appropriate conservative assumptions.

15.5 - ADB-3.c. If only three source/receptor pairs were considered, as implied by the χ/Q values listed in Table 5.5-14, explain why they were the limiting cases. For example, was this determined by examination of plant drawings or plant walk-downs?

Response: As indicated in the response above, the χ/Q computations considered numerous sources and receptors, and the worst case χ/Q values were selected for use in the dose analysis.

15.5 - ADB-3.d. Do the postulated accident scenarios and generated χ/Q values model the limiting doses considering multiple release scenarios, including those due to loss of offsite power or other single failures?

Response: The dose analysis and associated release scenarios were based on NRC guidance documentation and considered worst case single failures. Loss of off-site power is assumed for all accidents, in addition to the worst case single failure.

15.5 - ADB-4. Please provide an electronic copy of the PAVAN computer code input, if available. Otherwise, provide a list of all inputs and assumptions used in the PAVAN calculations. A copy of the summary pages of the PAVAN outputs is acceptable to show inputs.

Response: TVA did not use the PAVAN code to calculate χ/Q values for the exclusion boundary and LPZ. Per the first paragraph of Unit 2 FSAR 2.3.4.1 (see Amendment 99 version), "Revised estimates of atmospheric diffusion expressed as dispersion factors (X/Q) have been calculated for accident releases considered as ground-level releases from the Watts Bar Nuclear Plant for specified time intervals and distances. The revised X/Q values are based on an updated onsite meteorological data base for 1974 through 1993 and RG 1.145 calculation methodology. …"

The methodology is discussed in further detail in Unit 2 FSAR Section 2.3, and the results are reported in Unit 2 FSAR Tables 2.3-66A (Atmospheric Dispersion Factors (X/q), Sec/m^3, For Design Basis Accident Analyses Based On Onsite Meteorological Data For Watts Bar Nuclear Plant) and 15A-2 [Accident Atmospheric Dilution Factors (sec/m^3)].

15.5 - ADB-5. The choice of wind speed categories used in the PAVAN computer code calculations appears to result in some clustering of the data in the lower categories. NRC Regulatory Issues Summary (RIS) 2006-4, "Experience with Implementation of Alternative Source Terms," states that input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results. Therefore, please provide justification that the wind speed categories used in the PAVAN calculations have produced adequate estimates of the exclusion area and low population zone χ/Q values for the Watts Bar site.

Response: As stated in the response to **RAI 15.5 - ADB-4.**, TVA did not use the PAVAN code to calculate χ/Q values for the exclusion boundary and LPZ.

ENCLOSURE 1

Response to Preliminary RAIS and RAIs Regarding Unit 2 FSAR
Tennessee Valley Authority - Watts Bar Nuclear Plant - Unit 2, Docket No. 50-391

Preliminary RAIs (taken from NRC e-mail of 06/08):

Quality and Vendor Branch RAIs for FSAR Section 14.2

EQVB 14.2-1. FSAR Chapter 14, Table 14.2-1, Sheet 48 of 90, "[Alternating Current] AC Power Distribution System Test Summary," TVA deleted the requirement to verify item 5 “Demonstrate manual and automatic transfer schemes operate in accordance with design drawings” under "Test Method." The NRC staff did not find a basis or justification for deletion of this test.

a. The NRC staff requests TVA to provide a basis or justification why this test is not required for start of WBN Unit 2 to demonstrate the capability of the manual and automatic transfer schemes for the AC power distribution system for dual unit operation.

Response:

1. The automatic and manual transfer features of the Unit 2 6.9kV Shutdown Boards have been tested since Unit 1 began operation and are tested every 18 months on all four 6.9kV Shutdown boards in accordance with 0-SI-211-1 which demonstrates fulfillment of TS SR 3.8.1.8.

2. Unit 2 startup and operation will be the same as Unit 1’s. Thus, there are no planned design changes to the transfer schemes of the Unit 2 6.9kV Shutdown Boards.

3. For Unit 2, the manual transfer from normal to alternate feeder is performed remotely by control room handswitch and locally by handswitches provided on the compartment doors. This is the same as the Unit 1 methodology.

4. Automatic transfers of both the Unit 1 and Unit 2 6.9kV Shutdown Boards from normal feeder to alternate or Standby power supply feeder are initiated as follows:

    a. A fault condition on the normal power supply feeder common station service transformer (CSST) or a line fault on the Preferred Offsite power supply, 161kV line, originating in the Watts Bar Hydro switchyard results in transfer to the Alternate power supply feeder. Note 4 on TVA Drawings 1-45W760-211-1 and 2-45W760-211-1 provides additional detailed information on the automatic transfer schemes for the 6.9kV Shutdown Boards.

b. A Loss of Voltage or Degraded Voltage condition of the Preferred Offsite Power supply on either the Normal or Alternate feeder, as sensed at the 6.9kV Shutdown Board(s), results in transfer to the Onsite Standby power supply. This transfer scheme will be tested on the Unit 2 6.9kV Shutdown Boards prior to Unit 2 operations in accordance with 2-PTI-262-01 and 2-PTI-262-02.

5. Both the automatic and manual transfer schemes, from the Normal feeder to the Alternate feeder, are not affected by loads or loading on the 6.9kV Shutdown Board electrical buss. The transfer scheme is solely dependent upon the initiating condition and source of the condition as noted in **4.a.** above.

6. To test the condition noted in paragraph **4.a.** above, the automatic transfer scheme test is actuated by simulating an electrical fault on the Normal feeder Offsite Power supply by rotating the appropriate protective electrical relay disc closing a contact in the transfer circuit initiating an automatic transfer from the normal feeder to the alternate feeder.

7. The protective relay that initiates the transfer from Normal feeder to Alternate feeder initiates a transfer on both the Unit 1 and the Unit 2 6.9kV Shutdown Boards simultaneously due to the common power supply.

b. Also, provide a description of the transfer scheme to include whether running loads are shed and then re-sequenced on, or if the loads are block loaded.

Response:

1. The automatic transfer schemes are described in paragraph 4. of the response to **RAI EQVB 14.2-1.a.**

2. For all automatic transfers due to transformer or line faults, the transfer from Normal feeder to Alternate feeder is classified as a Fast transfer and does not interrupt any power or loads on the 6.9kV Shutdown Boards. No loads are shed, and loads on the buss are not required to be shed and sequenced back onto the buss.

3. For all Loss of Voltage or Degraded Voltage conditions which initiate an automatic transfer to the Onsite Standby power supply, the following actions take place

once the transfer is initiated:

a. Any normal or alternate power source breaker is opened and all Normal or Alternate feeder breakers are locked out, loads are shed, the affected diesel generator (DG) is started, voltage and frequency are checked on the DG and, if within limits, the emergency DG breaker is closed. Once proper voltage is returned to the buss, the loads will then sequence on. See drawing 45W760-211-16 for sequence times.

b. There is no block loading of loads onto the buss once load shedding has occurred when supplied by the Onsite Standby power source. By design, there are some loads not shed. For example, the 6.9kV load breakers providing power to the 480V AC Shutdown Board Transformers are not shed; they remain closed, and will reenergize upon the DG breaker closing in to the 6.9kV Shutdown Board.

c. If TVA is taking credit for the Technical Specification (TS) Surveillance Requirement (SR) 3.8.1.8 currently performed for WBN Unit 1 every 18 months for deleting this test requirement then describe how the loads of WBN Unit 2 are included in this surveillance.

Response:

1. As far as performing the manual and automatic transfers for the Unit 2 6.9kV Shutdown Boards, there are no design changes being made to the transfer schemes. The schemes for Unit 2 remain the same as Unit 1 and are currently being tested at the required surveillance frequency.

2. Loads that will be added to the Unit 2 6.9kV Shutdown Boards as a result of completing Unit 2 will be tested under 2-PTI-262-01 and 2-PTI-262-02 for the Loss of Voltage and Degraded Voltage transfer schemes that result in the Onsite Standby power supply providing power to the shutdown boards.

   These preop tests will verify that, when required, the shutdown boards load shed, start the Onsite standby power supply, sequence on the required loads, with or without an accident signal present, and verify that the Onsite Standby power supply meets all required loading design calculations for Unit 2 operation.

3. The surveillance (0-SI-211-1) does not take into consideration loading on the 6.9kV Shutdown boards

when testing is conducted.

4. The manual transfers of the boards are performed under normal loading as well as the automatic transfer initiated by simulation of the Preferred Offsite power supply CSST or line fault as noted in paragraph 4.a. of the response to **RAI EQVB 14.2-1.a**.

Summary

It is appropriate to not require the preoperational testing of the Unit 2 6.9kV Shutdown Boards manual and automatic transfer due to a transformer or line fault. For the Unit 2 6.9kV Shutdown Boards, the Loss of Voltage and Degraded Voltage automatic transfer to the Onsite Standby power supply will be tested with the new loading for Unit 2 operation in accordance with 2-PTI-262-01 and 2-PTI-262-02.

References

- 45W760-211 series drawings
- 45W760-82 series drawings
- 0-SI-211-1 (18 Month 6.9 KV Shutdown Boards Transfer From Normal To Alternate Supply)
- 2-PTI-262-01 (Unit 2 Integrated Safeguards Test, Train 2A)
- 2-PTI-262-02 (Unit 2 Integrated Safeguards Test, Train 2B)

EQVB 14.2-2. FSAR Chapter 14, Table 14.2-1, Sheet 48 of 90, "AC Power Distribution System Test Summary," TVA revised Item 7 (renumbered item 6) to state that it will verify the capability of each common station service transformer (CSST) to carry the load required to supply engineered safety feature (ESF) loads for its respective load group under WBN Unit 2 loss-of-coolant accident (LOCA) conditions. The NRC staff finds this commitment to be different than the staff's acceptance documented in the Supplements 14 and 16 to the NUREG -0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Units 1 and 2." In Supplement 16 to NUREG-0847, the NRC staff accepted TVA's position to demonstrate the capability of each CSST to carry the load required to supply ESF loads on one unit (WBN Unit 1**)** under LOCA condition since TVA, at that time, was not seeking an operating license for WBN Unit 2. The NRC staff documented its acceptance of TVA's position based on the commitment that before issuance of an Operating License for WBN Unit 2, TVA would have to demonstrate the capability of each CSST to carry the load required to supply ESF loads of one unit (WBN Unit 2) under LOCA conditions in addition to power required for shutting down the non-accident unit (WBN Unit 1). Therefore, the NRC staff requests that TVA revise its test commitment to verify the capability of each CSST with LOCA conditions in WBN Unit 2 in addition to power required for normal shutdown (non-accident loads) of WBN Unit 1, or provide an explanation and justification why the original commitment was revised.

Response: See the response to **RAI 14 - 1**.

EQVB 14.2-3. FSAR Chapter 14, Table 14.2-1, Sheet 49 of 90, "AC Power Distribution System Test Summary," TVA deleted the requirement to verify item 2 under Acceptance Criteria. The NRC staff did not find a basis or justification for deletion of this test.

a. The NRC staff requests TVA to provide a justification why this test is not required for start of WBN Unit 2.

Response: This test will be performed by 2-PTI-262-01 (Unit 2 Integrated Safeguards Test, Train 2A) and 2-PTI-262-02 (Unit 2 Integrated Safeguards Test, Train 2B).

Amendment 100 to the Unit 2 FSAR will add item 2, “Power supply to safety related loads will automatically and manually transfer to the onsite (standby) diesel units from the normal or alternate supply or manually from the diesel generator units back to the normal or alternate supply as described by FSAR Section 8.3.1” back to Table 14.2-1, Sheet 49.

b. TVA should describe how the loads of WBN Unit 2 are addressed with respect to the capability of the manual and automatic transfer schemes for the AC power distribution system between onsite (standby) diesels units from normal or alternate supply for dual-unit operation.

Response: Unit 2 startup and operation will be the same as Unit 1’s. Thus, there are no planned design changes to the transfer schemes. The only change in loading will be the addition of Unit 2 loads not previously in service and tested with Unit 1. The additional Unit 2 Engineered Safety Features loads are accounted for by tests 2-PTI-262-01 (Unit 2 Integrated Safeguards Test, Train 2A) and 2-PTI-262-02 (Unit 2 Integrated Safeguards Test, Train 2B).

EQVB 14.2-4. FSAR Chapter 14, Table 14.2-1, Sheet 44 of 89, "Diesel Generator Test Summary," TVA deleted the requirements to verify items 1 through 5 under Test Method. The NRC staff did not find a basis or justification for deletion of these tests. The NRC staff requests TVA to provide a justification why these tests are not required for start of WBN Unit 2. If these tests are currently being performed under the WBN Unit 1 TS SRs, then provide a summary of the impact on these tests for the WBN Unit 2 diesel generators due to dual- unit operation. Also, identify the WBN Unit 1 TS surveillances that are currently being performed to accomplish these tests.

Response:

1. These tests were previously performed as part of declaring the DGs functional and operable for Unit 1 operation. All four DGs were designated as required for Unit 1 operations.

2. Preop testing for these Test Methods was completed for all DG systems and support systems on Unit 1. The DGs are being maintained operable per the Unit 1 TS, including surveillance requirements, for single unit operation.

3. In changing to a dual unit operation, the only impact on the current testing methodology is to add the Unit 2 loads not currently being tested and to revise the appropriate surveillance instructions. Unit 2 preop testing will test the additional loads on the Unit 2 DGs to confirm design calculations.

4. The Unit 2 DGs will be tested again to account for the additional loading required for Unit 2 operations prior to being declared operable per the Unit 2 TS for Unit 2 operations. The testing will satisfy Unit 1 TS SRs during the Unit 2 preop testing in order to be able to call the Unit 2 DGs and Unit 2 Shutdown Boards TS operable for Unit 1 coming out of Unit 1 RF10. The Unit 2 PTIs will contain all SRs required for Unit 1 and later Unit 2 SRs, as currently known.

<u>Summary</u>

It is appropriate to not require the five items listed under Test Method due to the previous testing and acceptance for Unit 1 operations. The Unit 2 DGs have been maintained operable per the Unit 1 TS since that acceptance testing. The only impact on the Unit 2 DGs due to dual unit operations is the new Unit 2 loads that will be added to the DG. Integrated Safeguards testing for Unit 2 operations will retest the Unit 2 DGs and 6.9kV Shutdown Boards to ensure the components will fulfill their required safety function as well as all design features.

References:

- 1-PTI-262-01, Integrated Safeguards Test, Unit 1
- 2-PTI-262-01, Integrated Safeguards Test, Unit 2, Train 2A
- 2-PTI-262-02, Integrated Safeguards Test, Unit 2, Train 2B
- 0-SI-82-5, 18 Month Loss of Offsite Power - DG 2A-A
- 0-SI-82-6, 18 Month Loss of Offsite Power - DG 2B-B
- 0-SI-82-12-A, Monthly DG Start and Load Test DG 2A-A
- 0-SI-82-12-B, Monthly DG Start and Load Test DG 2B-B
- 0-SI-82-15, 24 Hour Load Run - DG 2A-A
- 0-SI-82-16, 24 Hour Load Run - DG 2B-B
- 0-SI-82-19-A, 184 Day Fast Start and Load Test DG 2A-A
- 0-SI-82-20-B, 184 Day Fast Start and Load Test DG 2B-B
- 0-SI-215 series - Series of surveillances on the DG batteries and chargers.

EQVB 14.2-5. FSAR Chapter 14, Table 14.2-1, Sheet 45 of 89, "Diesel Generator Test Summary," TVA deleted the requirement to verify items 10 under Test Method.

a. If this test is currently performed on WBN Unit 2 diesel generators under the WBN Unit 1 TS SR 3.8.1.14 requirements, then confirm that the WBN Unit 1 TS SR 3.8.1.14 accomplishes this test.

Response: 0-SI-82-15 (24 Hour Load Run - DG 2A-A) and 0-SI-82-16 (24 Hour Load Run - DG 2B-B) are performed to satisfy Unit 1 TS SR 3.8.1.14 requirements. This TS SR has a frequency of 18 months and may be performed anytime in Modes 1 through 4, if required.

This 24-hour test fulfills the requirements of the items mentioned in Test Method 10.

b. Also, confirm that this surveillance performed for verifying WBN Unit 2 diesel generator capacity envelops the design-basis accident (DBA) loads of WBN Unit 2 plus the power required for the WBN Unit 1 loads.

Response: The capacity of the Unit 2 diesel generators to envelope the Unit 2 design-basis accident (DBA) loads plus the common (shared) Unit 1 loads will be demonstrated by tests 2-PTI-262-01 (Unit 2 Integrated Safeguards Test, Train 2A) and 2-PTI-262-02 (Unit 2 Integrated Safeguards Test, Train 2B).

EQVB 14.2-6. The FSAR Table 14.2-1, Chapter 14 (Sheets 44 and 45 of 89), "Diesel Generator Test Summary," does not list a test that the WBN Unit 2 diesel generators automatic trip are bypassed on automatic or emergency start signal except for engine over speed and generator differential current.

a. The NRC staff requests TVA to provide a justification why this test is not required for start of WBN Unit 2.

Response: Currently, surveillances performed by Unit 1 per 0-SI-82-3 (18 Month Loss of Offsite Power With Safety Injection – DG 1-AA) and 0-SI-82-4 (18 Month Loss of Offsite Power With Safety Injection – DG 1-BB) check the engine and generator trips are disconnected as required every 18 months. When the new surveillances for Unit 2 (i.e., 2-SI-82-5, and 2-SI-82-6) are written, the Unit 2 DG checks will be removed from the Unit 1 surveillances.

This feature will also be verified as part of 2-PTI-262-01 (Unit 2 Integrated Safeguards Test, Train 2A) and 2-PTI-262-02 (Unit 2 Integrated Safeguards Test, Train 2B).

To be in compliance with Reg. Guide 1.9 and System Description N3-82-4002 (Standby Diesel Generator System) Amendment 100 to the Unit 2 FSAR will add a requirement to test the above features to Table 14.2-1.

b. If this test is currently performed on WBN Unit 2 diesel generators under the WBN Unit 1 TS SRs, then identify the WBN Unit 1 TS SR that accomplishes this test.

Response: The current Unit 1 TS SRs are applied to both the Unit 1 and the Unit 2 diesel generators. Unit 2 is tested to the same requirements as Unit 1 (i.e., TS SR 3.8.1.13).

ENCLOSURE 2

List of Regulatory Commitments

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

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

2. The analysis for Panels 2-L-11A and 2-L-11B will be submitted to the NRC by November 30, 2010.

3. The qualification for PAMS Cabinet and Components and Main Control Room Components will be submitted to the NRC by January 14, 2011.

4. A non-proprietary version of and an affidavit for withholding for EQ-EV-39-WBT, Revision 1 will be submitted to the NRC by November 30, 2010.

5. A non-proprietary version of and an affidavit for withholding for TR-1136 will be submitted to the NRC by December 17, 2010.

6. A non-proprietary version of and an affidavit for withholding for 16690-QTR, Revision 0 will be submitted to the NRC by November 30, 2010.

7. A corrected proprietary version of, a non-proprietary version of, and an affidavit for withholding for Thermo Fisher Scientific Qualification Report No. 864, Rev. 0 will be submitted to the NRC by November 15, 2010.

8. Unit 2 System Description for the Reactor Coolant System (WBN2-68-4001) will be revised to reflect required revisions to the PTLR by September 17, 2010.

9. Once the date for issuance of an operating license becomes more certain, TVA will update the sentence regarding the Code of Record in accordance with 10 CFR 50.55a(f)(4)(i).

ENCLOSURE 3

List of Files Provided on Enclosed Optical Storage Media (OSM)

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

File Name	Fill Size - Bytes
001 - ATTACHMENT 1 - Foxboro Test Results	338,284,559
002 - ATTACHMENT 2 - EQ-EV-39-WBT, Revision 1 (Proprietary)	938,156
003 - ATTACHMENT 3 - TR-1136 (Proprietary)	14,939,714
004 - ATTACHMENT 4 - 16690-QTR (Proprietary)	21,740,229
005 - ATTACHMENT 5 - Qualification Report 864 (Proprietary)	9,983,274
006 - ATTACHMENT 6 - Pictorial Depiction of APS	400,199
007 - ATTACHMENT 7 - AC APS Analysis	213,153
008 - ATTACHMENT 8 - AC APS Analysis	142,157
009 - ATTACHMENT 9 - 125V DC Vital Battery System Analysis	1,408,247
010 - ATTACHMENT 10 - 125V DC Vital Battery System Analysis	361,406
011 - ATTACHMENT 11 - Atmospheric Dispersion Factors Supporting Information	1,651,483
012 - ATTACHMENT 12 - Drawings to Support Accident Dose Review	1,321,171