Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-15-170

August 28, 2015

10 CFR 50.90

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

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NFP-90
NRC Docket No. 50-390

Subject: **Responses to NRC Audit Review Questions for Watts Bar Nuclear Plant Unit 1 Essential Raw Cooling Water and Component Cooling Water System License Amendment Request**

References:

1. Letter from TVA to NRC, "Watts Bar Nuclear Plant Unit 1 - Application to Revise Technical Specifications for Component Cooling Water and Essential Raw Cooling Water to Support Dual Unit Operation (TS-WBN-15-13)," dated June 17, 2015 [ML15170A474]

2. Email from NRC to TVA, "Preliminary Draft RAIs Associated with Proposed WBN 1 ERCW and CCS Technical Specifications LAR," dated July 2, 2015

3. Letter from NRC to TVA, "Watts Bar Nuclear Plant, Unit 1 - Supplemental Information Needed for Acceptance of Requested Licensing Action Regarding Application to Add Technical Specifications to Support Dual-Unit Operations (TAC No. MF6376)," dated July 9, 2015 [ML15187A403]

4. Letter from TVA to NRC, "Responses to NRC Acceptance Review Questions for Watts Bar Nuclear Plant Unit 1 Essential Raw Cooling Water and Component Cooling Water System License Amendment Request (TAC No. MF6376)," dated July 14, 2015 [ML15197A357]

By letter dated June 17, 2015, Tennessee Valley Authority (TVA) submitted a request for a change to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant (WBN) Unit 1 (Reference 1). The proposed change would create new Technical Specifications (TS) 3.7.16, "Component Cooling System (CCS) - Shutdown," and TS 3.7.17, "Essential Raw Cooling Water (ERCW) System - Shutdown," to support dual unit operation of WBN Units 1 and 2. By email dated July 2, 2015, the Nuclear Regulatory Commission (NRC) provided requests for additional information (RAI) on the proposed WBN Unit 1 license amendment (Reference 2). By letter dated July 9, 2015, the NRC requested supplemental information associated with the proposed WBN Unit 1 license amendment (Reference 3). By letter dated July 14, 2015 (Reference 4), TVA submitted the requested supplemental information and responses to the NRC acceptance review questions, including proposed changes to TS 3.7.16 and TS 3.7.17.

Following submittal of the requested supplemental information and responses to the NRC RAIs, the NRC indicated that sufficient information was provided by TVA to support the NRC review of the proposed license amendment request (LAR). However, to facilitate a more efficient and timely interaction between the NRC and TVA, the NRC decided to perform an audit of the proposed LAR in the NRC White Flint offices located in Rockville, MD during the weeks of July 27 to July 31, 2015, August 3 to August 7, 2015, and August 25 to August 28, 2015. During the audit, the NRC and TVA discussed numerous questions related to the LAR.

The enclosure provides the TVA responses to the NRC audit review questions. As a result of the TVA responses to the NRC audit review questions, changes are required to TS 3.7.16, TS 3.7.17, and the associated Bases. To address the NRC concern regarding the availability of Reactor Coolant System loops within the initial seven hours after reactor shut down, a change to additional TSs may be required. In addition, the enclosure includes wording additions for Final Safety Analysis Report (FSAR) Chapters 6, 9 and 10 to clarify ice bed sublimation assumptions and water sources available to the AFW System. The proposed changes to the TS will be submitted in a license amendment request by September 11, 2015. The FSAR changes will be incorporated in WBN Unit 2 FSAR Amendment 114.

Consistent with the standards set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.92(c), TVA has determined that the responses, as provided in this letter, do not affect the no significant hazards considerations associated with the proposed license amendment to add TS 3.7.16 and TS 3.7.17 previously provided in Reference 1. TVA has further determined that the proposed amendment still qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosure to the Tennessee Department of Environment and Conservation.

There are no new regulatory commitments associated with this letter. Please direct any questions concerning this matter to Gordon Arent at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 28th day of August 2015.

Respectfully,

J. W. Shea

Digitally signed by J. W. Shea
DN: cn=J. W. Shea, o=Tennessee Valley Authority, ou=Nuclear Licensing, email=jwshea@tva.gov, c=US
Date: 2015.08.28 22:26:28 -04'00'

J. W. Shea
Vice President, Nuclear Licensing

Enclosure

cc: See Page 3

Enclosure:

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

cc (Enclosure):

U.S. Nuclear Regulatory Commission, Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant, Unit 1
NRC Project Manager - Watts Bar Nuclear Plant, Unit 1
Director - Division of Radiological Health – Tennessee State Department of Environment and Conservation

ENCLOSURE

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Background

By letter dated June 17, 2015, Tennessee Valley Authority (TVA) submitted a request for a change to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant (WBN) Unit 1 (Reference 1). The proposed change would create new Technical Specifications (TS) 3.7.16, "Component Cooling System (CCS) - Shutdown," and TS 3.7.17, "Essential Raw Cooling Water (ERCW) System - Shutdown," to support dual unit operation of WBN Units 1 and 2. By email dated July 2, 2015, the Nuclear Regulatory Commission (NRC) provided requests for additional information (RAI) on the proposed WBN Unit 1 license amendment (Reference 2). By letter dated July 9, 2015, the NRC requested supplemental information associated with the proposed WBN Unit 1 license amendment (Reference 3). By letter dated July 14, 2015 (Reference 4), TVA submitted the requested supplemental information and responses to the NRC acceptance review questions, including proposed changes to TS 3.7.16 and TS 3.7.17.

Following submittal of the requested supplemental information and responses to the NRC RAIs, the NRC indicated that sufficient information was provided by TVA to support the NRC review of the proposed license amendment request (LAR). However, to facilitate a more efficient and timely interaction between the NRC and TVA, the NRC decided to perform an audit of the proposed LAR in the NRC White Flint offices located in Rockville, MD during the weeks of July 27 to July 31, 2015, August 3 to August 7, 2015, and August 25 to August 28, 2015. During the audit, the NRC and TVA discussed numerous questions related to the LAR.

This enclosure provides the TVA responses to the NRC audit review questions. As a result of the TVA responses to the NRC audit review questions, changes are required to TS 3.7.16, TS 3.7.17, and the associated Bases. To address the NRC concern regarding the availability of Reactor Coolant System loops within the initial seven hours after reactor shut down, a change to additional TSs may be required. In addition, this enclosure includes wording additions for Final Safety Analysis Report (FSAR) Chapters 6, 9 and 10 to clarify ice bed sublimation assumptions and water sources available to the AFW System. The proposed changes to the TS will be submitted in a license amendment request by September 11, 2015. The FSAR changes will be incorporated in WBN Unit 2 FSAR Amendment 114.

ENCLOSURE

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

References

1. Letter from TVA to NRC, "Watts Bar Nuclear Plant Unit 1 - Application to Revise Technical Specifications for Component Cooling Water and Essential Raw Cooling Water to Support Dual Unit Operation (TS-WBN-15-13)," dated June 17, 2015 [ML15170A474]

2. Email from NRC to TVA, "Preliminary Draft RAIs Associated with Proposed WBN 1 ERCW and CCS Technical Specifications LAR," dated July 2, 2015

3. Letter from NRC to TVA, "Watts Bar Nuclear Plant, Unit 1 - Supplemental Information Needed for Acceptance of Requested Licensing Action Regarding Application to Add Technical Specifications to Support Dual Unit Operations (TAC No. MF6376)," dated July 9, 2015 [ML15187A403]

4. Letter from TVA to NRC, "Responses to NRC Acceptance Review Questions for Watts Bar Nuclear Plant Unit 1 Essential Raw Cooling Water and Component Cooling Water System License Amendment Request (TAC No. MF6376)," dated July 14, 2015 [ML15197A357]

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
1.	BOP - 1 TS 3.0.4 does not allow Mode changes when applicable LCOs for that Mode are not met. The problem with possible Mode change from 4 to 3 after shutdown is that equipment may have been taken out of service that is required to meet the LCOs of TS that are required for Mode 3. A similar statement can be made for a Mode change from 5 to 4. The proposed TS make no provision for suspending or stopping the process of making more equipment inoperable that otherwise would be required to support Mode change, if the proposed LCO is not met. Explain why the proposed TSs do not include an Action to stop making more equipment inoperable that would be required for the Mode change when the proposed TS LCOs are not met, or make provisions to correct the issues. <u>Date Posted</u>: 07/31/15	With one Component Cooling Water System (CCS) or Essential Raw Cooling Water (ERCW) System train inoperable, a loss of redundancy has occurred. However, the capability to mitigate an accident in one unit and cooldown the other unit (or maintain the other unit in a cooldown condition) is maintained by the remaining operable CCS and ERCW trains. Therefore, a Mode change from 5 to 4 or from 4 to 3 is not anticipated. During a normal shutdown, decay heat removal is via the reactor coolant system (RCS) loops until sometime after the unit has been cooled down to Residual Heat Removal (RHR) System entry conditions (T_{cold} < 350ºF). Therefore, as LCO 3.7.16 and LCO 3.7.17 become Applicable (entry into Mode 4), the RCS loops are still operable. At this point, LCO 3.7.16 requires an additional CCS Train B pump powered from and aligned to the CCS Train B header, and LCO 3.7.17 requires one additional ERCW pump be capable of being powered from and aligned to each ERCW train. However, the requirement of LCO 3.4.6, "RCS-Loops - MODE 4," is still being met by the two operable RCS loops. If the requirement of either LCO 3.7.16 or LCO 3.7.17 is not met, maintaining the unit in Mode 4 with decay heat removal from the RCS loops is preferred, given the additional methods available to remove decay heat (i.e., RCS loops). However, if TS Required Actions require entry into Mode 5, the remaining operable RHR loop is sufficient to cooldown the unit to and maintain it in Mode 5, even with a concurrent loss of coolant accident (LOCA) in the other unit. Therefore, it is unnecessary for TS to provide provisions for suspending or stopping the process of making more equipment inoperable.
2.	BOP – 2 TS 3.7.7 and TS 3.7.8 have provision to enter LCO 3.4.6 with one train inoperable because ERCW and CCS are support systems for decay heat removal. Otherwise TS 3.0.6 could allow LCO 3.4.6 to not be entered. Standard TS are similarly worded. It appears that proposed TS 3.7.16 and 3.7.17 actions for one train inoperable should similarly have provisions for entering TS 3.4.6. TS 3.4.6 would lead to different action than what TS 3.7.16 and TS 3.7.17	The Actions of LCO 3.7.16 and LCO 3.7.17 are predicated on the preference to maintain the unit in a condition with multiple methods of decay heat removal available, i.e., maintain the unit in Mode 4 with two RCS loops operable in addition to the remaining operable RHR loop. This action precludes entry into the LCO 3.4.6 Actions, as LCO 3.4.6 is met with two operable RCS loops and one RCS loop in operation. However, if it is necessary to place the unit in Mode 5 to comply with TS Required Actions, LCO 3.7.16 and LCO 3.7.17 Actions require the unit to be placed in Mode 5 in 24 hours. If the Action to verify two RCS loops operable and one RCS loop in operation cannot be met, no actions are specified. Therefore, LCO 3.0.3 applies, requiring the unit to be placed in MODE 5 in 37 hours. With one ERCW train inoperable and Required Actions

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	currently propose for 1 loop inoperable. Explain why the proposed TS differ from 3.7.7 and 3.7.8 in this respect, or correct the issue. Date Posted: 07/31/15	require the unit to be placed in MODE 5; Condition A applies, requiring the unit to be placed in MODE 5 in 24 hours. These Actions are conservative to the Required Actions of LCO 3.7.7, LCO 3.7.8, and LCO 3.4.6 when there are two operable RCS loops, and are consistent with the requirements of LCO 3.7.7, LCO 3.7.8, and LCO 3.4.6 when there are no operable RCS loops and one inoperable RHR loop.
3.	BOP - 3 TS 3.7.16 and TS 3.7.17 have required action to verify T_{avg} > 200 F. But the bases for TS 3.4.6 says with one RHR train available, it would be safer to be in Mode 5 because if the remaining train of decay heat was lost, the loss would occur at a lower temperature. Explain this difference between TS 3.4.6 and proposed TS 3.7.16 and 3.7.17. Date Posted: 07/31/15	Similar to the response to Question #2, decay heat removal is via the RCS loops until sometime after the unit has been cooled down to RHR entry conditions (T_{cold} < 350ºF). As LCO 3.7.16 and LCO 3.7.17 become applicable, the requirement of LCO 3.4.6 is still being met by the two operable RCS loops. If the requirements of either LCO 3.7.16 or LCO 3.7.17 are not met, maintaining the unit in Mode 4 with decay heat removal from the RCS loops is preferred, given the additional methods available to remove decay heat. This action precludes entry into the LCO 3.4.6 Actions, as LCO 3.4.6 is met with two operable RCS loops and one RCS loop in operation. However, if it is necessary to place the unit in Mode 5 to comply with TS Required Actions, LCO 3.7.16 and LCO 3.7.17 Actions require the unit to be placed in Mode 5 in 24 hours. If the Action to verify two RCS loops operable and one RCS loop in operation cannot be met, no actions are specified. Therefore, LCO 3.0.3 applies, requiring the unit to be placed in MODE 5 in 37 hours. With one ERCW train inoperable and Required Actions require the unit to be placed in MODE 5, Condition A applies, requiring the unit to be placed in MODE 5 in 24 hours. These Actions are conservative to the Required Actions of LCO 3.7.7, LCO 3.7.8, and LCO 3.4.6 when there are two operable RCS loops, and are consistent with the requirements of LCO 3.7.7, LCO 3.7.8, and LCO 3.4.6 when there are no operable RCS loops and one inoperable RHR loop.
4.	BOP - 4 TS 3.7.8 for ERCW has a provision for entering TS 3.8.1 for emergency diesel generators made	LCO 3.7.16 and LCO 3.7.17 provide requirements in addition to those of LCO 3.7.7 and LCO 3.7.8. However, the additional requirements of LCO 3.7.17 are not required for DG operability. There is sufficient flow to the diesel generators (DGs) from ERCW without a third ERCW pump in each train to support DG Operability. Although the requirements of

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	inoperable by ERCW when one ERCW loop is inoperable. Discuss why TS 3.7.17 does not have the same provision. Date Posted: 07/31/15	LCO 3.7.17 may not be met (i.e., a third pump capable of being aligned to each ERCW Train) the requirements of LCO 3.7.8 are still met. If the requirements of LCO 3.7.8 are not met, the Actions of LCO 3.7.8 include the requirement to enter the Conditions and Required Actions of LCO 3.8.1 for DGs made inoperable by ERCW.
5.	BOP - 5 TS 3.7.17 has a Note for LCO 3.0.3 to suspend Mode change. For a scenario of two ERCW trains inoperable per TS 3.7.17, but at least one ERCW train operable per TS 3.7.8, provide discussion and justification why it is safer to stay in Mode 4 and not continue cooldown to Mode 5 for this TS condition. Date Posted: 07/31/15	With two ERCW trains inoperable for LCO 3.7.17, there may be insufficient ERCW flow available to place the non-accident unit in Mode 5 during the mitigation of a LOCA on the other unit. Otherwise, the preference is to maintain the unit in a condition with multiple methods of decay heat removal available, i.e., maintain the unit in Mode 4 with two RCS loops operable.
6.	BOP - 6 In TS 3.7.16 and TS 3.7.17 for one train inoperable, the TS Actions make a distinction between a normal shutdown and a TS required shutdown. Explain the distinction in shutdown requirements between a normal shutdown and a TS required shutdown for TS 3.7.16 and 3.7.17 Date Posted: 07/31/15	Similar to the response to Question #2, decay heat removal is via the RCS loops until sometime after the unit has been cooled down to RHR entry conditions (T_{cold} < 350ºF). As LCO 3.7.16 and LCO 3.7.17 become applicable, the requirement of LCO 3.4.6 is still being met by the two operable RCS loops. If the requirements of either LCO 3.7.16 or LCO 3.7.17 are not met, Condition B requires that the unit be maintained in Mode 4 (with decay heat removal from the RCS loops). Maintaining the unit in Mode 4 with additional methods of decay heat removal available minimizes the likelihood of a situation where the decay heat and residual heat of the unit exceeds the capability of the available RHR loop resulting in the possibility of an unintentional Mode change. If the Required Actions and Completion Times of Condition B are not met, no actions are specified. Therefore, LCO 3.0.3 applies, requiring the unit to be placed in Mode 5 in 37 hours. With one CCS train inoperable and Required Actions require the unit to be placed in Mode 5, Condition A applies, requiring the unit to be placed in Mode 5 in 24 hours. If TS Required Actions require entry into Mode 5, the remaining operable RHR loop is sufficient to cooldown the unit to and maintain it in Mode 5, even with a concurrent LOCA in the other unit.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
7.	BOP - 7 TVA has made a general statement that there is adequate clean water on site to support a 48 hour cooldown with aux feedwater since: • Unit 1 CST @ 395,000 gallons (TS SR 3.7.6.1 requires 200,000 gallons) • Unit 2 CST @ 395,000 gallons (TS SR 3.7.6.1 (Rev J) requires 200,000 gallons) • Demin tank @ 500,000 gallons • Aux feedwater storage tank @ 500,000 gallons (added for FLEX mitigating strategies) TVA's response in the supplement included appeared to credit cross-tie between the Unit 1 and Unit 2 CSTs as well as reliance on water stored in the FLEX tank. It is not clear how these clean water sources are currently identified and credited within the licensing basis to support "normal" hot standby cooling functions, as this does not appear to be described in the FSAR, and other licensing basis related references, as noted below: <u>From Watts Bar FSAR:</u> 9.2.6.3 The ERCW system pool quality feedwater will be used during an extreme emergency when safety is the prime consideration and steam generator cleanliness is of secondary importance. 10.4.9.2 Since the ERCW system supplies poor quality water, it is not used except in emergencies when the condensate supply is unavailable.	See responses to BOP-10 and BOP-11.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	From SE in NUREG 0847, Supp 23 (June 2011): TVA's proposed clarification to the FSAR is acceptable to the NRC staff. Because the CSTs are credited only for the SBO event under 10 CFR 50.63, and TVA does not plan to share CSTs between the units during plant operation, the staff concludes that TVA satisfies GDC 5 regarding the CSTs. Confirmation by the staff of TVA's change to FSAR Section 10.4.9 to reflect TVA's intention to operate with each CST isolated from the other is Open Item 62 (Appendix HH). Date Posted: 07/31/15	
8.	BOP - 8 Because the AFW takeoff is at the end of the ERCW system, after heat removal from the safety related heat exchangers, confirm ERCW to AFW will still be at the assumed maximum allowable temperature to satisfy Chapter 15 requirements (80-120°F). Date Posted: 07/31/15	TVA notes the AFW inlet temperatures are affected by the component cooling heat exchangers and the containment spray heat exchangers as well as miscellaneous loads. The component cooling heat exchanger analysis demonstrates that ERCW discharge temperature may exceed 120ºF but these are cases where AFW is not required. However, an examination of the containment spray heat exchanger discharge layout with respect to AFW suction indicates cases where AFW could exceed 120ºF when AFW is required. The following arrangements are noted: • Motor driven AFW 1A-A/TDAFW 1A-S is upstream of CCS heat exchanger 1A but downstream of CCS heat exchanger 2A. • Motor driven AFW 1B-B/TDAFW 1A-S is upstream of CCS heat exchanger 2B but downstream of CCS heat exchanger 1B. • Motor driven AFW 2A-A/TDAFW 2A-S is upstream of CCS heat exchanger 1A but downstream of CCS heat exchanger 2A. • Motor driven AFW 2B-B/TDAFW 2A-S is upstream of CCS heat exchanger 2B but downstream of CCS heat exchanger 1B. If a LOCA was postulated in unit 1:

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		• MDAFW 1A-A would be bounded by 120ºF and MDAFW 2A-A would be bounded by 120ºF. • MDAFW 1B-B would be bounded by 130ºF and MDAFW 2B-B would be bounded by 130ºF. If a LOCA was postulated in unit 2: • MDAFW 1A-A would be bounded by 130ºF and MDAFW 2A-A would be bounded by 130ºF. • MDAFW 1B-B would be bounded by 120ºF and MDAFW 2B-B would be bounded by 120ºF. Therefore, there are four cases where the AFW inlet temperature might exceed 120ºF. Impacts would be as follows: <u>LBLOCA</u> The large break LOCA is described in FSAR Section 15.4. For the case where the AFW is on the same unit as the large break LOCA (U1/MDAFW 1B-B and U2/MDAFW 2A-A) the AFW is only used to initially fill the steam generators. The large break LOCA does not credit the continued use of AFW. This filling would occur prior to switchover to containment spray recirculation and therefore would happen prior to the temperature reaching 130ºF at the discharge of the containment spray heat exchanger. The large break LOCA peak clad temperature and the containment peak pressure analysis would not be impacted. For the case where the large break LOCA was in the opposite unit (U1/MDAFW 2B-B and U2/MDAFW 1A-A), the AFW would be used to shutdown the non-LOCA unit and may be at a higher ERCW discharge temperature. This would result in slightly warmer water for unit shutdown and would have no deleterious effect on the non-LOCA unit. <u>SBLOCA</u> The small break LOCA is described in FSAR Section 15.3. Unlike the large break LOCA, the small LOCA credits the continued use of AFW for cooling the LOCA unit. Break sizes for small LOCAs examined in the FSAR range from 2 inches to over 8 inches in diameter. It can be observed from FSAR Table 15.3-2 and supporting calculations that for all but the smallest break size, the transient has peaked prior to switchover to the containment sump recirculation and therefore would not be impacted. The smallest break peaks later in the

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		transient but at a much lower peak clad temperature. Because the only impact of the higher AFW temperature is a slightly increased enthalpy for the liquid entering the generator and because the majority of the energy is removed by vaporization (inlet enthalpy of 81 Btu/lb versus 91 Btu/lb for AFW compared to steam at approximately 1183 Btu/lb), a less than 1% change in enthalpy would be insignificant to the smallest LOCA. The smallest LOCA peaks at 1009ºF for Unit 2 with considerable margin to the limiting peak clad temperature of 2200ºF and margin to the limiting small break LOCA peak clad temperature of 1183.9ºF. <u>NPSH</u> A NPSH analysis indicates AFW will perform acceptably at 130ºF with NPSH margin. Condition Report (CR) 1072659 has been initiated to document this issue. A copy of the CR has been uploaded to the Sharepoint Site. The AFW system description, FSAR, and associated calculations will be revised to note cases where the AFW maximum temperature of 120ºF may be exceeded.
9.	BOP – 9 Describe the ERCW and CCS analysis as it pertains to a LOCA in one unit while in MODE 4 and a controlled shutdown of the other Unit as it enters MODE 4 or 5. <u>Date Posted</u>: 07/31/15	When the opposite unit has been shutdown for a period of time, the additional CCS and ERCW pump requirements of LCO 3.7.16 and LCO 3.7.17 are not required to ensure adequate decay heat removal by the RHR System. However, there may be some scenarios when the opposite unit has been shutdown for greater than 48 hours that the heat removal capacity of the RHR System is insufficient without the CCS and ERCW System requirements of LCO 3.7.16 and LCO 3.7.17 being applicable. Therefore, TVA will remove Applicability Note b, so that the Applicability of LCO 3.7.16 and LCO 3.7.17 in Modes 4 and 5 is dependent on whether the associated unit has been shutdown for less than 48 hours. This response supersedes the response provided to Acceptance Review Question #2 provided in TVA letter dated July 14, 2015. The following events are required to be supported by the CCS and ERCW configurations proposed in TS 3.7.16 and TS 3.7.17. The CCS shall be designed to remove heat from potentially or normally radioactive heat loads during any mode of normal operation, and incidents of moderate frequency. In addition, the CCS shall be designed to remove heat from the RHR HXs and various pump

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		seal and/or lube oil coolers during infrequent incidents, and limiting faults. The CCS is required to mitigate the consequences of Design Basis Events (DBEs). The required DBEs and associated safety functions for the CCS are in WB-DC-40-64. <u>EVENTS IN WB-DC-40-64 THAT CREDIT CCS</u> Fire Operating Basis Earthquake Safe Shutdown Earthquake Tornado Combustible Gases Inside Containment Control Room Evacuation Internally Generated Missiles General High Energy Line Break Heavy Load Drop Small Break LOCA Large Break LOCA Steam Generator Tube Rupture Rupture of a Control Rod Drive Mechanism Housing Waste Gas Decay Tank Rupture Fuel Handling Accident Loss of External Electrical Load and/or Turbine Trip Loss of Offsite Power Main Steam Line Break Main Feedwater Line Rupture Event Accidental Depressurization of Main Steam System Loss of Normal Feedwater Excess Heat Removal Due to Feedwater System Malfunction Moderate Energy Line Break Partial Loss of Forced Reactor Coolant Flow Single Reactor Coolant Pump Locked Rotor or Shaft Break Complete Loss of Forced Reactor Coolant Flow Excessive Load Increase Incident Accidental Depressurization of The Reactor Coolant System Inadvertent Safety Injection Operation - Power Operation Uncontrolled RCCA Bank Withdrawal From a Subcritical or Hot Zero Power Condition Uncontrolled RCCA Bank Withdrawal At Power

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		Single RCCA Withdrawal At Full Power RCCA Misalignment Uncontrolled Boron Dilution Improper Fuel Assembly Loading Anticipated Transient Without Scram Failure of Nonsafety-Related Control Systems as an Initiating Event Minor Secondary System Pipe Breaks Loss of All AC Power (Station Blackout) Loss of RHR During Mid-Loop Operations The ERCW System is required to mitigate the consequences of plant Design Basis Events described in WB-DC-40-64. It performs a Primary Safety Function by providing cooling and makeup for essential safety-related plant equipment and components in response to adverse plant operating conditions which impose safety-related performance requirements on the systems being served. <u>EVENTS IN WB-DC-40-64 THAT CREDIT ERCW</u> Fire Design Basis Flood Operating Basis Earthquake Safe Shutdown Earthquake Tornado Combustible Gases Inside Containment Control Room Evacuation Internally Generated Missiles General High Energy Line Break Heavy Load Drop Small Break LOCA Large Break LOCA Steam Generator Tube Rupture Rupture of a Control Rod Drive Mechanism Housing Waste Gas Decay Tank Rupture Fuel Handling Accident Loss of External Electrical Load and/or Turbine Trip Loss of Offsite Power Main Steam Line Break

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		Main Feedwater Line Rupture Event Accidental Depressurization of Main Steam System Loss of Normal Feedwater Excess Heat Removal Due to Feedwater System Malfunction Moderate Energy Line Break Partial Loss of Forced Reactor Coolant Flow Single Reactor Coolant Pump Locked Rotor or Shaft Break Complete Loss of Forced Reactor Coolant Flow Excessive Load Increase Incident Accidental Depressurization of the Reactor Coolant System Inadvertent Safety Injection Operation - Power Operation Uncontrolled RCCA Bank Withdrawal From a Subcritical or Hot Zero Power Condition Uncontrolled RCCA Bank Withdrawal at Power Single RCCA Withdrawal at Full Power RCCA Misalignment Uncontrolled Boron Dilution Improper Fuel Assembly Loading Anticipated Transient Without Scram Failure of Nonsafety-Related Control Systems as an Initiating Event Minor Secondary System Pipe Breaks Loss of All AC Power (Station Blackout) Loss of RHR During Mid-Loop Operations
10.	BOP – 10 Follow-up to NRC acceptance review Question 3 – letter date July 14, 2015. The summary of that question was related to maintaining in Mode 3 or Mode 4 with decay heat being removed through the steam generators for at least 48 hours as one of the options for managing a unit shutdown and for TVA to address the use of available and approved clean water sources, ERCW, and the CST in accordance with the approved licensing basis. From the TVA response:	a. ERCW is the safety-related source of water to AFW. Whenever the steam generators are relied on for heat removal, the switchover from the CST to ERCW is required to be operable. b. TVA considers the current Applicability of LCO 3.3.2, Table 3.3.2-1, Item 6.f, Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low, appropriate as written. TVA will develop a TRM to control this function while a change is being evaluated for a generic approach with PWROG. Procedure guidance will be provided in the cooldown procedure (GO-6). Draft TRM has been posted to SharePoint site.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	"The safety-related water supply for AFW is ERCW. The AFW suction source automatically switches from the CST to ERCW when a low pressure condition exists in the AFW pump suction piping from the CST. The switchover to ERCW will occur whenever AFW is in service to assure heat removal through the steam generators if the low pressure condition exists. This assures the safety function of decay heat removal is accomplished." TS Table 3.3.2-1, Engineered Safety Feature Actuation System Instrumentation. Item 6.f describes the auxiliary feedwater pump suction transfer on suction low. The applicable Mode for this item is Modes 1, 2, 3. TS 3.7.5, Auxiliary feedwater system, applicable is Modes 1, 2, and 3 plus Mode 4 when steam generator is relied upon for heat removal. TS 3.7.6 Condensate storage tank, applicability is Modes 1, 2, and 3 plus Mode 4 when steam generator is relied upon for heat removal. BTS 3.7.6 states that as the preferred water source to satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat for 2 hours following a reactor trip from 100.6% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, or before isolating	

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	AFW to a broken line. The CST level required is equivalent to a usable volume of 200,000 gallons, which is based on holding the unit in MODE 3 for 2 hours, followed by a cooldown to RHR entry conditions at 50 F/hour. This basis is established in Reference 4 and exceeds the volume required by the accident analysis. UFSAR 9.2.6.3, Safety Evaluation, states that the condensate storage tanks are the preferred source of clean water supply for the auxiliary feedwater pumps and a storage reservoir for secondary system water. The tanks are not an engineered safety feature. The engineered safety feature water source for the auxiliary feedwater system is the ERCW system (Safety Class 2b). Either tank is isolable, but auxiliary feedwater can be obtained from both tanks. The ERCW system pool quality feedwater will be used during an extreme emergency when safety is the prime consideration and steam generator cleanliness is of secondary importance. NRC Questions: a. Since the Unit 1 CST water is limited in volume to support operations beyond 7 hours and the proposed changes state that AFW operations is now needed for operations out to 72 hours, describe the bases for the CST to ERCW automatic switchover to support Mode 4 operations. b. Based on the response to part a, describe the necessary changes to the UFSAR, TS, and TS Bases. <u>Date Posted</u>: 08/03/15	

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
11.	BOP – 11 Follow-up to NRC acceptance review Question 3 – letter date July 14, 2015. The summary of that question was related to maintaining in Mode 3 or Mode 4 with decay heat being removed through the steam generators for at least 48 hours as one of the options for managing a unit shutdown and for TVA to address the use of available and approved clean water sources, ERCW, and the CST in accordance with the approved licensing basis. From the TVA response: "There is adequate clean water to support a unit being maintained on AFW for 48 hours. The capacity of each of the two CSTs is 395,000 gallons and the normal maximum volume in the CSTs is approximately 385,000 gallons. Review of operational data for the past five years shows that the WBN Unit 1 CST has been maintained at approximately 330,000 gallons. Because AFW is not the only system that uses CST water, a standpipe is provided in the tank to assure that a minimum of 200,000 gallons of water is available for the sole use of AFW. Thus, the site maintains approximately 130,000 gallons of water in the CST above the TS limit. Normal make-up to the CST comes from the Demineralized Water Storage (DWST) Tank and the Make-up Water Treatment Plant (MWTP). The DWST tank has a capacity of 500,000 gallons and the level has historically been maintained between 65 and 90 percent full. There have been instances, including one earlier this year, where WBN Unit 1 was maintained in Mode 3 for more than two days using the DWST and the MWTP.	*a.* The Condensate Storage Tank (CST) is sized to provide seven hours of clean water (200,000 gallons). This assumes maintaining operation in Mode 3 for two hours and then cooling down for five hours at 50 degrees per hour. At seven hours post trip, the required amount for AFW flow is 175 gpm. An extrapolation of this amount for the remaining 41 hours would result in the conclusion that approximately 430,500 gallons of clean water would be required to feed AFW. This estimate is very conservative in that maintaining a stable level requires much less water than cooling down and also in the fact that this level does not account for the slow decay of AFW flow required over the period. As an example, at the 48 hour point, only approximately 110 gpm is required to satisfy the AFW demand. b. Normal makeup to the CST is provided from vendor operated equipment. Clean water is produced and added to the Demineralized Water Storage Tank (DWST). The capacity of the normal makeup system is sufficient to maintain CST inventory for extended continued operation in Mode 3 or 4. As an actual example, in 2014, WBN was maintained in Mode 3 for the period from July 13 at 1937 to July 15 at 0503. During the first 27 hours, the normal makeup system to the CST maintained CST level at the normal level of 325,000 gallons while supplying the required AFW flow in Mode 3. The level did decrease during the final five hours of Mode 3 operation, but this decrease was due to increased water usage associated with feed and bleed on the condensate system to establish secondary parameters to support plant startup. Thus, for a normal extended operation in Mode 4, such as one in which the plant cannot continue into Mode 5 due to inability to establish the required CCS and ERCW alignments to support entry into Mode 5, the normal DWST makeup to the CST will be used to replenish the CST inventory with clean water. Should augmentation to the normal DWST makeup method be required, another historical example is provided. On February 23, 2012 at 0235, it was discovered that the normal makeup source was lost, forcing the site to bring in portable equipment to replenish the CSTs. The equipment was in place and operating on the same day at 2105 or approximately 18.5 hours following the loss. The portable trailers are capable of providing 200 gpm to the CST. The procedure 0-SOI-59.01 Section 8.4, provides

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	"WBN recently added the Auxiliary Feedwater Storage Tank (AFWST) as part of the FLEX mitigating strategies. This tank has a capacity of 500,000 gallons and is an immediately available source of clean water. The tank was designed to be seismically robust and to withstand the effects of tornados. The AFWST supply piping is normally isolated by air operated valves (AOVs) from the Unit 1 and Unit 2 condensate piping that supply the suction for the AFW pumps. The AOVs open on a low pressure signal from the upstream condensate piping, a loss of AC power, or a loss of control air. Water can be transferred from the DWST to the AFWST using hoses and pumps that are maintained by the FLEX program if power cannot be provided to the DWST booster pumps. The two CSTs have a cross tie that when opened provides an additional approximately 330,000 gallons of clean water for the case of a LOCA on one unit and the other unit being shutdown." NRC Questions: a. Provide the total expected water volumes of clean water to support operations of the AFW system for the 48 hour durations. b. Provide the procedure steps (alarm responses, AOPs, EOPs, etc) that direct operators to supplement the Unit 1 CST clean water supply from the new Flex tank, opposite unit's CST and DWST (in front of the ERCW automatic switchover). c. Provide access to the design change package for the addition of the new Flex tank. This should include the 50.59	guidance for placing portable trailers in-service. In all cases, the following guidance is provided in the Annunciator Response Instructions (ARI). The low level alarm for the CSTs comes in at 210,000 gallons. The response to this alarm is contained in procedure 1/2-ARI-36-42. Below is the wording associated with low level response for Unit 1. The Unit 2 instruction is identical with the exception of switching the designated tanks (A for B and B for A). [3] **IF** level is low, **THEN, REFER TO** Tech Specs (LCO 3.7.6), and INITIATE makeup to CST A from one of the following sources as listed in preferred order: [3.1] From CST B per SOI-2&3.01, *CONDENSATE AND FEEDWATER SYSTEM* [3.2] From DI Water Storage Tank per SOI-59.01, *DEMINERALIZED WATER SYSTEM.* Makeup from the opposite unit CST is specified as the first response, as the normal makeup to the CST is from the Demin Water System. Receipt of this alarm will not normally be expected if the normal method remains available as the capacity of the Demin makeup system exceeds the requirements for AFW flow. The CST LoLo level alarm comes in at approximately 11,600 gallons. The actions specified for this alarm are. [1] **REDUCE** demand from CST, if possible. [2] **MAKEUP** to CST A at maximum possible rate. [3] **IF** AFW Pumps are running, **THEN** **MONITOR** the following: • AFW Storage Tank (AFWST) level decrease. • AFW Pump Suction Valves for swap to ERCW Discharge Header suction. [4] **REFER TO** Tech Specs (LCO 3.7.6). These actions acknowledge that if continued AFW demand is required and adequate clean water makeup can not be established, the crew should ensure that the safety related supply to AFW (ERCW) properly aligns to supply AFW needs when required. e. The following changes will be addressed in the FSAR.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	reviews performed, P&ID drawings, piping isometrics, piping safety classification and new AOVs logic. d. Describe how the new Flex tank automatic switchover and ERCW automatic switchover interact. Specifically, the set-point differences should be described. e. Based on the response above, describe any necessary changes to the UFSAR, TS, and TS Bases based on this new required water volume. Date Posted: 08/03/15	**9.2.6.1 Design Bases** The condensate storage facilities are designed to serve as a receiver of water from the main condenser high level dump and to provide treated water for makeup to the main condenser while reserving a minimum amount for the auxiliary feedwater system. This amount is required to hold the plant for two hours after a Design Basis Event (DBE) and 5 hours to cool RCS from no-load hot standby at 50°F per hour to the point at which the residual heat removal system can take over. When the CSTs are intact and offsite power is available, the inventory available in the CSTs plus makeup from the make-up water plant and the demineralized water storage tank (FSAR Section 10.4), is capable of supplying clean water to support maintaining the plant on auxiliary feedwater for longer than seven hours without the need to transfer the AFW pump suction to ERCW. No credit is taken for this additional water in the design and safety evaluations of condensate storage or AFW. The condensate storage tanks are not an engineered safety feature and are not seismically qualified. The supply from the make-up water tank and the demineralized water storage tank and associated piping are not engineered safety features and are not seismically qualified. The storage tanks supply the preferred source of water to the auxiliary feedwater system, but the engineered safety feature source is the ERCW System (Safety Class 2b). **9.2.6.3 Safety Evaluation** The condensate storage tanks are the preferred source of clean water supply for the auxiliary feedwater pumps and a storage reservoir for secondary system water. The tanks are not an engineered safety feature. The engineered safety feature water source for the auxiliary feedwater system is the ERCW system (Safety Class 2b). Either tank is isolable, but auxiliary feedwater for either unit can be obtained from both tanks. This will be done only if necessary since each condensate storage tank normally contains auxiliary feedwater for just one unit. The ERCW system pool quality feedwater will be used during events when safety is the prime consideration and steam generator cleanliness is of secondary importance. Piping connected to the condensate storage tanks is routed through a heated tunnel

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		under the tanks. Ice formation in the tanks during a period of prolonged low temperatures can be prevented, if necessary, by recirculation of water through the condensate transfer pump. The tank and its connecting piping can accommodate water whose temperature is in the range of 40°F to 130°F. **10.4.9.2 System Description** The two reactor units have separate AFW systems, as shown in Figure 10.4-21. Each system has two electric motor-driven pumps and one turbine-driven pump. Each of the electric pumps serves two steam generators; the turbine pump serves all four. All three pumps supporting a unit automatically deliver rated flow within one minute upon a trip of both turbine-driven main feedwater pumps, loss of offsite power, an AMSAC signal, a safety injection signal or low-low steam generator water level. The motor driven pumps (MDPs) start on a two-out-of-three low-low level signal in any steam generator and the turbine driven pump starts on a two-out-of-three low-low level signal in any two steam generators. Each pump supplies sufficient water for evaporative heat removal to prevent operation of the primary system relief valves or the uncovering of the core. The operator has the capability to open an additional recirculation line on the MDPs when there is low decay heat required to be removed from the SG. These lines contain a normally closed valve that closes on an accident signal. The valve is operable after the accident signal, but if an additional accident signal occurs, the valve would be reclosed. This ensures that the forward flow requirements to remove decay heat have been satisfied. Significant pump design parameters are given in Table 10.4-1. The preferred sources of water for all auxiliary feedwater pumps are the two 395,000 gallon condensate storage tanks. A minimum of 200,000 gallons in each tank is reserved for the AFW Systems by means of a standpipe through which other systems are supplied. The two CSTs are normally isolated from each other, with one CST dedicated to each unit. The AFW safety analyses take no credit for the ability to crosstie the CSTs. As an unlimited backup water supply for each unit, a separate ERCW system header feeds each motor-driven pump. The turbine-driven pump can receive backup water from either ERCW header. The ERCW supply is automatically (or remote-manually) initiated on a two-out-of-three low pressure signal in the AFW system suction lines. Pump protection during the automatic transfer to the ERCW supplies is assured by providing sufficient suction head and flow to the pumps and is

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td></td><td>verified by system analysis. Since the ERCW system supplies poor quality water, it is not used except when the condensate supply is unavailable.

In addition, the high pressure fire protection (HPFP) system which is cross-connected to the discharge of each motor driven AFW pump can be aligned to supply unlimited raw water directly to the steam generators, in the unlikely event of a flood above plant grade. Water from the HPFP system is supplied by four high pressure, vertical turbine, motor-driven, Seismic Category I pumps conforming to the requirements of ASME B&PV Code Section III, Class 3 with each having a rating of 1590 gpm at 300 feet head. These pumps are installed in the Seismic Category I Intake Pumping Station with motors above the maximum possible flood level. Each pump is capable of supplying 100% of the auxiliary feedwater demands for both units during a flood above plant grade. The four pumps are supplied from normal and emergency power with two pumps assigned to each of the two emergency power trains. Each pair of pumps on the same power train takes suction from a common sump which receives water through a settling baffle arrangement for all normal, and flood reservoir levels.

Generated CR 1075753 to address interactions from a GDC-5 perspective for Unit 1 and Unit 2 CSTs and AFW storage tank. This CR has been posted to SharePoint.</td></tr>
<tr><td>12.</td><td>BOP-12

What controls are in place to revisit flow calculations to ensure assumed HX tube plugging and fouling factors reflect actual HX degradation or required plugging?</td><td>Tube plugging for the safety related heat exchangers are controlled by issued design output. These values may be found in the System Descriptions for System 70, Component Cooling System (5%), System 72, Containment Spray System (10%), and System 74, Residual Heat Removal System (5%).

TVA’s system descriptions are design output under the 10 CFR 50 Appendix B, Quality Assurance Program.</td></tr>
<tr><td>13.</td><td>BOP - 13

The licensee is requested to provide the following information:
With the loss of Train A:
a) What is the peak heat removal rate demand on the CCS C heat exchanger to mitigate a</td><td>a) CCS HX C peak heat removal rate for mitigation of LOCA on one unit with loss of offsite power and loss of Train A:

<table><tr><th>Item</th><th>Heat Load (Btu/hr)</th></tr><tr><td>RHR Heat Exchanger</td><td>54,800,000</td></tr></table></td></tr>
</table>

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	design basis accident (loss of coolant accident with a loss of offsite power and failure of Train A)? And what is the corresponding required ERCW flow rate and CCS flow rate to meet this peak heat removal demand? b) What is the concurrent ERCW flow rate required to be sent to the Containment Spray Heat Exchanger 1B or 2B? c) What is the concurrent required ERCW flow rate to the operating EDGs? d) What is the concurrent required ERCW flow rate to the CCS heat exchanger A or B as a heat sink for the spent fuel pools? e) What is the concurrent required ERCW flow rate to the other safety related ERCW loads for the LOCA unit?	Centrifugal Charging Pump: 66,760 RHR Pump: 100,000 Safety Injection Pump: 46,000 Radiation Monitor: 0 Containment Spray Pump: 14,746 Total: 55,027,506 Train B ERCW Flows are NOT separated by unit. The total required Train B ERCW flow to the CCS HX C for one unit in LOCA-Recirculation and the other unit in Hot Shutdown for less than 48 hours is provided below. Item / CCS Flow (gpm) / ERCW Flow to CCS HX C (gpm) RHR Heat Exchanger: 5,000 / 9,200 Centrifugal Charging Pump: 28 RHR Pump: 10 Safety Injection Pump: 15 Radiation Monitor: 6 Containment Spray Pump: 2 Total: 5,061 b) For one Containment Spray Heat Exchanger: 5,200 gpm c) For two DGs: 2,600 gpm d) CCS HX A: 1,370 gpm e) Concurrent ERCW flow rates to other safety-related loads for LOCA unit:

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td></td><td>
<table>
<tr><th>Item</th><th>ERCW Flow (gpm)</th></tr>
<tr><td>Electric Board Room Chiller (Unit Common)</td><td>300.0</td></tr>
<tr><td>Main Control Room Chiller (Unit Common)</td><td>240.0</td></tr>
<tr><td>Shutdown Board Room Chiller (Unit Common)</td><td>560.0</td></tr>
<tr><td>Auxiliary Control Air System Compressor (Unit Common)</td><td>3.5</td></tr>
<tr><td>2 X ERCW Pump Cooling (Unit Common)</td><td>12.0</td></tr>
<tr><td>2 X ERCW Pump PreLube (Unit Common)</td><td>1.6</td></tr>
<tr><td>ERCW Screen Wash (Unit Common)</td><td>10.0</td></tr>
<tr><td>2 X ERCW Strainer (Unit Common)</td><td>900.0</td></tr>
<tr><td>AFW & Boric Acid Transfer Pump Area Cooler (Unit Common)</td><td>60.0</td></tr>
<tr><td>AFW & Component Cooling System Area Cooler (Unit Common)</td><td>102.0</td></tr>
<tr><td>Emergency Gas Treatment System Room Cooler (Unit Common)</td><td>10.0</td></tr>
<tr><td>Spent Fuel Pool & Thermal Barrier Booster Pumps Area Cooler (Unit Common)</td><td>29.0</td></tr>
<tr><td>Containment Spray Pump Room Cooler</td><td>28.0</td></tr>
<tr><td>Centrifugal Charging Pump Room Cooler</td><td>25.0</td></tr>
<tr><td>Elevation 692’ Penetration Room Cooler</td><td>12.0</td></tr>
<tr><td>Elevation 713’ Penetration Room Cooler</td><td>11.0</td></tr>
<tr><td>Elevation 737’ Penetration Room Cooler</td><td>12.0</td></tr>
<tr><td>Pipe Chase Room Cooler</td><td>15.0</td></tr>
<tr><td>RHR Pump Room Cooler</td><td>19.0</td></tr>
<tr><td>Safety Injection Pump Room Cooler</td><td>22.0</td></tr>
<tr><td>Total</td><td>2,372.1</td></tr>
</table>
</td></tr>
<tr><td>14.</td><td>BOP – 13 – 1 Clarification Question

The total CCS flow is stated to be 5061 GPM to each</td><td>Spent fuel pool cooling is provided by the 1B-B CCS pump through the “A” CCS HX which is realigned later in the event. The flow for the spent fuel pool is therefore not included in the “C” CCS HX flow.</td></tr>
</table>

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td>unit in BOP-12 and BOP-13.

Is additional CCS flow to CCS HX A or B required for Spent Fuel Pool cooling and how much? What will then be the required total CCS flow from the CCS B Train when in the plant conditions described in BOP-12 and BOP-13?</td><td>TVA will submit a license amendment request to maintain Auxiliary Feedwater capability in support of TS 3.4.6 Loops Operable requirements for 7 hours.</td></tr>
<tr><td>15.</td><td>BOP – 14

The licensee is requested to provide the following information:
With a loss of Train A:
a) What is the peak heat removal rate demand on the CCS C heat exchanger to maintain a unit in Mode 4 assuming the unit achieved Mode 4 in the minimum amount of time after shutdown? And what is the corresponding ERCW flow rate and CCS flow rate to meet this peak heat removal rate assuming steam generators are not in use for heat removal?

b) What is the concurrent required ERCW flow rate to the other safety related ERCW loads for the Mode 4 unit?</td><td>a) CCS HX C peak heat removal rate for unit in Mode 4 with loss of Train A:

<table>
<tr><th>Item</th><th>Heat Load (Btu/hr)</th></tr>
<tr><td>RHR HX</td><td>89,265,200</td></tr>
<tr><td>Centrifugal Charging Pump</td><td>66,760</td></tr>
<tr><td>RHR Pump</td><td>100,000</td></tr>
<tr><td>Safety Injection Pump</td><td>0</td></tr>
<tr><td>Containment Spray Pump</td><td>0</td></tr>
<tr><td>Radiation Monitor</td><td>0</td></tr>
<tr><td>Total</td><td>89,431,960</td></tr>
</table>

ERCW flow rate and CCS flow rate to meet peak heat removal rate assuming steam generators are not in use for heat removal:

Note: Even though certain non-accident unit pumps may not be running (no heat load), they may still receive CCS cooling water flow.

<table>
<tr><th>Item</th><th>CCS Flow (gpm)</th><th>ERCW Flow to CCS HX C (gpm)</th></tr>
<tr><td>RHR Heat Exchanger</td><td>5,000</td><td rowspan="5">9,200</td></tr>
<tr><td>Centrifugal Charging Pump</td><td>28</td></tr>
<tr><td>RHR Pump</td><td>10</td></tr>
<tr><td>Safety Injection Pump</td><td>15</td></tr>
<tr><td>Radiation Monitor</td><td>6</td></tr>
</table>
</td></tr>
</table>

<table>
<tr><th>Item No.</th><th>NRC Question/Request Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td></td><td>
<table>
<tr><td>Containment Spray Pump</td><td>2</td><td></td></tr>
<tr><td>Total</td><td>5,061</td><td></td></tr>
</table>

b) Concurrent required ERCW flow rate to the other safety related ERCW loads for the Mode 4 unit:

<table>
<tr><th>Item</th><th>ERCW Flow (gpm)</th></tr>
<tr><td>ERCW Pump Cooling (Unit Common)</td><td>6.0</td></tr>
<tr><td>ERCW Pump PreLube (Unit Common)</td><td>0.8</td></tr>
<tr><td>ERCW Screen Wash (Unit Common)</td><td>10.0</td></tr>
<tr><td>Centrifugal Charging Pump Room Cooler</td><td>25.0</td></tr>
<tr><td>Elevation 692’ Penetration Room Cooler</td><td>12.0</td></tr>
<tr><td>Elevation 713’ Penetration Room Cooler</td><td>11.0</td></tr>
<tr><td>Elevation 737’ Penetration Room Cooler</td><td>12.0</td></tr>
<tr><td>Pipe Chase Room Cooler</td><td>15.0</td></tr>
<tr><td>RHR Pump Room Cooler</td><td>19.0</td></tr>
<tr><td>2 X Upper Containment Vent Cooler (non-safety load)*</td><td>46.0</td></tr>
<tr><td>2 X Lower Containment Vent Cooler (non-safety Load)*</td><td>612.0</td></tr>
<tr><td>2 X Control Rod Drive Mechanism Cooler (non-safety load)*</td><td>248.0</td></tr>
<tr><td>2 X Reactor Coolant Pump Motor Air Cooler (non-safety load)*</td><td>220.0</td></tr>
<tr><td>Reactor Building Instrument Room Chiller (non-safety load)</td><td>30.0</td></tr>
<tr><td>Total</td><td>1,266.8</td></tr>
</table>

* These non-safety containment coolers are sized for normal power operation. During Hot Shutdown (Mode 4) and Cold Shutdown (Mode 5), their cooling loads (and flow requirements) are significantly reduced.
</td></tr>
<tr><td>16.</td><td>BOP 14 - 1
Please clarify the simultaneous validity of the following statements as submitted by TVA and respond to the following questions.

• Page E1-5 of the June 17, 2015, letter states that the RHR system is normally placed in service</td><td>TVA will submit a license amendment request to maintain Auxiliary Feedwater capability in support of TS 3.4.6 Loops Operable requirements for 7 hours.</td></tr>
</table>

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td>four hours after reactor shutdown.

• Page E1-4 of the July 14 submittal states that it will be approximately 7 hours before RHR is placed in service.

• TVA response to NRC Audit Review Questions BOP-14 states that the peak heat removal rate from the RHR heat Exchanger for a unit in Mode 4 with a loss of Train A is 89,265,200 BTU/hr.

A) Is 89,265,200 BTU/hr applicable to 4 hours or 7 hours after shutdown of the non-accident unit?

B) With RHR in service 4 hours after shutdown of the non-accident unit, how is removal of residual heat from the non-accident unit assured between hours 4 and 7 after shutdown?</td><td></td></tr>
<tr><td>17.</td><td>BOP – 15

The licensee is requested to provide the following information:
With a loss of Train B:
a) What is the peak heat removal rate demand on the CCS A or B heat exchanger to mitigate a design basis accident (loss of coolant accident with a loss of offsite power and failure of Train B)? And what is the corresponding required ERCW flow rate and CCS flow rate to meet this peak heat removal demand?

b) What is the concurrent ERCW flow rate</td><td>a) CCS HX A or B peak heat removal rate for mitigation of LOCA on one unit with loss of offsite power and loss of Train B:

<table>
<tr><th>Item</th><th>Heat Load (Btu/hr)</th></tr>
<tr><td>RHR Heat Exchanger</td><td>54,800,000</td></tr>
<tr><td>Centrifugal Charging Pump</td><td>66,760</td></tr>
<tr><td>RHR Pump</td><td>100,000</td></tr>
<tr><td>Safety Injection Pump</td><td>46,000</td></tr>
<tr><td>Containment Spray Pump</td><td>14,746</td></tr>
<tr><td>Seal Water Heat Exchanger</td><td>941,000</td></tr>
<tr><td>Non-Regenerative Letdown Heat Exchanger *</td><td>0</td></tr>
<tr><td>Sample Heat Exchanger A</td><td>0</td></tr>
</table></td></tr>
</table>

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	required to be sent to the Containment Spray Heat Exchanger 1A or 2A? c) What is the concurrent required ERCW flow rate to the operating EDGs? d) What is the concurrent required ERCW flow rate to the CCS heat exchanger A or B as a heat sink for the spent fuel pools? e) What is the concurrent required ERCW flow rate to the other safety related ERCW loads for the LOCA unit?	

Sample Heat Exchanger B	0
Sample Heat Exchanger C	0
Hot Sample Chiller Package	0
Radiation Monitor	0
Waste Gas Compressor	135,135
Total	56,103,641

* There is no heat load on the "Non-Regenerative Letdown Heat Exchanger" during LOCA conditions; however, the flow control valve fails OPEN on loss of air or loss of power so there may be up to 1,000 gpm of flow going through it.

Corresponding required ERCW flow rate and CCS flow rate to meet peak heat removal demand:

Item	CCS Flow (gpm)	ERCW (gpm)
RHR Heat Exchanger	5,000	4,000
Centrifugal Charging Pump	28	
RHR Pump	10	
Safety Injection Pump	15	
Containment Spray Pump	2	
Seal Water Heat Exchanger	200	
Non-Regenerative Letdown Heat Exchanger	1,000	
Sample Heat Exchanger A	20	
Sample Heat Exchanger B	28	
Sample Heat Exchanger C	20	
Hot Sample Chiller Package	22	
Radiation Monitor	6	
Waste Gas Compressor	50	
Spent Fuel Pool Hx**	~2,000	
Total	8,401	

* There is no heat load on the "Non-Regenerative Letdown Heat Exchanger" during LOCA conditions; however, the flow control valve fails OPEN on loss of air or loss of power so there may be up to 1,000 gpm of flow going through it.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted

** CCS Train 2A only (flow exists however it is not required)

b) For one Containment Spray Heat Exchanger: 5,200 gpm

c) For two DGs: 2,600 gpm

d) This is not separately available. The required ERCW flow is included in the non-accident CCS Heat Exchanger flow, i.e., 5,050 gpm for > 48 hours; 7,100 gpm for < 48 hours.

e) Concurrent required ERCW flow rate to the other safety related ERCW loads for the LOCA unit:

Item	ERCW Flow (gpm)
Electric Board Room Chiller (Unit Common)	300.0
Main Control Room Chiller (Unit Common)	240.0
Shutdown Board Room Chiller (Unit Common)	560.0
Auxiliary Control Air System Compressor (Unit Common)	3.5
2 X ERCW Pump Cooling (Unit Common)	12.0
2 X ERCW Pump PreLube (Unit Common)	1.6
ERCW Screen Wash (Unit Common)	10.0
2 X ERCW Strainer (Unit Common)	900.0
AFW & Boric Acid Transfer Pump Area Cooler (Unit Common)	60.0
AFW & Component Cooling System Area Cooler (Unit Common)	102.0
Emergency Gas Treatment System Room Cooler (Unit Common)	10.0
Spent Fuel Pool & Thermal Barrier Booster Pumps Area Cooler (Unit Common)	29.0
Containment Spray Pump Room Cooler	28.0
Centrifugal Charging Pump Room Cooler	25.0
Elevation 692’ Penetration Room Cooler	12.0
Elevation 713’ Penetration Room Cooler	11.0
Elevation 737’ Penetration Room Cooler	12.0
Pipe Chase Room Cooler	15.0

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td></td><td>
<table>
<tr><td>RHR Pump Room Cooler</td><td>19.0</td></tr>
<tr><td>Safety Injection Pump Room Cooler</td><td>22.0</td></tr>
<tr><td>Total</td><td>2,372.1</td></tr>
</table>
</td></tr>
<tr><td>18.</td><td>BOP – 16

The licensee is requested to provide the following information:
With a loss of Train B:
a) What is the peak heat removal rate demand on the CCS A or B heat exchanger to maintain a unit in Mode 4 assuming the unit achieved Mode 4 in the minimum amount of time after shutdown? And what is the corresponding ERCW flow rate and CCS flow rate to meet this peak heat removal rate assuming steam generators are not in use for heat removal?

b) What is the concurrent required ERCW flow rate to the other safety related ERCW loads for the Mode 4 unit?</td><td>a) CCS HX A or B peak heat removal rate to maintain one unit in Mode 4 with loss of Train B:
<table>
<tr><th>Item</th><th>Heat Load (Btu/hr)</th></tr>
<tr><td>RHR Heat Exchanger</td><td>89,265,200</td></tr>
<tr><td>Centrifugal Charging Pump</td><td>66,760</td></tr>
<tr><td>RHR Pump</td><td>100,000</td></tr>
<tr><td>Safety Injection Pump</td><td>0</td></tr>
<tr><td>Containment Spray Pump</td><td>0</td></tr>
<tr><td>Seal Water Heat Exchanger</td><td>517,000</td></tr>
<tr><td>Non-Regenerative Letdown Heat Exchanger</td><td>2,530,000</td></tr>
<tr><td>Sample Heat Exchanger A</td><td>0</td></tr>
<tr><td>Sample Heat Exchanger B</td><td>0</td></tr>
<tr><td>Sample Heat Exchanger C</td><td>0</td></tr>
<tr><td>Hot Sample Chiller Package</td><td>0</td></tr>
<tr><td>Radiation Monitor</td><td>0</td></tr>
<tr><td>Waste Gas Compressor</td><td>0</td></tr>
<tr><td>Spent Fuel Pool Heat Exchanger</td><td>21,000,000</td></tr>
<tr><td>Total</td><td>113,478,960</td></tr>
</table>
Corresponding ERCW flow rate and CCS flow rate to meet peak heat removal rate assuming steam generators are not in use for heat removal:
<table>
<tr><th>Item</th><th>CCS Flow (gpm)</th><th>ERCW Flow (gpm)</th></tr>
<tr><td>RHR Heat Exchanger</td><td>5,000</td><td rowspan="3">7,100</td></tr>
<tr><td>Centrifugal Charging Pump</td><td>28</td></tr>
<tr><td>RHR Pump</td><td>10</td></tr>
</table>
</td></tr>
</table>

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted

Safety Injection Pump	15
Containment Spray Pump	2
Seal Water Heat Exchanger	200
Non-Regenerative Letdown Heat Exchanger	1,000
Sample Heat Exchanger A	20
Sample Heat Exchanger B	28
Sample Heat Exchanger C	20
Hot Sample Chiller Package	22
Radiation Monitor	6
Waste Gas Compressor	50
Spent Fuel Pool Heat Exchanger	2,000
Total	8,401

b) Concurrent required ERCW flow rate to the other safety related ERCW loads for the Mode 4 unit:

Item	ERCW Flow (gpm)
ERCW Pump Cooling (Unit Common)	6.0
ERCW Pump PreLube (Unit Common)	0.8
ERCW Screen Wash (Unit Common)	10.0
Centrifugal Charging Pump Room Cooler	25.0
Elevation 692’ Penetration Room Cooler	12.0
Elevation 713’ Penetration Room Cooler	11.0
Elevation 737’ Penetration Room Cooler	10.0
Pipe Chase Room Cooler	15.0
RHR Pump Room Cooler	19.0
2 X Upper Containment Vent Cooler (non-safety load)*	46.0
2 X Lower Containment Vent Cooler (non-safety Load)*	612.0
2 X Control Rod Drive Mechanism Cooler (non-safety load)*	248.0
2 X Reactor Coolant Pump Motor Air Cooler (non-safety load)*	220.0
Reactor Building Instrument Room Chiller (non-safety load)	30.0
Total	1,264.8

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td></td><td>* These non-safety containment coolers are sized for normal power operation. During Hot Shutdown (Mode 4) and Cold Shutdown (Mode 5), their cooling loads (and flow requirements) are significantly reduced.</td></tr>
<tr><td>19.</td><td>BOP – 17

The licensee has stated on page E1-10 of the June 17, 2015, submittal:

“once Unit 1 has been shutdown for 48 hours or more, the total ERCW heat removal and thus, flow requirements, drop below the flowrate provided by two ERCW pumps.”

Provide the reasons for the 48 hour time by identifying the heat load on the ERCW system from the shutdown unit and from the LOCA unit at time 48 hours and explain how two ERCW pumps provides adequate flow at this time.</td><td>The 48 hour time-delay was selected due to limitations in the CCS and the ERCW System to simultaneously mitigate a design basis LOCA on one unit and remove core decay heat from the non-accident unit when the non-accident unit was relying on RHR to remove core decay heat. After 48 hours, the non-accident unit core decay heat is sufficiently low (~ 56.7 MBtu/hr) that the CCS and the ERCW System can support both the accident and non-accident units with any single active failure. The actual time delays vary depending on system availability and the single active failure postulated between CCS and the ERCW System. The most limiting time delay is 48 hours due to the availability of only one CCS pump aligned to CCS HX C.</td></tr>
<tr><td>20.</td><td>BOP – 18

The licensee has stated on page E1-11 of the June 17, 2015, submittal:

“The requirement to have two ERCW pumps running on one DG is required for the scenario of a LOCA on one unit and the other unit cooled by RHR within 48 hours of shutdown. The single failure of a loss of a train of power must also occur to require two ERCW pumps to be loaded on a single DG. Other single failures including the loss of a DG or a 6.9 kV shutdown board will not require two ERCW pumps to be loaded on a single DG.”</td><td>The scenario postulates a case where a single failure does not remove a complete train of electrical power and poses the question whether a second ERCW pump would be required on a single train to supply sufficient flow. This scenario has not been explicitly analyzed. However, for this event, three ERCW pumps would be available for cooldown, with two ERCW pumps on one train and one ERCW pump on the other train. In addition, at least three CCS HXs are available with multiple CCS pumps. Qualitatively, the following table compares the two scenarios:
<table>
<tr><th>Item</th><th>3 ERCW Pumps on 1 Train</th><th>3 ERCW Pumps on 2 Trains</th></tr>
<tr><td>ERCW Pumps Available</td><td>3</td><td>3</td></tr>
<tr><td>ERCW Flow</td><td>High Flow Velocities / Resistance</td><td>Low Flow Velocities / Resistance</td></tr>
<tr><td>CCS HXs Available</td><td>1 or 2</td><td>3</td></tr>
</table></td></tr>
</table>

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td>For the above scenario with a loss of a single Train A DG or Train A ERCW pump, where one Train A ERCW pump is running and two Train B ERCW pumps are running:

Has this scenario been analyzed to demonstrate that a second ERCW pump on a DG will not be required as stated? If so, describe the assumptions and results of this analysis.</td><td>
<table>
<tr><td>CCS Pump Available</td><td>3 maximum</td><td>3 or 4</td></tr>
<tr><td>CCS Flow</td><td>High Flow Velocities for Train B</td><td>Lower Velocity for Train B</td></tr>
<tr><td>RHR HXs Available</td><td>2 maximum</td><td>3 or 4</td></tr>
<tr><td>SFP HXs Available</td><td>0 for Train B</td><td>2</td></tr>
</table>
By comparison, it can be seen that the failure of one DG or one ERCW pump results in more equipment available than in the limiting case of a loss of an entire train of 6.9 kV shutdown boards. Therefore, conditions are more favorable for cooldown.
</td></tr>
<tr><td>21.</td><td>BOP – 19

The licensee has stated on page E1-4 of the July 14, 2015 letter:

“When the assumptions include a loss of offsite power and the loss of Train A power, two CCS pumps need to be aligned to the CCS Train B header and in operation when RHR is in service on both units and both units have been shutdown for less than 48 hours.

TVA agrees that the submittal did not provide much discussion of the non-accident case because the LOCA plus shutdown case is more limiting. There is discussion on pages E1-10 and E1-11 of the license amendment with respect to required ERCW pumps.”

The licensee has stated, in effect, if one unit has been shutdown for 48 hours or greater and the other unit has reached Mode 4 at the earliest opportunity, then TS 3.7.7 and TS 3.7.8 are sufficient without proposed TS 3.7.16 and TS 3.7.17.</td><td>The adequacy of two ERCW pumps per train and one CCS pump per train to removal all decay heat from both units for the scenarios provided by the NRC cannot be assured under all worst case conditions. Therefore, the Applicability Note of TS 3.7.16 and TS 3.7.17 will be revised by the removal of part b regarding the condition of the opposite unit. The proposed changes to TS 3.7.16 and TS 3.7.17 will be provided in a separate submittal.

In light of this information and the removal of TS 3.7.16 and TS 3.7.17 Applicability Note b, the third paragraph of the response to NRC Acceptance Review Question 2 provided in TVA letter to NRC, dated July 14, 2015, page E1-4, is revised as follows:

“When the assumptions include a loss of offsite power and the loss of Train A power, two CCS pumps need to be aligned to the CCS Train B header and in operation when RHR is in service on both units and ~~both units have~~ either unit has been shutdown for less than 48 hours.”</td></tr>
</table>

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td>Accordingly, the staff requests TVA to address the heat removal requirements for Unit 2 shutdown at 48 hours and Unit 1 having just reached Mode 4 (approximately 7 hours after shutdown) and the following scenarios:

1) DBA Unit 2 (LOCA/LOOP loss of Train A or B) Are the one CCS pump and 2 ERCW pumps for Train A or Train B sufficient to mitigate the DBA and maintain cool down on Unit1? Identify associated heat loads and CCS and ERCW flowrates.

2) DBA (LOOP/Loss of Train A or B) Are the one CCS pump and 2 ERCW pumps for Train A or Train B sufficient to maintain cool down on both units? Identify associated heat loads and CCS and ERCW flowrates.</td><td></td></tr>
<tr><td>22.</td><td>BOP – 20

The response to BOP-13 part (a) listed the heat load on the RHR heat exchanger as 89,265,200 Btu/hr with a CCS flow rate of 5000 gpm. FSAR Table 5.5-8 (RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DATA) lists the design heat removal capacity of the RHR heat exchanger as 37,400,000 Btu/hr with a CCS design flow of approximately 5000 GPM.

Explain how the RHR HX will transfer 89,265,200 Btu/hr as listed in BOP-13, including inlet and outlet temperatures on the shell and tubes sides and CCS and RC flow rates.</td><td>The values in FSAR Table 5.5-8 represent the “design point” of the RHR Heat Exchangers. This is just one set of conditions under which the RHR Heat Exchangers can operate. With one (1) RHR Heat Exchanger removing a core decay heat of ~ 89.4 MBtu/hr, one set of flow and temperature conditions for Loss of Train A and Loss of Train B, each, are as follows:
<table>
<tr><th>Parameter</th><th>Loss of Train A EPMJN010890, Table C7.7.89</th><th>Loss of Train B EPMJN010890, Table C7.5.9</th></tr>
<tr><td>CCS Inlet Temperature (°F)</td><td>108</td><td>105</td></tr>
<tr><td>CCS Outlet Temperature (°F)</td><td>144</td><td>141</td></tr>
<tr><td>CCS Flow (Mlb_m/hr)</td><td>2.48</td><td>2.48</td></tr>
<tr><td>CCS Flow (gpm)</td><td>5,000</td><td>5,000</td></tr>
<tr><td>RHR Inlet Temperature (°F)</td><td>239</td><td>247</td></tr>
</table></td></tr>
</table>

<table>
<tr><th>Item No.</th><th>NRC Question/Request
Date Posted</th><th>TVA Response/Dated Posted</th></tr>
<tr><td></td><td></td><td>
<table>
<tr><td>RHR Outlet Temperature (°F)</td><td>153</td><td>148</td></tr>
<tr><td>RHR Flow (Mlb_m/hr)</td><td>1.05</td><td>0.90</td></tr>
<tr><td>RHR Flow (gpm)</td><td>2,146</td><td>1,828</td></tr>
<tr><td>Core Decay Heat (MBtu/hr)</td><td>89.4</td><td>89.4</td></tr>
<tr><td>UA (MBtu/(hr - °F))</td><td>1.48</td><td>1.43</td></tr>
</table>
During actual plant operations, the CCS flow is set to approximately 5,000 gpm and the operators control the rate of cooldown and CCS outlet temperature from the RHR Heat Exchanger by manually throttling the RHR flow rate. All parameters (temperature, pressure, flow rate) are within the RHR Heat Exchanger design conditions as shown on the Vendor datasheets.</td></tr>
<tr><td>23.</td><td>BOP – 21

TVA Calculation EPMJN010890 Revision 19, “Performance of CCS Heat Exchanger” Appendix C, Table C7.7.69, and Appendix E, Tables E1 and E2.

(a) The total heat load on the CCS heat exchanger (HX) C stated in Table C7.7.69 is 144.72 MBtu/hr (sum of 89.4 MBtu/hr shutdown unit load, 54.8 MBtu/hr LOCA unit load, and 0.53 MBtu/hr miscellaneous loads) is more limiting than the total heat load of 112.03 MBtu/hr given in Table E1. Explain why the data in Table E1 is used to calculate the UA for the **virtual** CCS HXs for containment analysis and shutdown cooling analysis.

(b) In Table E2, the CCS virtual HX assigned for containment analysis has UA = 3.17 MBtu/hr°F which is consistent with the value used in Westinghouse containment analysis report in Reference 1. What is the UA for the CCS virtual HX assigned for shutdown cooling and how is it</td><td>a. Based on Westinghouse analysis sensitivity runs using a UA of 2.0, the containment pressure change is minimal (11.73 to 11.76 psig). TVA will docket the sensitivity analysis in support of the UA used in the TVA Calculation EPMJN010890 Revision 19, “Performance of CCS Heat Exchanger” Appendix C, Table C7.7.69, and Appendix E, Tables E1 and E2.

The purpose of Appendix E was to develop a representative Heat Exchanger “UA” value for use by Westinghouse in containment analyses. Since the Watts Bar “C” HX is shared between the two units, Westinghouse needed to know what the effective part of the heat exchanger was supporting the unit having the LOCA. Appendix E apportions the “C” HX “UA” value according to the percent mass flow going to the heat exchanger from each unit.

Appendix E1 was started prior to the project to add a third ERCW pump to support the Hot Shutdown / LOCA-Recirculation mode of operation within 48 hours of a unit shutdown. The flows and heat loads reflect two ERCW pumps in operation and after 48 hours from shutdown. Heat Exchanger “UA” values are a function of heat exchanger geometry and flow and are not dependent on heat duty on the device. Using the lower two ERCW pump flow of 7,125 gpm would produce lower UA values than the use of the higher 9,200 MBtu/hr flow from three ERCW pumps. Note that the LOCA containment analysis must support both two and three ERCW pumps in-service.</td></tr>
</table>

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	calculated based on the real CCS HX UA = 6.44 MBtu/hr-ºF? (c) Provide the analysis that shows that the calculated value of shutdown unit virtual HX CCS UA (based on UA = 3.17 MBtu/hr-ºF for the CCS LOCA virtual HX and 6.44 MBtu/hr-ºF for the real CCS HX) can handle the shutdown cooling load of 89.4 MBtu/hr plus 0.53 MBtu/hr miscellaneous loads. (d) Table C7.7.69 states the RHR HX heat load for the shutdown unit is 89.4 MBtu/hr. At what time during the shutdown transient does this load occur? (e) Table E1 states RHR HX heat load for shutdown unit s 56.7 MBtu/hr. At what time during the shutdown transient does this load occur? (f) Tables C7.7.69 and E1 states the RHR HX heat load for the LOCA unit is 54.8 MBtu/hr. Please confirm this is the heat load at the initiation of the RHR sprays assumed to start operating at 3600 second from the LOCA initiation. (g) At what time does the LOCA occur in relation to the initiation of the shutdown transient assumed in Tables C7.7.69 and E1? (h) Tables C7.7.69 and E1, miscellaneous heat load 0.53 MBtu/hr is for which unit. Confirm whether this is a combined miscellaneous load for both units, and if so, how much is imposed on each	During discussions with NRC reviewers, a question was posed as to the conservatism of using a mass flow based allotment of "UA." A counter proposal was made that the allotment should be made based on the heat load from each unit. The spreadsheets at the end of this enclosure provide a comparison between the two methods, using 9,200 gpm total ERCW flow. These spreadsheets will be added to Calculation EPMJN010890, Appendix E. c. Response to "a" and "b" will address "c". Results of a and b will also be provided. b. Add data sheets reflecting virtual CCS HX performance based on CCS Flow and CCS heat load. Include sensitivity analysis based on ERCW flow based on heat load. Using the larger 9,200 gpm ERCW flow the "real" HX UA is 6.82 MBtu/hr-^{0}F. With the HX apportioned by Flow, the shutdown cooling HX UA is 3.35 (the same as the LOCA unit). With the HX apportioned by Heat Load, the shutdown cooling HX UA is 4.12 MBtu/hr-^{0}F. It is calculated the same as above for the LOCA unit. c. This is demonstrated by Table C7.7.69 and is reproduced in the first table above for a total ERCW flow of 9,200 gpm. All ERCW and CCS temperatures are acceptable. Note: the miscellaneous heat load and flow gets its own "virtual HX" with UA apportioned by either Flow or Heat Load. d. 7 - hours e. 48 - hours f. In TVA's calculation, the 54.8 MBtu/hr is assumed from initiation of recirculation mode (~40 minutes assuming the loss of train event). g. For GDC-5 analysis, the LOCA occurs 7-hours after the initiation of a shutdown on the non-accident unit. h. 0.53 MBtu/hr is the combined miscellaneous heat load for two units. The miscellaneous loads are accounted for using an additional virtual heat exchanger.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	CCS virtual HX. (i) What is the maximum operating temperature of the CCS fluid under the condition of the most limiting heat load on the RHR HX C? (j) In Tables C7.7.69, E1, and E2, explain the difference between the design and actual values for “ho” and “hi” for the CCS and RHR HXs. (k) Refer to Section 6.1.4 and 6.1.5 of the calculation which states the HX tubes were changed from 90-10 copper nickel to stainless steel. Confirm that the CCS U and UA given in Tables C7.7.69 and E1 are based on the as-built HX material thermal conductivity, tube thickness, worst fouling resistance and tube plugging. (l) In Tables C7.7.69, E1, and E2, what is represented by “F”, “R”, and “S” and the HX correction factor “r”, and “p”?	i. The limiting temperature is 146°F CCS temperature at the outlet of the RHR HX (analyzed pipe stress limit). The Operating Modes calculation EPMJK022988 shows that CCS temperatures through the RHR HX should not exceed 146ºF. The stress calculation is N3-70-04A. j. See EPMJN010890, Section 7.1, Sheet 31 and 32. k. See EPMJN010890, Section 6.1 l. See EPMJN010890, Section 7.0.2, Equation 10 and Section 7.3, Sheets 41 and 42.
24.	BOP – 22 In BOP-13-16, the licensee listed the design heat loads and corresponding required flow rates for CCS and ERCW during a DBA and loss of Train A or Train B. One of the design basis conditions is a loss of downstream dam. Describe and justify how the ERCW design accounts for a loss of downstream dam during a DBA with the equipment and system lineups that are specified in both LCOs of TS 3.7.8 and proposed TS 3.7.17.	As documented in MDQ00006720080341 Appendices 11 (two ERCW Pumps) and 17 (three ERCW Pumps), MULTIFLOW demonstrates that the ERCW pumps develop sufficient head to supply all required users during loss of downstream dam conditions. Calculation EPM-WUC-072489 demonstrates that three ERCW pumps have required NPSHA for loss of downstream dam conditions.
25.	ELEC – 1-3 As documented in NUREG-0847, "Safety Evaluation	Response Summary from TVA Letter to NRC dated August 3, 2015:

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2," Supplement 22, (SSER 22) published February 2011, the licensing basis of Watts Bar Nuclear Units is: 1. Dual-unit trip as a result of an abnormal operational occurrence 2. Accident in one unit and concurrent shutdown of the second unit (with and without offsite power) 3. Accident in one unit and spurious engineered safety feature actuation in the other unit (with and without offsite power) The license amendment request (LAR) for Watts Bar Nuclear Plant (WBN) Unit 1, dated June 17, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15170A474) proposes realignment of Component Cooling System pumps and Essential Raw Cooling Water (ERCW) Pumps to support heat removal capability. The staff is also reviewing a LAR submitted by letter dated August 1, 2013 (ADAMS Accession No. ML13220A103), to modify limiting conditions for operation for Technical Specification 3.8.1, "AC Sources - Operating," for the available maintenance feeder for Common Station Service Transformers (CSSTs) A and B. The staff is requesting clarification on loading of onsite and offsite power systems and has determined that the following additional information is needed to complete the review of the LAR: 1. For the scenarios related to dual unit shutdown with offsite power system available, please	Q1 With offsite power available, there is no change to the licensing basis documented in SSER 22. Changes to the cases in the calculations for the dual unit shutdown with offsite power available were not revised in association with the June 17, 2015 LAR. The SI pump, Containment Spray pump, and AFW pump combined horsepower is approximately 1660 horsepower. This is approximately twice the horsepower requirement of an ERCW pump. Since these three pumps are not running for the GDC-5 case, there is considerable margin compared to the limiting case thus demonstrating that the start of a second ERCW pump on a non-accident shutdown board is acceptable and bounded by the LOCA/inadvertent SI case. The evaluations of CSSTs A and B are included in WBN calculation EDQ00099920070002, "AC Auxiliary Power System Analysis." TVA submitted a response to NRC Open Items from SSER 22 on April 6, 2011 for WBN Unit 2. This submittal described the margin studies done for all four CSSTs. The margin studies that TVA provided in the submittal were discussed in SSER 24 in relation to the closure of SSER Open Items 27 and 28. Because the LOCA on one unit with an inadvertent SI on the other unit results in a higher load than the scenario discussed in the LAR, additional margin studies were not required. Q2 The DG loading for the first 20 minutes is the base case loading described in the WBN Calculation EDQ00099920080014, "Diesel Generator Loading Analysis." The CCS pumps are assumed to start and run in the base case, so the proposed amendment related to CCS does not represent a change from an electrical standpoint. The loading of a second ERCW pump on an individual DG occurs no earlier than 40 minutes after DG start. This is why the base case applies for the first 20 minutes. Table 3 of the June 17, 2015 submittal represents the bounding cases after 20 minutes. The values in the table provide the horsepower assumed for each of the large motor loads for each available DG and provide the total kW loading on each available DG for the scenarios in Table 3. Attachment 1 of that letter provided excerpts from the DG loading calculation including tables that summarized the loads on each DG for a variety of cases including Loss of Offsite Power (LOOP) / LOCA and dual unit cases.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	provide a summary of the calculations performed to evaluate capability of offsite power transformers CSST A and B for the licensing basis documented in SSER 22. Provide the impact of changes proposed in the June 17, 2015, LAR on the licensing basis documented in SSER 22. 2. Table 3 in Enclosure 1 of the WBN Unit 1, LAR dated June 17, 2015, contains the Summary of Steady-State Diesel Generator (DG) Loading with 3 ERCW Pumps (>20 minutes) only. Please clarify whether the electrical system loadings considered in Table 3 of the LAR is bounding for all the scenarios without offsite power addressed in SSER 22 summarized above. For the scenarios related to shutdown using onsite power systems, please provide details (calculations or explanation) related to large motor loads (Rating and horse power value) considered for the specific scenarios. Provide details of additional kilo-Watt (kW) loading considered in the total kW loading of each DG. Also provide DG loadings during ≤ 20 minutes. 3. Please provide details on any load shedding that may be procedurally controlled to preclude overloading the power source(s). Date Posted: 07/31/15	Q3 Before a second ERCW pump can be loaded on its DG, the AFW Pump, if running, will be stopped and the main control room hand switch placed in pull-to-lock. This action assures that the AFW pump will not inadvertently start to preclude overloading the DG. TVA currently plans to include these actions in the same procedure that starts the second ERCW pump. The actions will be placed in a step that precedes the start of the second ERCW pump. This is the only load shed assumed in the DG loading analysis.
26.	ELEC - 4 Resolve the apparent discrepancy between the DG loading calculation, Appendix N for AFW load on LOCA unit and the June 17, 2015 Enclosure 1, Table 3 AFW load indicated on the LOCA unit (i.e., 600 hp versus 300 hp).	A revision to the DG loading calculation has been completed. Discrepancies between the revised calculation and Enclosure 1, Table 3 of the June 17, 2015 letter have been resolved. An updated version of the DG loading calculation has been posted on Sharepoint. An updated version of Enclosure 1, Table 3 is included at the end of this enclosure.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	Also update Enclosure 1, Table 3 to reflect other corrections made to DG loading Calculation.	
27.	ELEC - 5 Evaluate the differences in DG loading for a small break LOCA. Is the large break LOCA assumed in the GDC 5 analysis bounding?	A revision to the DG loading calculation is in progress with an outcome that is expected to be favorable. An updated version of the DG loading calculation has been posted on Sharepoint.
28.	HF - 1 Describe what cues will be provided to the operator indicating that new and changed manual actions (as described in TVA's response to NRC Acceptance Review Question 5, Item 1) are required. In your response, identify the specific plant condition, annunciator status, associated alarms, and procedure steps that will provide instructions to the operator. Further, identify information that is required to inform the operator that these manual actions have been correctly performed, and that they can be terminated. Date Posted: 07/31/15	The first cue provided to the operating staff is a procedural step in E-1, Loss of Reactor or Secondary Coolant to check the status of electrical power. Should a complete loss of either train of 6.9 kilovolt (kV) shutdown board be detected in this step, the procedure will require that additional actions be taken. The first action will be to determine if the remaining train of power is supplied from offsite or DG source. If the offsite power source is supplying the bus, the operator is directed to start an additional ERCW pump associated with the shutdown unit's 6.9 kV shutdown board that remains powered. Depending on the train of power lost, the operations staff may be required to start an additional CCS pump. Should the source of power remaining be the DGs, the operator is directed to perform actions in accordance with an appendix to the E-1 procedure. This appendix is handed off to the shutdown unit. The shutdown unit will first determine if RHR cooling is in service. If RHR cooling is NOT in service, the shutdown unit is directed to secure plant cooldown and maintain current plant temperature. An additional action is specified to perform throttling of ERCW cooling flow through the CCS heat exchanger (HX), if required due to the specific loss of power. If RHR cooling is in service, actions are contained in the Appendix to: place the motor driven AFW (MDAFW) pump hand switch in pull-to-lock (PTL) if the turbine driven AFW (TDAFW) pump is in operation, dispatch an operator to the 6.9 kV shutdown board to place the ERCW bypass switch in the bypass condition and start the ERCW pump when 20 minutes have elapsed.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		The main control room (MCR) will be alerted that the bypass switch has been placed in bypass by a MCR annunciator that alarms when the switch is placed in bypass. Should the field operator position the wrong switch, the MCR staff would become aware of this fact upon attempting to start the ERCW pump. The expected indications, breaker position lights, pump amps, discharge pressure and flow, would not be observed. This would cue the MCR staff to request that the field operator verify the correct switch had been operated. An additional cue is provided in a note contained in the main body procedure response of E-1 that the additional ERCW pump must be placed in service prior to aligning containment sump recirculation on the accident unit. Once the additional alignments are put in place by E-1, they will remain in place until a determination is made by the Emergency Response Organization (ERO) that conditions no longer require their operation. In this event, the Technical Support Center (TSC) will be staffed and the Shift Manager (SM) will transfer Site Emergency Director duties to the TSC. Since the continued need of the additional alignments will vary depending on the specific conditions at the start of the event, no attempt was made to proceduralize securing the alignments. The SM will consult with the TSC to determine that plant conditions are such that the alignments are no longer required.
29.	HF - 2 Describe how the operators of each of two units will be informed of the status of the other unit, and how their actions will be coordinated. Clarify if one of the two units will be put in lead, and what events will result in changes to the chain of command. <u>Date Posted</u>: 07/31/15	In this case, the accident unit will be considered to be in the “lead” during the event since the emergency operating procedure (EOP) for the accident unit is the driver for actions required. Upon EOP entry, the accident unit’s Unit Supervisor (US) will perform a crew update. Although this update is primarily intended to focus the attention of the particular unit’s crew, the common design of the Watts Bar Nuclear Plant (WBN) MCR allows either units operating staff to hear a crew update performed on either unit. In addition, the many alarms that are received during a LOCA will be immediately noticed by the shutdown unit, so there is no potential that the shutdown unit will not realize that conditions have degraded on the accident unit. Upon EOP entry, the standard practice at WBN is to recall the Shift Technical Advisor (STA) and the SM to the MCR, if they are not currently present. This is practiced routinely during operator requalification training by removing all persons from the simulator except for the minimum staffing requirements. A condition is then inserted on the simulator, and the operators remaining in the MCR are required to recall the rest of the staff to the

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		simulator to combat the casualty. The standard chain of command will remain in effect during this event. The SM retains overall oversight and will have overall responsibility for ensuring dual unit activities are adequately prioritized and supported. The STA provides additional oversight and backup that the accident crew is taking appropriate actions based on the plant conditions at the accident unit. The accident unit US directs the crew's response in the EOP. The shutdown unit US directs the procedural activities required for shutdown. The accident unit operating staff will conduct the electrical power monitoring activities directed in procedure E-1. When a complete loss of one train of power is detected, the US will direct the appropriate mitigating steps based on the existing conditions. The duties required of the shutdown unit will be directed by handing off the attachment of E-1 that contains the necessary actions. In this event, the communication between the units will be provided via US communication or the SM. The nature of this event is that a large number of alarms will occur, so the fact that something unusual has happened to the accident unit will be readily apparent to the shutdown unit. It is predictable that the annunciators, coupled with the observed level of activity on the accident unit, will prompt one of the shutdown units operators to be dispatched to gather information on what is happening on the accident unit. It would also be the expectation that if the shutdown units condition allows, one of the Unit Operators (UOs) from the shutdown unit would function to support mitigating activities on the accident unit.
30.	HF - 3 Describe if TVA identified any relevant, pre-existing performance issues associated with procedural guidance, training, and operator manual actions related to starting and stopping ERCW and CCW pumps and operating switches on the 6.9 kV shutdown board or issues associated with dual unit operation at other sites. <u>Date Posted</u>: 07/31/15	The Institute of Nuclear Power Operations (INPO) Operating Experience (OE) database and the TVA OE database were searched for industry events associated with ERCW, CCS and dual unit operations. The list obtained from these searches was reviewed to determine lessons that might need to be incorporated in the development of this design change. In addition, WBN benchmarked both Sequoyah and Browns Ferry for lessons learned on dual unit operations. This benchmarking ultimately resulted in the generic post-accident response that was developed in coordinating the WBN actions for this postulated event. The most common identified OE associated with dual unit operation of equipment involved plant transients due to operation of the wrong unit component. In this case, the changes

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		at WBN do not fit this common error mode, in that the ERCW is common unit equipment, so that there is no potential for operating the correct component on the wrong unit. The method chosen for alignment of the second train B CCS pump (normally from the shutdown unit and operated by the shutdown unit's staff) also provides protection from wrong unit equipment concerns. The CCS pump will only be started on a LOCA accompanied with a loss of train of electrical power. In this case, the B train CCS pump for the accident unit will automatically start, so the chances that the operating unit staff will attempt to manipulate this component are nonexistent. Another common identified OE involved misoperation of the expected component. This condition is precluded by the WBN design in that the MCR staff will be alerted that the interlock bypass switch has been placed in the correct position by MCR annunciation.
31.	HF - 4 Describe any changes that were required of the Control Room task analysis that was done as part of TVA's Detailed Control Room Design Review. If no update to the task analysis was necessary, describe how task requirements were developed for the identified new and changed operator actions. Describe what reasonable or credible potential errors associated with the new and changed operator actions were identified during task analysis. <u>Date Posted</u>: 07/31/15	During development of this design change, operations, training and engineering personnel worked to determine the final product that would be installed in the plant. Operations is a quorum member of all Design Change Notice (DCN) meetings in order to ensure that operations provides input into all design changes implemented in the plant. No new switches or components were added in the MCR for this issue. Components operated in the MCR are: 1 CCS pump, 1 ERCW pump, 1 AFW handswitch and ERCW flow control valves (FCVs) for CCS HX. These components are already routinely operated by the MCR staff for multiple normal alignments and casualty situations. A new annunciator designed consistent with current design requirements, MCR standards, and NUREG-0700 requirements has been added to the MCR. The WBN Annunciator Response Instructions contain the response to this annunciator, consistent with the method contained in the existing annunciator response guidance. An additional bypass switch has been added to the 6.9 kV shutdown boards to allow starting the third ERCW pump if DG power is all that is available. If required, a Nuclear Assistant Unit Operator (NAUO) will be dispatched to the area to operate the switch. The switch itself is new, but the task of aligning electrical board transfer switches is one that is routinely performed by the NAUOs. The new switches most closely resemble the Appendix R transfer switches that are located on multiple electrical components that are currently operated by NAUOs when required.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		The interlock bypass switches are labeled consistent with other plant equipment as required by TI-12.14, "Replacement and Upgrade of Plant Component Identification Tagging and Labeling." The positions of the switch are clearly discernable. Attached to this response are several pictures of the new switches to convey location and labeling that is installed in the plant. When installed, the switches for each electrical board will be checked and independently verified to be in the correct position at least once per month. The switches will be added to the DG standby checklist which is performed after each monthly DG surveillance or for any evolution that has removed the DG from service. This will require 8 NAUOs to view the switches and their positions locally each month. This frequent check, coupled with standard labeling and similarity of this task to those already performed by the NAUOs, gives the station great confidence that switch operation can be successfully performed by the NAUOs.
32.	HF - 5 Describe any increase in operator workload that will occur with the proposed license amendment Date Posted: 07/31/15	The increase in workload due to this license amendment is considered well within the existing capabilities of the current required minimum staffing complement. The actions added are: monitoring status of electrical power and taking the appropriate actions to start an ERCW/CCS pump. If offsite power is supplied to the remaining electrical power train, additional actions are limited to starting the ERCW pump, and potentially the B train CCS pump. An additional potential action is throttling CCS HX flows if required. These activities can be performed by one individual in a 1 to 2 minute timeframe. If DG power is the only source to the remaining electrical train, actions in addition to the ones above are required. These actions will have minimum impact on the accident unit operating staff as the Attachment directing these actions will be handed off to the shutdown unit to complete. The actions of the attachment to prepare to start an additional ERCW pump will require dispatching an NAUO to the 6.9 kV shutdown board to place the bypass switch in the bypass position. The switch positioning and access to the board room area will require approximately 3 minutes and 35 seconds as outlined in the previously submitted dose assessment. The shutdown unit will then start an additional ERCW pump after 40 minutes has elapsed.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
33.	HF - 6 Describe the process used to design the ERCW pump interlock bypass switch. In your response, explain if the switch complies with control room standards and the applicable guidance of NUREG-0700, "Human-System Interface Design Review Guidelines." Further, describe how the bypassed and non-bypassed states are labeled, and whether they are augmented with status lights showing actual valve position Date Posted: 07/31/15	NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," outlines the process in which the bypass switches were designed and installed. In this particular instance, DCN 53785 is being utilized to install the ERCW pump interlock bypass switches. The bypass switches are not augmented with status lights. As shown on the pictures at the end of this enclosure, the positions of bypass and normal are clearly labeled. When the switch is taken to bypass, an alarm is received in the MCR. Thus, when the MCR operators dispatch the NAUO to position this switch, the alarm that is received will inform them that the action is complete prior to the communication from the NAUO. Should the NAUO fail to position the switch correctly, the lack of alarm will provide the MCR staff with opportunity to identify the error. Should the NAUO position the wrong switch, the ERCW pump will fail to start which will alert the operators that verification is needed to ensure that previous actions were performed correctly. The switches conform to the requirements for local workstation controls outlined in NUREG-0700. Labeling of the switches is in accordance with labeling requirements of TI-12.14, "Replacement and Upgrade of Plant Component Identification Tagging and Labeling." This labeling is consistent with other components that the operating staff manipulates during routine evolutions. Standard abbreviations are used on the switch, are easily recognizable to the station staff and are defined in 0-TI-12.13, "Acronyms/Abbreviations Listing for Labeling." The positions required to be selected are clearly marked and align with the instruction that is outlined in procedures. A picture of the switch and labeling provided at the end of this enclosure.
34.	HF - 7 TVA's response to NRC Acceptance Review Question 5, Item 2, paragraph b states, in part: "The final decision about what procedures will be affected by this license amendment request is part of the impact review that occurs once the submittal is approved." NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, Section 3.8, "Procedure Design," states, in part: "The objective of	The information contained in this response supersedes the information originally provided to the NRC concerning the procedures that will be developed for this event. TVA's initial response indicated that E-0 and ES-1.3 would contain the guidance that would be implemented post LOCA. After initial drafts were reviewed, it was determined that this guidance more appropriately belonged in the EOP associated with a LOCA (E-1). A list of procedures that will ultimately be modified based on the changes needed to implement the requirements is included near the end of this enclosure. The previous TVA statement regarding "the final decision about what procedures will be affected by this license amendment request is part of the impact review that occurs once the submittal is approved" requires clarification. The procedures will not actually be implemented until the license amendment request (LAR) process is complete. However, part of the LAR

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	this review is to verify that applicable plant procedures have been appropriately modified, where needed, to provide adequate guidance for the successful completion of the [Human Actions] HAs, and that the procedures adequately reflect changes in plant equipment and HAs." Identify any new, revised, or deleted procedures required to support the proposed LAR not previously identified in docketed submittals. Provide procedure number, revision, title, and a summary of the actions changed, added, or deleted. Date Posted: 07/31/15	process and the accompanying DCN process require that impacts be identified in advance of the LAR or DCN. Therefore, at this time, TVA has a good understanding of the scope of the procedure revisions required. Drafts of the proposed changes have been distributed for review to the appropriate personnel. The only procedure that has been changed at this time is 0-SI-82.02, "Diesel Generator (DG) 1B-B System Operating Instruction." Revision 005 of this instruction was issued to ensure that the bypass switch on the 6.9 kV shutdown board is checked during the performance of the DG standby checklist. This change has been field verified to ensure the guidance in the procedure is consistent with the equipment information in the field. The remaining procedures needed to implement this change will be issued as the implementing process requires. For the bypass switches, this will occur on return to operation of the DCN paperwork following the work to install the switches. For the remainder, it will follow LAR approval. The GO and EOP procedures will be changed as required to support the issuance of Unit 2 EOPs. This is currently procedurally defined as sometime between completion of Hot Functional Testing and fuel load. Although an actual date can not be identified, the entire Unit 2 EOP network will be required to be in place prior to operation of Unit 2. The guidance needed to realign lineups required in the CCS and ERCW systems is currently in place. Therefore, no future change to these procedures is required to implement these actions. Although it is possible that validations of the procedures remaining to be issued might identify a note or other informational enhancements that might be required, they would not constitute a condition that would prevent issuing the revised procedures.
35.	HF - 8 Describe how the operators access and use Technical Specifications (e.g., if they are available electronically, as a hard copy, etc.). Describe the interface between the Emergency Operating procedures and Technical Specifications, specifically, if the procedures refer the operator to the Technical	Two hard copies of each units TS are available in the MCR. One is maintained at the US station and the other is at the SM desk. TS are also available electronically via the WBN electronic document storage and retrieval system known as the Business Support Library (BSL). Access to BSL for documents is routinely performed by the operating staff to print copies for performing surveillance instructions and to verify that the hard copies used are the current revision of the procedure. The Abnormal Operating Instructions (AOI) identify the potential TS that could be

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	Specifications Date Posted: 07/31/15	impacted by the particular event in a list format. When entry into the AOI procedures occurs, this list and TS will be referenced to identify any TS limitations or required actions. This is particularly important in the AOI network, as these are the instructions that are used to combat emergencies in a condition where the plant may maintain operation in Modes 1 and 2. The emergency operating instructions (EOI) do not contain direct reference to TS information. This network will be utilized following a reactor trip and presuppose a condition adverse to TS has occurred such as loss of primary pressure boundary, complete loss of AC power, steam generator (SG) tube leak, or steam line rupture for which the potential TS limitations or required actions are secondary to ensuring that immediate actions are taken to place or restore the plant to a stable condition. In the case of entry into the EOI network, conforming to the Westinghouse developed Emergency Response Guidelines (ERG) ensures the plant is maintained in the safest condition for the event. It would then become the responsibility of the operating staff and the emergency response organization to identify potential impacts to TS equipment that may influence future actions that are the result of the plant condition.
36.	HF - 9 Describe the plans and schedules for revising the training program, to reflect the changes in the proposed license amendment. Clarify if training will be provided prior to implementation of the proposed changes. Date Posted: 07/31/15	Training representatives are part of the team that reviews DCN and LAR impacts. In addition, one of the responsibilities of the Operations member reviewing impacts is to make a determination if training is impacted and to inform the appropriate training management representative if training is required. The training associated with these changes will be performed in multiple formats. Relevant DCN impacts to the plant are included each licensed operator requalification (LOR) cycle in the “Plant Changes” portion. The Plant Changes training covers DCN changes, procedure changes, relevant industry events and significant corrective action events. The training changes required for this change are tracked in the corrective action program. The training needs analysis has been performed. Additions to the Changes Lesson Plan and the lesson plans associated with the ERCW System and CCS will be complete for the next scheduled LOR training cycle. All training will be complete prior to the implementation of the license amendment request.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
37.	HF - 10 Describe the process used to verify and validate the ability of TVA's operators to accomplish tasks required for the proposed license amendment. In lieu of a description, relevant administrative procedure(s) for verification and validation or a verification and validation plan for the proposed change (if developed) may be provided. In your response, clarify if the validation will include a representative sample of operators and whether it will be performed with Technical Specification minimum staffing and nominal staffing. <u>Date Posted</u>: 07/31/15	NPG-SPP-01.2.1, "Interim Administration of Site Technical Procedures for Watts Bar 1 and 2," contains the requirements that must be met when developing or revising technical procedures. This would apply to the SOIs and GOs associated with the change. As defined in the procedure review verification checklist (Attachment 3) of this procedure, various activities are performed to ensure that the procedure guidance is developed in a manner that includes consideration for human factors. The following is a list of some of the items that are required to be verified in procedure development: Does the procedure agree with and reference applicable drawings? Can the procedure be correctly performed in the designated sequence? Are equipment numbers and nomenclature used in the procedure identifiable to those displayed on the equipment? Can equipment identified in the procedure be easily located? Are the units of measurement used in the procedure to record readings the same as those displayed on the equipment? Have human factors and system interactions been properly considered? Procedures are walked down after development to ensure that the information provided in the procedure agrees with conditions in the field. Personnel who would normally perform the task are the individuals who are tasked with these walkdowns to ensure that the developed content provides the level of detail that is needed to successfully perform the evolution. For the emergency operating network, TI-12.11, "Emergency Operating Instruction (EOI) Control," contains a more specific validation process. This instruction defines what validation method should be used based on the change, the persons that should make up the validation team, how the validation is conducted and how the validation is documented. In all cases involving the EOI network, consideration is taken on whether the task can be performed by the minimum shift compliment. If it is judged that the task would interfere with the ability of minimum shift staffing requirements, then efforts are taken to either redevelop the desired process or increase the required staffing as an interim measure.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
38.	HF - 11 Describe the process used to monitor new and changed operator actions to ensure that they remain feasible and reliable over the long term, and are not degraded due to design changes, inadequate training, or other mechanisms. Date Posted: 07/31/15	TI-12.11, "Emergency Operating Instruction (EOI) Control," contains direction that is intended to address this very concern. For the current modification, the EOI network contains the only procedures that this issue might apply to. Limitations presented by having additional persons to perform the CCS and ERCW alignments prior to entering Mode 4 do not represent a safety concern. If insufficient staff is available to accomplish these alignments, then entry into Mode 4 will be prohibited. This represents a station efficiency and outage completion concern and not a safety concern. In practice, Operations crews transition to "super crew" alignment approximately two weeks prior to the start of the outage, so sufficient personnel will be available to prevent this from being a concern. The actions taken in the EOI network require an initial assessment for this very concern. In addition, any future revisions are required to "evaluate whether minimum operator staffing levels are impacted by the proposed change. Operator requalification training on EOIs provides a means of periodically verifying the technical adequacy of emergency instructions. Operators and training personnel are responsible for ensuring that problems or discrepancies discovered in EOIs during training are documented using a Condition Report or Procedure Change Request (PCR), as appropriate. Proposed enhancements and suggestions for improvement of EOIs are also encouraged. Should a future attempt be made to change the operation of the bypass switches, plant processes would identify that this would require a 10 CFR 50.59 review. This review would prevent future manipulations that would have a negative impact on maintaining plant safety.
39.	HF - 12 TVA's response to NRC Acceptance Review Questions dated July 14, 2015 (ADAMS Accession Number ML15197A357), Question 5, Item 4 states: "An interlock bypass switch for ERCW pumps will be installed as described above and in the license amendment request. An annunciator window will be	There will be a total of four bypass switches installed, one on each 6.9 kV shutdown board. Each bypass switch will bypass the interlock that prevents two ERCW Pumps from being powered from the same shutdown board. Only one annunciator will be added. The ERCW annunciator panel in the MCR is a common annunciator panel (does not have a separate panel for Unit 1 and Unit 2). Pictures of the MCR annunciator panel and the window for this alarm are included at the end of this enclosure.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	added in the main control room to show when an interlock bypass switch has been activated." Clarify how many interlock bypass switches will be installed and how many annunciator windows will be added. Further, clarify if all bypass switches will be identical in design an appearance, with the exception of identification and labeling	The main function of the alarm is to alert the operations staff should the switch be moved from its normal position during normal operations. (See pictures of switch at the end of this enclosure). In this case, the ARI will direct that an operator be dispatched to the 6.9 kV shutdown boards to determine which switch has been moved to the bypass position. All bypass switches will be identical in design and appearance, with the exception of identification and labeling, which is unique for each 6.9 kV shutdown board.
40.	HF – 13 TVA's application dated June 17, 2015 (ADAMS Accession Number ML15170A474), Enclosure 1, Section 4.1.2, "Postulated GDC 5 Event," states, in part: "The following procedures will be affected… System Operating Instruction SOI-70.01, "Component Cooling Water (CCS) System." This SOI does not require a revision because the steps to realign the CCS Train B pumps are currently in the procedure." Clarify how SOI-70.01 is affected by the changes proposed in this LAR, if it does not require a revision.	This response supersedes the response provided on June 17, 2015. The procedure guidance for aligning either CCS Pump 1B-B or 2B-B previously existed in 0-SOI-70.01, thus no change is required to this instruction for the proposed changes.
41.	SCVB-RAI-1 Reference 1, Attachment to Enclosure 1, "Westinghouse Summary Report, Section 4.4.1.3, seventh bullet in the summarized assumptions for mass and energy release analysis states "Density and specific heat values of 501 lbm/ft^2 and 0.145 BTU/lbm-°F, respectively, model a volumetric heat capacity which bounds the values found in Part D of the ASME boiler pressure vessel code." (a) Provide the material specification for which the	(a) The material properties chosen are intended to represent the three most common structural materials in the RCS; stainless steel 304, stainless steel 316, and low alloy carbon steel. (b) The ASME Boiler and Pressure Vessel Code, Section II, Part D (Reference 1) [hereafter referred to as "the ASME BPVC" (Reference 1)] was used as the source of the material property data. Table PRD provides material densities in units of lbm/in^3. The density of 0.29 lbm/in^3 (converted to 501 lbm/ft^3) is representative of the density at a cold state of 70°F for Stainless Steel 304 and Stainless Steel 316. The density of carbon steel at a cold state of 70°F is listed as 0.28 lbm/in^3 (converted to 484 lbm/ft^3). The bulk of the metal mass in the reactor vessel and the steam generators is carbon steel. The density of stainless steel, 501 lbm/ft^3, was conservatively applied to all steel alloys (note that the steam generator tube material

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	above density and specific heat values are assumed. (b) Confirm that the assumed density 501 lbm/ft3 bounds the values given in "Part D of the ASME Section II Boiler and Pressure Vessel code" instead of "Part D of ASME boiler pressure vessel code", If not provide more information regarding the source of the assumed value of density. (c) ASME Boiler and Pressure Vessel Code (BPVC), Section II, Part D does not provide specific heat values. Please state the source document of the ASME specific heat of the RCS metal which is bounded by the assumed specific heat of 0.145 BTU/lbm-°F given in the above statement.	is treated separately) in the LOCA Mass and Energy (M&E) calculations to maximize the metal mass. (c) The specific heat (as a function of temperature) was determined for each material by using the following information from the ASME BPVC (Reference 1); the equation relating thermal conductivity, thermal diffusivity, density, and specific heat from the Table TCD General Notes, thermal expansion coefficients in Tables TE-1 through TE-4, thermal conductivities in Table TCD, and thermal diffusivities in Table TCD. An uncertainty of 10% was added to the calculated specific heat values. The computer codes that are part of the WCAP-10325-P-A evaluation model are constructed such that a single specific heat value is required. Therefore, a bounding value of 0.145 BTU/lbm°F provides a conservative estimate of the total amount of metal energy over the temperature range that is relevant in a LOCA M&E calculation (metal cools down from approximately 600°F to 200°F) for all three of the structural steel alloys. Reference: 1. An International Code, 2010 ASME Boiler and Pressure Vessel Code, 2010 Edition, July 1, 2010, Section II, Part D, "Properties (Customary), Materials," ASME Boiler and Pressure Vessel Committee on Materials, Three Park Avenue, New York, NY, 10016 USA.
42.	SCVB-RAI-2 Reference 1, Enclosure 1, Section 4.0, "Conclusions", first paragraph states calculated values of the ice weight from containment analysis 2,585,000 lbs with an average ice basket weight of 1,330 lbs. The second paragraph states an ice weight specified in Surveillance Requirement (SR) 3.6.11.2 will be 2,750,700 lbs with a per basket value of 1415 lbs specified in SR 3.6.11.2 and SR 3.6.11.3. (a) Please confirm that the difference between the SR value 2,750,000 lbs and the analytical value	The difference between 2,750,700 lbs and the analytical value of 2,585,000 lbs considers losses due to sublimation only. The surveillance requirements address uncertainties introduced through weighing. (a) The Surveillance Requirement value of 2,750,700 pounds contains a sublimation allowance of six percent. The value does not include a margin for measurement uncertainty. The fact that the value does not include the measurement uncertainty is stated in the Technical Specification Bases for SR 3.6.11.2 and 3. Surveillance Instruction 1-SI-61-2, "18 Month Ice Condenser Surveillance" states that the instrument uncertainty must be added to the SR ice weight value to determine an acceptable basket weight. The allowance for instrument uncertainty is approximately +15 lbs.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	2,585,000 lbs considers the sublimation allowance and the measurement uncertainty during surveillance. If so, how much individual margins are provided for sublimation and measurement uncertainty. If not, explain the basis for the difference between the two values. (b) Please explain the methodology used for calculating the sublimation allowance. What are the assumed initial parameters (such as ice temperature, environment temperature in the ice condenser, etc) for the calculation?	(b) The method used for determining the sublimation rate is from cycle to cycle ice basket weighing. The as-left basket weight is compared to the as-found basket weight in the next cycle for the same basket. Overall trends for the ice bed at large and on a row group basis are also used to validate the sublimation rate. Historical data for WBN Unit 1 and for Sequoyah Nuclear Plant have shown cycle to cycle sublimation rates of around three percent. The selection of six percent is based on engineering judgment to provide a large safety margin. The FSAR 6.7.14.3 will be updated to reflect the following. **Sublimation - Historical** The following information was developed during the design and initial operation of the ice condenser system. Actual sublimation rates have been established during the operation of Watts Bar and are discussed in the Section entitled Sublimation – Actual. The other mechanism that affects the long-term storage of the ice is sublimation. Sublimation has several effects inside the ice condenser. The geometry of the ice mass changes where sublimation occurs, and the resulting vapor is deposited on a colder surface at another location inside the ice condenser. In normal cold storage room application, the cooling coil is exposed to the air in the room, and moisture in the air freezes on the coil. If ice is stored in the room, all of ice eventually migrates to the coil (which is defrosted periodically, draining the water outside the room) through a sublimation-mass transfer mechanism. To avoid the mechanism, and maintain a constant mass of ice, the ice condenser is provided with double wall insulation. The annular gap between the insulated walls is provided with a heat sink in the form of a flow of cool, dry air that enters arid and leaves through the insulated panels. However, a small amount of heat enters the system through the inlet doors, which are not double insulated, and also through the double layer insulation system. The effect of this heat gain on the ice condenser has been examined analytically. An analytical model of the sublimation process has been developed to provide an estimate of the expected sublimation rate as well as identify the significant parameters affecting the sublimation rate. The model developed a relationship identifying the fraction of total heat input which sublimes ice (the rest of the heat raises the temperature of the air, which transports the

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		vapor to the cold surface where it freezes). The sublimation fraction depends on the difference in vapor pressure between warmest and coldest air temperatures within the ice condenser. The sublimation fraction decreases as the ΔT decreases and also as the average ice condenser temperature decreases. For an average temperature of 15°F in the ice condenser compartment, the analytical model predicts a sublimation rate of about 1% of the ice mass sublimed per year per ton (12,000 Btu/hr) of heat gain to the ice storage compartment. The final heat gain calculations identified a heat gain into the ice storage compartment of 1 to 1.5 tons, most of which enters the compartment through the doors. For the purposes of this report, it is assumed that the reference heat gain for the unit is 1 ton, and therefore, the calculated reference sublimation rate would be 1% of the ice weight per year. Selected baskets are weighed as indicated in Technical Specifications to verify that the actual sublimation rate has not excessively depleted the ice inventory. **Sublimation – Operational** The method used for determining the sublimation rate is from cycle to cycle ice basket weighing. The as-left basket weight is compared to the as-found basket weight in the next cycle for the same basket. Overall trends for the ice bed at large and on a row group basis are also used to validate the sublimation rate. Historical data for Watts Bar Unit 1 and for the Sequoyah Nuclear Plant have shown cycle to cycle sublimation rates of around three percent. The selection of six percent is based on engineering judgment to provide a large safety margin. The surveillance acceptance criteria contain a sublimation allowance of six percent. The value does not include a margin for measurement uncertainty. Instrument uncertainty must be added to the surveillance requirement ice weight value to determine an acceptable basket weight. The allowance for instrument uncertainty is approximately +15 lbs. The Ice Bed Temperature is maintained between 15°F and 20°F during plant operation. The empirical sublimation rates described above are the results of operating in this

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
		temperature range. Procedures require actions to restore normal operating temperatures when ice bed temperatures reach 23°F.
43.	SCVB-RAI-3 FSAR Amendment 113, under heading "Sublimation", page 6.7-64 states: "For an average temperature of 15oF in the ice condenser compartment, the analytical model predicts a sublimation rate of about 1% of the ice mass sublimed per year per ton (12,000 Btu/hr) of heat gain to the ice storage compartment." The ice condenser compartment temperature of 15ºF specified in the above statement is not consistent with the ice bay air temperature of 27ºF specified in Table 1 of Reference 1, Enclosure 1. (a) Correct the ice condenser compartment temperature in the above statement from 15ºF to 27oF, or provide justification for the difference. (b) Recalculate the sublimation rate based on ice compartment air temperature of 27ºF, and provide its impact on the sublimation allowance.	TVA has clarified the referenced FSAR Section (6.7) to designate the current verbiage as historical and provided the empirical operational basis for how sublimation is calculated in response to SCVB-RAI-2. (a) and (b) The verbiage in question is part of the historical basis for ice condenser operation. The new operational basis consists of the following. The Ice Bed Temperature is maintained between 15°F and 20°F during plant operation. The empirical sublimation rates described above are the results of operating in this temperature range. Procedure (ARI-138-144) requires actions to restore normal operating temperatures when ice bed temperatures reach 23°F.
44.	SCVB-RAI-4 Reference 1, Enclosure 2, "Watts Bar Nuclear Plant Unit 2 Revised FSAR Section 6.2.1 Pages", page 6.2.1-8, under heading "Structural Heat Removal", states a Tagami heat transfer coefficient for the lower containment compartment structures was limited to 72 Btu/hr-ft^2. Enclosure 1, "Attachment Westinghouse Summary	The proposed long-term LOCA containment integrity analysis used the reviewed and approved LOTIC-1 code. Internal to the configured LOTIC-1 code is a limit placed on the stagnation heat transfer coefficient of 72 Btu/hr-ft^2-°F. This can be seen from the stagnation heat transfer coefficient presented in Equation 54 in WCAP-8355-P-A, "Long Term Ice Condenser Containment Code LOTIC Code," April 1976. When the steam to air ratio is 1.4, H_{stag} is limited to 72 Btu/hr ft^2-°F.

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	Report", Section 4.4.3.3, "Structural Heat Removal", page 42, mentions Tagami correlations for the heat transfer coefficient for the lower compartment structures were used in the proposed analysis, but does not provide its value. Confirm that the proposed analysis used heat transfer coefficient of 72 Btu/hr-ft^2 consistent with the FSAR. In case it is changed, please revise the FSAR and justify if a less conservative (greater) value was used.	The exponential decrease of the heat transfer coefficient is given by: $H_s = H_{stag} + [H_{max} - H_{stag}]\, e^{-.05\,[t-t_p]} \quad t > t_p \qquad (54)$ where $H_{stag} = 2 + 50\chi \quad 0 \leq \chi \leq 1.4$ H_{stag} = H for stagnant conditions (Btu/hr ft^2°F) χ = steam to air weight ratio in containment No changes were made to the code for this proposed analysis, so the structural heat transfer coefficient limit of 72 Btu/ht-ft^2-°F described in the Structural Heat Removal subsection at the end of FSAR Section 6.2.1.3.3 continues to exist.
45.	SCVB-RAI-5 Reference 1, Enclosure 3, "Watts Bar Nuclear Plant Unit 2 Revised Pages for TS and TS Bases 3.6.11", SR 3.6.11.2a weighs samples ≥ 144 ice baskets and verifies each basket contains ≥ 1415 lbs of ice. **This surveillance allows individual baskets to weigh greater than 1415 lbs**. FSAR Amendment 113, Section 6.7.6.1, page 6.7-22, specifies maximum total ice weight 3x10^6 lbs which results in an individual ice basket weight (3x10^6/1944) = 1543.2 lbs. (a) Confirm that the maximum ice weight of 3x10^6 lbs is based on seismic qualification test results. (b) Explain how, during surveillance, it would be verified that the individual ice basket weight and	According to the Revision log for SDD N3-61-4001, Westinghouse performed a post-LOCA containment sump boron concentration analysis to address PER 03-006899-000. The maximum ice weight used as an input to the Westinghouse Analysis was 3.0 x 10^6 lbs. DCN D51416-A revised the SDD to ensure the maximum total ice condenser weight does not exceed 3.0 x 10^6 lbs. Westinghouse Letter WAT-D-10850, Section 1.1, concludes the evaluation performed by Westinghouse determined that the WBN ice basket maximum average loading limits and configuration requirements are acceptable based on seismic design allowables. The configuration shown is 1/3 of 1944 baskets (648 baskets) at a maximum ice weight including basket of 1809 lbs, another 648 baskets at a maximum ice weight including basket of 1909 lbs and the final 648 baskets at a maximum ice weight including basket of 2009 lbs for a total of 3,711,096 lbs of ice (including the weight of the baskets). The weight of the ice baskets as 250 lbs for a total empty ice basket weight of 250 x 1944 = 486,000 lbs. Therefore, WBN is seismically analyzed for 3,711,096 – 486,000 = 3,225,096 lbs of ice in the ice condenser. As mentioned above, the limiting maximum

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	the total ice weight in the ice condenser would not exceed the maximum limits.	value of ice is not based on the seismic qualification of the ice condenser but rather the post-LOCA sump boron concentration. Surveillance requirements provide acceptance criteria not to exceed the 3,000,000 lb total weight based on the sample of ice basket weights as defined by Technical Specifications. 1-SI-61-2, "18 Month Ice Weighing," surveillance. Additionally, individual ice baskets weights as described above are also controlled in the surveillance procedure as follows. 1) The structural limit for the Ice Condenser per basket in rows 1, 2 & 3 is 1,809 lbs, but 1,794 lbs will be the adjusted weight to allow for M&TE inaccuracy when using the electronic load cells. 2) The structural limit for the Ice Condenser per basket in rows 4, 5 & 6 is 1,909 lbs, but 1,894 lbs will be the adjusted weight to allow for M&TE inaccuracy when using the electronic load cells. 3) The structural limit for the Ice Condenser per basket in rows 7, 8 & 9 is 2,009 lbs, but 1,994 lbs will be the adjusted weight to allow for M&TE inaccuracy when using the electronic load cells.
46.	SCVB-RAI-6 Reference 1, Enclosure 1, Attachment, Section 4.4.1.1 states: "In addition to the design basis, this analysis accounted for the effects of other plant changes of which Westinghouse is aware. These include increased……..." Please describe all other changes that were incorporated in the mass and energy analysis beside the four changes described in Section 4.4.1.1.	In addition to the design basis, this analysis accounted for the effects of other plant changes of which Westinghouse is aware. These include increased valve stroke time (of +13 seconds) to open the containment spray flow control valves (Reference 1), initial condition uncertainties on RCS temperature of +7°F, and 17x17 Robust Fuel Assembly-2 (RFA-2) fuel (which may incorporate tritium-producing burnable absorber rods (TPBAR)). Also, the evaluation that was provided in Reference 17 with the conclusion that a +/- 0.2 Hz variation in the diesel frequency would have a negligible impact on the LOCA mass and energy release analysis remains valid. It should be noted that these items were included for completeness even though they may not be currently implemented at WBN Unit 2.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
47.	SCVB-RAI-7 Please confirm that Loss of train A concurrent with loss of offsite power (LOOP) during which CCS heat exchanger (HX) 'C' carries the heat load of both the LOCA and shutdown unit is the most limiting condition for the CCS fluid temperature, containment analysis parameters (containment peak pressure and temperature) and the shutdown cooling analysis. For the most limiting case of heat loads, specify at what time after initiation of shutdown cooling in the "Shutdown Unit" the LOCA is assumed to occur in the "LOCA Unit."	For the time after initiation of shutdown cooling in the "Shutdown Unit," the LOCA is assumed to occur at 7-hours after initiation of the shutdown. The loss of Train A power is the limiting case for the containment analysis.
48.	TS – 2 Please provide a more complete explanation of why Conditions B of both proposed LCO 3.7.16 and proposed LCO 3.7.17 have no effective completion, i.e. "Once per 12 hours" ? Is this condition proposed to continue until the LCO expires i.e. "… 48 hours after entry into Mode 3 from Mode 1 or 2" ? The current basis statement for each LCO does not adequately explain why no restoration action is needed. <u>Date Posted</u>: 08/25/15	Per LCO 3.0.2, if the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Actions is not required unless otherwise stated. Therefore, the Completion Times of LCO 3.7.16 and LCO 3.7.17 Condition B Required Actions continue until the LCO is met or is no longer Applicable. The TS Bases will be revised to reflect the follow: The purpose of the guidance contained in Condition B is to ensure clear direction is given to NOT enter Mode 5, if the additional ERCW and CCS alignments associated with TS 3.7.16 and TS 3.7.17 are not performed. This assumes that the CCS and ERCW System meet the requirements of TS 3.7.7 and TS 3.7.8, respectively, to support continued operation in Mode 4. In this case, with the plant in Mode 4, additional methods of decay heat removal are available and the potential for an uncontrolled heatup from Mode 5 should the postulated accident occur is avoided. Should additional inoperabilities impact compliance with TS 3.7.7, TS 3.7.8, or TS 3.4.6, the Actions associated with those TSs would prevail.
49.	TS – 4 The supplemental proposal changed the applicability statements for the new LCOs 3.7.16 and 3.7.17 without explanation in the document. Please further explain why the applicability statement "This LCO is	LCO 3.7.16 / LCO 3.7.17 Applicability Note b was originally proposed to preclude the requirement for additional CCS and ERCW pumps if complying with Required Actions to be in Mode 5, since additional failures, such as a loss of Train A 6.9 kV shutdown boards, does not have to be postulated while in a TS Action. However, as stated in the NRC scenario, if the postulated failure is the loss of Train A 6.9 kV shutdown boards, the "GDC 5 event" is still viable and the requirement for the additional CCS and ERCW pumps

Item No.	NRC Question/Request Date Posted	TVA Response/Dated Posted
	not applicable for either of the following conditions: … b. When complying with Required Actions to be in Mode 5" was essentially made new Condition A in each of the new TS "… AND Complying with Required Actions to be in Mode 5." Date Posted: 08/03/15	is still required. Therefore, Applicability Note b was removed, as well as the adoption of TSTF-273. If the requirement of either LCO 3.7.16 or LCO 3.7.17 is not met, maintaining the unit in Mode 4 with decay heat removal from the RCS loops is preferred. However, if TS Required Actions require entry into Mode 5, the remaining operable RHR loop is sufficient to cooldown the unit to and maintain it in Mode 5, even with a concurrent LOCA in the other unit. Therefore, the wording of Conditions A and B provide for these two scenarios.
50.	TS – 5 In a public meeting and audit discussions it has been presented that the additional ERCW pumps required by proposed LCO 3.7.17 are needed essentially for increased ERCW flow when the accident unit switches to recirculation mode to remove heat from the accident unit's Containment Spray heat exchangers. Please clearly articulate this point in a docketed submittal. Date Posted: 08/03/15	The following information was provided by TVA in letter dated June 17, 2015, Enclosure 1, Section 4.1.2, "Postulated GDC 5 Event," page E1-10: The ERCW System design was based on requiring two ERCW pumps to handle the cooling loads to the UHS for shutting down both units during either normal operation or in the event of a LOCA and the shut down of the non-accident unit. It has been determined, for the specific set of scenarios in this evaluation, that three ERCW pumps will be required if a cool down of the non-accident unit using RHR occurs within the first 48 hours after a shutdown. The higher heat loads associated with continuing the cool down of the unit that has been shut down for less than 48 hours, combined with the heat removal requirements of the safety analysis for the DBA LOCA via RHR and containment spray, necessitates the use of three ERCW pumps during the initial 48 hour time period. This additional cooling capacity is required prior to placing containment spray on recirculation mode. Once the unit has been shut down for 48 hours or more, the total ERCW heat removal and thus, flow requirements, drop below the flowrate provided by two ERCW pumps.

For integrated HX performance, See EPMJN010890 Table C7.7.69 and for U2 C7.7.89
At time 7-hours, provide two virtual HX models

LOCA Unit Apportioned by Flow

LOOP & Loss of Train A
LOCA Recirc
Supports Loss of Train A LOCA
Containment Analysis

	CCS HX C	RHR HX	RHR HX	MISC
UA	3.35	1.63	NOT IN	N/A
F	1.00	0.90	SERVICE	
m HOT, gpm	5,000.00	3,801.98		
m HOT, m#/hr	2.47	1.86		
DENSITY (of cold fluid)	61.23	61.95		
M cold, gpm	4,524.89	5,000.00		
M cold, m#/hr	2.22	2.48		
R	0.86	1.22		
S	-0.73	-1.15		
Q (MBtu/hr)	54.80	54.80		
t1 (deg. F)	85	103		
t2 (deg. F)	110	125		
T2 (deg. F)	103	136		
T1 (deg. F)	125	166		
delta t (deg. F)	25	22		
delta T (deg. F)	22	29		
HX CORRECTION FACTOR				
r		0.75		
p		0.47		
F		0.92		
UA ADJUSTMENT				
m HOT, DESIGN	2.21	1.48		
M cold, DESIGN	2.95	2.48		
ho, DESIGN	1,172.18	1,386.96		
ho, ACTUAL	1,252.52	1,390.11		
hi, DESIGN	1,013.34	2,159.37		
hi, ACTUAL	807.61	2,592.56		
U, ACTUAL	190.42	381.27		
UA, ACTUAL	3.35	1.63		

LOCA Unit Apportioned by Heat Load

LOOP & Loss of Train A
LOCA Recirc
Supports LoTA LOCA Containment Analysis

	CCS HX C	RHR HX	RHR HX	MISC
UA	2.64	1.63	NOT IN	N/A
F	1.00	0.90	SERVICE	
m HOT, gpm	5,000.00	3,801.98		
m HOT, m#/hr	2.47	1.86		
DENSITY (of cold fluid)	61.23	61.84		
M cold, gpm	3,483.59	5,000.00		
M cold, m#/hr	1.71	2.48		
R	0.62	1.22		
S	-0.88	-1.15		
Q (MBtu/hr)	54.80	54.80		
t1 (deg. F)	85	111		
t2 (deg. F)	117	133		
T2 (deg. F)	111	145		
T1 (deg. F)	133	174		
delta t (deg. F)	32	22		
delta T (deg. F)	22	29		
HX CORRECTION FACTOR				
r		0.75		
p		0.47		
F		0.92		
UA ADJUSTMENT				
m HOT, DESIGN	1.70	1.48		
M cold, DESIGN	2.27	2.48		
ho, DESIGN	1,172.18	1,386.96		
ho, ACTUAL	1,463.36	1,388.62		
hi, DESIGN	1,013.34	2,159.37		
hi, ACTUAL	807.61	2,592.56		
U, ACTUAL	194.67	381.16		
UA, ACTUAL	2.64	1.63		

LOOP & Loss of Train A
Shutdown Unit
Supports LoTA LOCA Containment Analysis

	CCS HX C	RHR HX	RHR HX	MISC
UA	4.12	1.57	NOT IN	N/A
F	1.00	0.90	SERVICE	
m HOT, gpm	5,000.00	3,025.23		
m HOT, m#/hr	2.46	1.48		
DENSITY (of cold fluid)	61.23	61.92		
M cold, gpm	5,679.60	5,000.00		
M cold, m#/hr	2.79	2.48		
R	1.22	1.47		
S	-0.63	-1.25		
Q (MBtu/hr)	89.27	89.27		
t1 (deg. F)	85	105		
t2 (deg. F)	117	141		
T2 (deg. F)	105	156		
T1 (deg. F)	141	216		
delta t (deg. F)	32	36		
delta T (deg. F)	36	60		
HX CORRECTION FACTOR				
r		0.60		
p		0.54		
F		0.90		
UA ADJUSTMENT				
m HOT, DESIGN	2.78	1.48		
M cold, DESIGN	3.70	2.48		
ho, DESIGN	1,172.18	1,386.96		
ho, ACTUAL	1,089.98	1,389.78		
hi, DESIGN	1,013.34	2,159.37		
hi, ACTUAL	807.61	2,159.37		
U, ACTUAL	186.19	368.36		
UA, ACTUAL	4.12	1.57		

The respective UAs are 3.35 apportioned by flow and 2.64 apportioned by heat load. A UA of 3.17 was used in the analysis.

A sensitivity run for Unit 2 by Westinghouse showed that a change in UA from 3.17 to 2.00 using an ERCW flow of 3504 gpm to the virtual component cooling heat exchanger resulted in an increase in containment pressure from 11.73 to 11.76 psig; therefore, the use of any of the three values (3.35, 3.17, or 2.64) will produce nearly identical containment pressures. Therefore, the containment design conditions are not exceeded for the two or three pump ERCW cases.

CCW HX UA		3.17	2.00
At 2718 sec Spray Recirc	spray=	20,032 BTU/sec	20,047 BTU/sec
	RHR=	10,253 BTU/sec	9,262 BTU/sec
At 3600 sec RHR Spray Start	spray=	19,551 BTU/sec	19,641 BTU/sec
	RHR=	9,128 BTU/sec	8,347 BTU/sec

Unit 1 W-COBRA/TRAC (Lotic 1) Results

A similar analysis was run on Unit 1 with a UA of 2.00 and an ERCW flow of 3504 gpm to the virtual component cooling heat exchanger. There is no impact to the peak calculated pressure using the new approved W-COBRA/TRAC Mass & Energies when the CCW HX UA is reduced to 2.0. The change in heat rates at the time of spray initiation is shown in the table below.

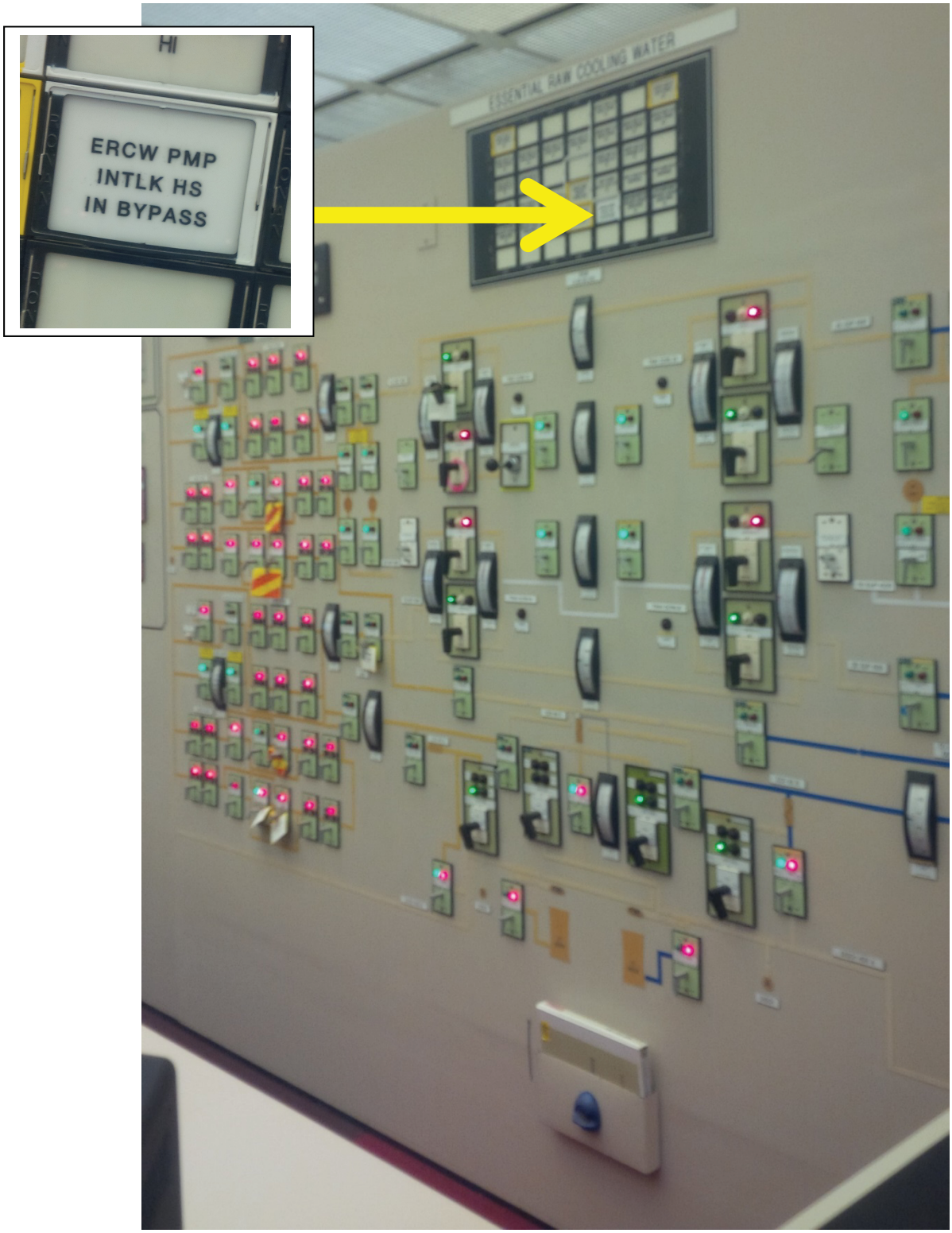
CCW HX UA		3.17	2.00
At 2718 sec Spray Recirc	spray=	22,798 BTU/sec	22,798 BTU/sec
	RHR=	11,770 BTU/sec	10,612 BTU/sec

There was no change in the spray heat removal rate and an approximately 10% reduction in the RHR heat removal rate. The RHR sprays are not credited in the analysis which uses W-COBRA/TRAC Mass & Energies because the calculated pressure will either not exceed 9.5 psig, or will only remain above 9.5 psig for a duration less than 3600 seconds.

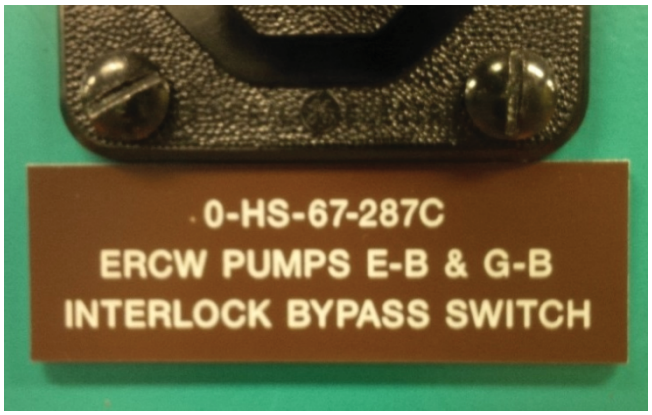
Therefore, similarly on U1 the containment design is not challenged and the results are acceptable for two or three ERCW pump cases.

Responses to NRC Audit Review Questions for WBN Unit 1 ERCW and CCS License Amendment Request

Procedure	Type	Description
0-SI-82.01, Diesel Generator (DG) 1A-A 0-SI-82.02, Diesel Generator (DG) 1B-B 0-SI-82.03, Diesel Generator (DG) 2A-A 0-SI-82.04, Diesel Generator (DG) 2B-B	System Operating Instructions	Contains a check of switch position in the diesel standby alignment. This is performed after each surveillance run, the most frequent of which is monthly. In addition, this check is performed upon return to service of the DG following any maintenance activities.
1/2-GO-4, Normal Power Operation 1/2-GO-5, Unit Shutdown from 30% 1/2-GO-6, Unit Shutdown from HS to CSD	General Operating Instructions	Contains direction to commence alignments (GO-4 and GO-5). Contains direction to ensure alignments are complete prior to entering Mode 4 in GO-6.
1/2-E-1, Loss of Reactor or Secondary Coolant	Emergency Operating Instructions	Contains direction to place in operation the equipment needed following a LOCA.



MCR ANNUNCIATOR - ERCW PUMP INTERLOCK HANDSWITCH IN BYPASS



ERCW PUMP INTERLOCK BYPASS SWITCH EXAMPLE

Updated June 17, 2015 letter, Enclosure 1, Table 3 Summary of Steady-State DG Loading with 3 ERCW Pumps (0 mins to end)																
Pumps	U1 LOCA / U2 Shutdown / Loss of Train A				U2 LOCA / U1 Shutdown / Loss of Train A				U1 LOCA / U2 Shutdown / Loss of Train B				U2 LOCA / U1 Shutdown / Loss of Train B			
DG	1A	2A	1B	2B	1A	2A	1B	2B	1A	2A	1B	2B	1A	2A	1B	2B
ERCW			805	1610			1610	805	805	1610			1610	805		
CCS			378	720			378	720	720***	378			720***	378		
AFW (motor-driven)			400*	**			**	400*	400*	**			**	400*		
Containment Spray			596					596	596					596		
Centrifugal Charging			695	532			532	695	695	532			532	695		
SI			460					460	460					460		
RHR			440	370			370	440	440	370			370	440		
Total / Large Motor Load (HP)			3774	3232			2890	4116	4116	2890			3232	3774		
Pressurizer Heaters (kW)				500			500			500			500			
DG Loading ****																
0 - 20 minutes																
kW			4182	4261			4188	4283	4004	4174			3984	4165		
kVA			4851	4797			4745	4930	4622	4684			4492	4781		
20 min - 2 hours																
kW			4218	4015			3941	4313	4040	3927			3738	4201		
kVA			4887	4514			4462	4960	4658	4401			4209	4818		
2 hours - End																
kW			4067	4015			3941	4163	4148	3927			4015	4033		
kVA			4688	4514			4462	4766	4769	4401			4513	4625		

Note: Refer to Table 1 for CCS and ERCW pump power alignments.

* 0 Min – 2 Hrs 600 hp until SGs refilled; thereafter 400 hp (for both LBLOCA and SBLOCA).
** 0 Min – 20 minutes 300 hp; then stopped
*** 378 hp until CCS Pump C-S is manually aligned after 2 hours for spent fuel pool cooling; then 720 hp (360hp each for CCSP 1A and C-S)
**** Values extracted from Appendix N-1, Pages 1 thru' 4 of Diesel Loading Calculation, EDQ00099920080014 R31.