UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 20, 2015

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING APPLICATION TO REVISE TECHNICAL SPECIFICATIONS FOR COMPONENT COOLING WATER AND ESSENTIAL RAW COOLING WATER TO SUPPORT DUAL UNIT OPERATION (TAC NO. MF6376)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. NPF-90 for the Watts Bar Nuclear Plant, Unit 1. This amendment consists of changes to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," and TS 3.4.6, "Reactor Coolant System Loops – MODE 4"; and the addition of new TS 3.7.16, "Component Cooling System – Shutdown," and new TS 3.7.17, "Essential Raw Cooling Water System – Shutdown," in response to your application dated June 17, 2015, as supplemented by letters dated July 14, August 3, August 28, September 3, and September 21, 2015.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Jeanne A. Dion

Jeanne A. Dion, Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures:
1. Amendment No. 104 to NPF-90
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. NPF-90

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Tennessee Valley Authority (TVA or the licensee) dated June 17, 2015, as supplemented by letters dated July 14, August 3, August 28, September 3, and September 21, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, the license is amended as indicated in the attachment to this license amendment.

   Paragraph 2.C.(2) of Facility Operating License No. NPF-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 104 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance, and shall be implemented after the issuance of the facility operating license for Watts Bar Nuclear Plant, Unit 2, and prior to Watts Bar Nuclear Plant, Unit 2, entry into Mode 4, "Hot Shutdown."

FOR THE NUCLEAR REGULATORY COMMISSION

Jessie F. Quichocho, Chief
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License and Technical Specifications

Date of Issuance: October 20, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Replace the following pages of Facility Operating License NPF-90 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
3	3

Technical Specifications

REMOVE	INSERT
3.3-38	3.3-38
3.4-11	3.4-11
3.4-12	3.4-12
3.4-13	3.4-13
-	3.7-33
-	3.7-34
-	3.7-35
-	3.7-36
-	3.7-37
-	3.7-38

(4) TVA, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, instrument calibration, or other activity associated with radioactive apparatus or components; and

(5) TVA, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 104 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Safety Parameter Display System (SPDS) (Section 18.2 of SER Supplements 5 and 15)

Prior to startup following the first refueling outage, TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

(4) Vehicle Bomb Control Program (Section 13.6.9 of SSER 20)

During the period of the exemption granted in paragraph 2.D.(3) of this license, in implementing the power ascension phase of the approved initial test program, TVA shall not exceed 50% power until the requirements of 10 CFR 73.55(c)(7) and (8) are fully implemented. TVA shall submit a letter under oath or affirmation when the requirements of 73.55(c)(7) and (8) have been fully implemented.

Table 3.3.2-1 (page 5 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. Loss of Offsite Power	1, 2 ,3	4 per bus	F	Refer to Function 4 of Table 3.3.5-1 for SRs and Allowable Values		
e.. Trip of all Turbine Driven Main Feedwater Pumps	1$^{(i)}$, 2$^{(j)}$	1 per pump	J	SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.10	≥ 48 psig	50 psig
f. Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure - Low	1, 2, 3, 4$^{(k)}$	3	B	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) 1.2 psig B) 2.0 psig
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA

(continued)

(i) Entry into Condition J may be suspended for up to 4 hours when placing the second Turbine Driven Main Feedwater (TDMFW) Pump in service or removing one of two TDMFW pumps from service.

(j) When one or more Turbine Driven Feedwater Pump(s) are supplying feedwater to steam generators.

(k) When steam generators are relied on for heat removal.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops shall be OPERABLE, and consist of either:

c. Any combination of RCS loops and residual heat removal (RHR) loops, and one loop shall be in operation, when the rod control system is not capable of rod withdrawal; or

d. Two RCS loops, and both loops shall be in operation, when the rod control system is capable of rod withdrawal.

---NOTES---

1. No RCP shall be started with any RCS cold leg temperature ≤ 350°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures.

2. For the initial 7 hours after entry into MODE 3 from MODE 1 or MODE 2, two loops shall consist of:

   a. Two RCS loops with one loop in operation when the rod control system is not capable of rod withdrawal; or

   b. Two RCS loops with both loops in operation when the rod control system is capable of rod withdrawal.

3. Average reactor coolant temperature shall be maintained > 200°F for the initial 7 hours after entry into MODE 3 from MODE 1 or MODE 2.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Only one RCS loop OPERABLE. AND Two RHR loops inoperable. OR Less than 7 hours since entry into MODE 3 from MODE 1 or MODE 2.	A.1	Initiate action to restore a second required RCS or RHR loop to OPERABLE status.	Immediately
B. One required RHR loop inoperable. AND No RCS loops OPERABLE.	B.1	Be in MODE 5.	24 hours
C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.	C.1	Restore required RCS loop to operation.	1 hour
	OR		
	C.2	De-energize all control rod drive mechanisms (CRDMs).	1 hour
D. Required RCS or RHR loops inoperable. OR No required RCS or RHR loop in operation.	D.1	De-energize all CRDMs.	Immediately
	AND		
	D.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	AND		
	D.3	Initiate action to restore one required loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify two RCS loops are in operation when the rod control system is capable of rod withdrawal.	12 hours
SR 3.4.6.2	Verify one required RHR or RCS loop is in operation when the rod control system is not capable of rod withdrawal.	12 hours
SR 3.4.6.3	Verify SG secondary side water levels are greater than or equal to 32% narrow range for required RCS loops.	12 hours
SR 3.4.6.4	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.7 PLANT SYSTEMS

3.7.16 Component Cooling System (CCS) - Shutdown

LCO 3.7.16 Two CCS trains shall be OPERABLE with one pump powered from Train A and aligned to the Train A header, and two pumps powered from Train B and aligned to the Train B header.

APPLICABILITY: MODES 4 and 5.

--NOTE--
This LCO is not applicable more than 48 hours after entry into MODE 3 from MODE 1 or 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCS train inoperable in MODE 4. <u>AND</u> Complying with Required Actions to be in MODE 5.	A.1 Be in MODE 5.	24 hours
B. One CCS train inoperable in MODE 4 for reasons other than Condition A.	B.1 Verify two OPERABLE reactor coolant system (RCS) loops and one RCS loop in operation. <u>AND</u> B.2 Verify $T_{avg} > 200°F$.	Once per 12 hours Once per 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two CCS trains inoperable in MODE 4.	C.1 ---------------NOTES-------------- 1. LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one CCS train is restored to an OPERABLE status. 2. Enter Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal (RHR) loops made inoperable by CCS. ---------------------------------------	
	Initiate action to restore one CCS train to OPERABLE status.	Immediately
D. One or more CCS train(s) inoperable in MODE 5.	D.1 ---------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," for RHR loops made inoperable by CCS. ---------------------------------------	
	Initiate action to restore CCS train(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify correct breaker alignment and indicated power available to the required pump(s) that is not in operation.	12 hours
SR 3.7.16.2	Verify two CCS pumps are aligned to CCS Train B.	12 hours

3.7 PLANT SYSTEMS

3.7.17 Essential Raw Cooling Water (ERCW) System - Shutdown

LCO 3.7.17 Two ERCW trains shall be OPERABLE as follows:

a. Three ERCW pumps aligned to Train A, including two pumps capable of being powered from 6.9 kV Shutdown Board 1A-A, and

b. Three ERCW pumps aligned to Train B, including two pumps capable of being powered from 6.9 kV Shutdown Board 1B-B.

APPLICABILITY: MODES 4 and 5.

--NOTE--

This LCO is not applicable more than 48 hours after entry into MODE 3 from MODE 1 or 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ERCW train inoperable in MODE 4. AND Complying with Required Actions to be in MODE 5.	A.1 Be in MODE 5.	24 hours
B. One ERCW train inoperable in MODE 4 for reasons other than Condition A.	B.1 Verify two OPERABLE reactor coolant system (RCS) loops and one RCS loop in operation. AND B.2 Verify T_{avg} > 200°F.	Once per 12 hours Once per 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Two ERCW trains inoperable in MODE 4.	C.1 ---------------NOTES-------------- 1. LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one ERCW train is restored to an OPERABLE status. 2. Enter Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal (RHR) loops made inoperable by ERCW. -- Initiate action to restore one ERCW train to OPERABLE status.	 Immediately
D. One or more ERCW train(s) inoperable in MODE 5.	D.1 ---------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," for RHR loops made inoperable by ERCW. -- Initiate action to restore ERCW train(s) to OPERABLE status.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify correct breaker alignment and indicated power available to the required pump(s) that is not in operation.	12 hours

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. NPF-90

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-390

1.0 INTRODUCTION

By letter June 17, 2015 (Reference 1), as supplemented by letters dated July 14 (Reference 2), August 3 (Reference 3), August 28 (Reference 4), September 3 (Reference 5), and September 21, 2015 (Reference 6 and Reference 7), Tennessee Valley Authority (TVA, the licensee) submitted a request for a change to Facility Operating License No. NPF-90 for Watts Bar Nuclear Plant (WBN), Unit 1. The proposed change would revise Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," and TS 3.4.6, "Reactor Coolant System Loops – MODE 4"; and adopt new 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. ERCW and CCS are safety-related cooling water systems that will be shared by WBN, Units 1 and 2, upon an affirmative decision to issue an operating license for WBN, Unit 2. The new TSs for CCS and ERCW provide supplemental cooling capability to safely manage the heat load attributable to dual-unit operation during the first 48 hours after a unit shutdown.

The licensee provided supplemental information in response to several requests for additional information (RAIs). The supplemental letters dated July 14, August 3, August 28, and September 21, 2015, provided additional information that clarified the application. These supplements did not change the NRC staff's proposed no significant hazards consideration. The supplemental letter dated September 3, 2015, provided additional information that clarified the application and expanded the scope of the application as originally noticed in the *Federal Register* (FR), 80 FR 42552 on July 17, 2015. A subsequent notice was published in the FR on September 15, 2015, along with the U.S. Nuclear Regulatory Commission (NRC) staff's proposed no significant hazards consideration determination (80 FR 55383).

In its June 17, 2015, letter, the licensee also proposed changes to TS 5.7.2.18, "Safety Function Determination Program," and described changes to the Bases for TS 3.0.6, "LCO [Limiting Condition for Operation] Applicability." The licensee stated that these proposed changes were consistent with Technical Specification Task Force (TSTF) Traveler TSTF-273-A, Revision 2. By its letter dated July 14, 2015, the licensee withdrew the proposed changes related to TS 5.7.2.18 and the Bases for TS 3.0.6

Enclosure 2

Background

TSs 3.7.7, "Component Cooling System (CCS)," and 3.7.8, "Essential Raw Cooling Water (ERCW) System," specify the LCOs for the CCS and ERCW. The LCOs establish minimum equipment operability of these systems to mitigate a design-basis accident (DBA) or transient. LCO 3.7.7 requires two operable trains of CCS, each capable of removing the peak heat load from the accident unit by supplying CCS water to the associated residual heat removal (RHR) heat exchanger (HX) for containment and sump water cooling, and to the engineered safety feature (ESF) pump seal and oil coolers. LCO 3.7.8 for ERCW requires two operable trains, each capable of removing the peak heat load from the CCS and containment spray (CS) HXs and other safety-related loads during the DBA or transient.

With the proposed licensing of WBN, Unit 2, TSs 3.7.7 and 3.7.8, would no longer adequately specify the LCOs for these shared systems under some dual-unit plant conditions. Specifically, with Unit 1 in Mode 4 or 5 within 48 hours after shutdown and without steam generator (SG) heat removal available and Unit 2 mitigating a DBA. In this situation, TSs 3.7.7 and 3.7.8 do not provide for sufficient ERCW and CCS cooling to mitigate the DBA and maintain Unit 1 in Mode 4 or 5. With insufficient cooling for the non-accident unit, the non-accident unit could heat up and possibly ascend into a higher TS-defined operational mode. TS 3.0.4 does not permit Mode change in this situation unless stated conditions are met. Because TS 3.7.7 and 3.7.8 would not ensure sufficient cooling for the plant conditions described above, the licensee proposed additional LCOs for CCS and ERCW for the first 48 hours after shutdown if Unit 1 is in Mode 4 or 5.

2.0 REGULATORY EVALUATION

2.1 System Descriptions

2.1.1 Component Cooling System

The CCS is a closed cooling water system with five pumps and three HXs. Section 9.2.2 of the WBN, Unit 1, Updated Final Safety Analysis Report (UFSAR) provides additional information regarding the CCS. The CCS provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. The safety functions of the CCS are to remove heat from the ESF loads, including the RHR HXs (used for reactor and containment cooling), and the oil/seal HX coolers for safety injection (SI) pumps, centrifugal charging pumps (CCP), CS pumps, and RHR pumps. During normal operation, the CCS also provides cooling for various nonessential components and the spent fuel storage pool. The CCS transfers heat to the ERCW, which supplies the tube side of each CCS HX.

The CCS has a Unit 1 Train A header supplied by CCS Pump 1A-A and cooled by CCS HX A. Unit 2 will have a separate Train A header containing CCS HX B supplied by CCS Pump 2A-A. The Train B header will be shared by Unit 1 and Unit 2 and contains CCS HX C. Flow through the Train B header is normally supplied by CCS Pump C-S. CCS Pump 1B-B can be aligned to supply the Train B header, but it is normally aligned to the Unit 1 Train A header. Similarly, CCS Pump 2B-B can supply cooling water to the Train B header, but is normally aligned to the Unit 2 Train A header.

2.1.2 Essential Raw Water Cooling System

The ERCW is a once through open end cooling water system providing cooling water from the ultimate heat sink to the CCS HXs described above and to additional safety-related components needed during a DBA or transient. Section 9.2.1 of the WBN, Unit 1, UFSAR provides additional information regarding the ERCW. During normal operation, the ERCW also provides cooling for various nonessential components. The ERCW consists of two trains each having four ERCW pumps, two traveling water screens, two strainers and associated piping, valves, and instrumentation. Each train will be shared by both units. The trains are redundant and independent of each other. The safety functions of the ERCW are to remove heat from the CCS HX, diesel generator (DG) HXs, CS HXs, Main Control Room and various electric switchboard room chillers, ESF room coolers, and various penetration room coolers. Train A ERCW supplies the CCS A and B HXs. As noted above, the CCS A HX supports the Unit 1 Train A CCS header and the CCS B HX supports the Unit 2 Train A CCS header. The Train B ERCW (redundant to Train A ERCW) supplies the CCS C HX that provides one shared Train B CCS header for both units.

2.1.3 Auxiliary Feedwater Supply Sources

The SGs are fed by the Condensate and Feed Water Systems or the Auxiliary Feedwater (AFW) system. UFSAR Section 10.4.9, AFW System, states the preferred source of water for the AFW pumps is the 395,000 gallon Condensate Storage Tank (CST). A minimum of 200,000 gallons in the Unit 1 tank is reserved for the AFW System by means of a standpipe through which other systems are supplied. The ERCW would provide an unlimited backup water supply. Since the ERCW supplies poor quality water, it is not used except in emergencies when the clean water supply is unavailable.

UFSAR Section 9.2.6 describes the Condensate Storage Facilities. It states that each unit's CST is the preferred source of clean water supply for the AFW pumps and a storage reservoir for secondary system water. The tanks are not designed as an ESF. Both tanks are isolable and AFW can be obtained from both tanks. The ESF water source for AFW is the ERCW (Safety Class 2b). The ERCW pool quality feedwater will be used during events when safety is the prime consideration and SG cleanliness is of secondary importance.

In addition, the licensee recently added the Auxiliary Feed Water 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 alternating current (AC) power, or a loss of control air.

2.1.4 Containment

The primary containment for the WBN units is an ice condenser type consisting of a vessel that is a freestanding steel structure made up of a vertical cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. It is divided into three main compartments, (a) the lower compartment, (b) the upper compartment, and (c) the ice condenser compartment. The lower compartment encloses the reactor, SGs, and associated auxiliary systems equipment. The upper compartment contains the refueling cavity, refueling equipment and polar crane used during refueling and maintenance operations. The ice condenser compartment contains the ice condenser, which is an ice bed consisting of borated ice stored in 1,944 baskets.

The primary purpose of the ice condenser is to provide a heat sink in the event of release of energy from a postulated design-basis loss-of-coolant (LOCA) or Main Steam Line Break (MSLB) accident inside the containment. The ice condenser extracts blowdown energy from the break fluid in the early phase of an accident. After ice melt, containment pressure control is provided by the air return fan system, the Containment Spray System (CSS), and the RHR system spray train. The ice condenser limits the containment pressure to below the design pressure for all reactor coolant pipe break sizes up to and including the largest double-ended guillotine break of the reactor coolant system (RCS).

2.1.5 Electrical Power Distribution System

The plant Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B DGs). The preferred offsite power for WBN is supplied from TVA's 161 kilovolt (kV) transmission grid at the Watts Bar Hydro Plant switchyard over two separate transmission lines, each connecting to 161/6.9 kV Common Station Service Transformers (CSSTs) A, B, C, and D. The onsite Class 1E AC Distribution System supplies electrical power to four power sub-trains, shared between the two units, with each sub-train powered by an independent Class 1E 6.9 kV shutdown board. Power sub-trains 1A and 2A comprise load group A, and power sub-trains 1B and 2B comprise load group B.

CSST C is normally aligned to power 6.9 kV Shutdown Boards 1A-A (Unit 1) and 2A-A (Unit 2). CSST D is normally aligned to power 6.9 kV Shutdown Boards 1B-B (Unit 1) and 2B-B (Unit 2). When CSST D is not available, CSST A can be aligned manually to provide power to the Train B 6.9 kV Shutdown Boards 1B-B and 2B-B. Similarly, when CSST C is not available, CSST B can be aligned manually to provide power to the Train A 6.9 kV Shutdown Boards 1A-A and 2A-A.

Two DGs associated with one load group provide all safety-related functions to mitigate a LOCA in one unit and safely shutdown the opposite unit. Each 6.9 kV shutdown board has two separate and independent offsite sources of power as well as a dedicated onsite DG source.

WBN uses four DGs (two for Train A, designated Load Group A, and two for Train B, designated Load Group B). Two DGs and one load group can provide all safety related functions to mitigate a LOCA in one unit while safely shutting down the other unit. Each DG starts automatically on an SI signal (i.e., low pressurizer pressure or high containment pressure signals) or on a 6.9 kV shutdown board degraded voltage or loss-of-voltage signal. After the DG starts, it automatically ties to its respective 6.9 kV shutdown board after offsite power is tripped as a consequence of 6.9 kV shutdown board loss-of-voltage or degraded voltage, independent of or coincident with an SI signal. The DGs also start and operate in the standby mode without tying to the 6.9 kV shutdown board on an SI signal alone. Following the trip of offsite power, a loss-of-voltage signal strips all nonpermanent loads from the 6.9 kV shutdown board. After the DG is tied to the 6.9 kV shutdown board, selected safety-related loads are sequentially connected to its respective 6.9 kV shutdown board by the automatic sequencer.

The continuous service rating of each DG is 4,400 kW with 10 percent overload permissible for up to 2 hours in any 24-hour period. The ESF loads that are sequentially powered from the DGs are provided in UFSAR Table 8.3-3.

2.2 Description of Proposed Changes

The proposed change would add new LCOs 3.7.16, "Component Cooling System (CCS) - Shutdown," and 3.7.17, "Essential Raw Cooling Water (ERCW) System - Shutdown," for WBN, Unit 1, to support dual-unit operation. The new LCO 3.7.16 requires two operable trains of CCS (Train A and Train B) with two Train B CCS pumps operable and aligned to the B Train CCS header for the first 48 hours after entry into Mode 3 from Mode 1 or Mode 2. The new LCO 3.7.17 requires two operable ERCW trains (Train A and Train B) with three ERCW pumps aligned to each train for the first 48 hours after entry into Mode 3 from Mode 1 or Mode 2.

The existing LCOs 3.7.7, "Component Cooling System (CCS)," and 3.7.8, "Essential Raw Cooling Water (ERCW) System," remain applicable, and are supplemented by the new LCOs 3.7.16 and 3.7.17 during the first 48 hours of a shutdown of Unit 1.

The associated Conditions, Required Actions, and Completion Times for proposed LCOs 3.7.16 and 3.7.17 are as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCS [ERCW] train inoperable in MODE 4. AND Complying with Required Actions to be in MODE 5.	A.1 Be in MODE 5.	24 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One CCS [ERCW] train inoperable in MODE 4 for reasons other than Condition A.	B.1 Verify two OPERABLE reactor coolant system (RCS) loops and one RCS loop in operation. AND B.2 Verify T_{avg} > 200°F.	Once per 12 hours Once per 12 hours
C. Two CCS [ERCW] trains inoperable in MODE 4.	C.1 ---------------NOTES-------------- 1. LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one CCS train is restored to an OPERABLE status. 2. Enter Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal (RHR) loops made inoperable by CCS [ERCW]. -------------------------------------- Initiate action to restore one CCS [ERCW] train to OPERABLE status.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more CCS [ERCW] train(s) inoperable in MODE 5.	D.1 ---------------NOTE---------------- Enter applicable Conditions and Required Actions of LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," for RHR loops made inoperable by CCS [ERCW]. -- Initiate action to restore CCS [ERCW] train(s) to OPERABLE status.	Immediately

The licensee also proposed surveillance requirements (SRs) to verify the availability and operability of the required CCS [or ERCW] pumps.

In its September 3, 2015, supplement, the licensee submitted additional proposed changes in response to NRC staff requests for additional information.

The licensee proposed to modify LCO 3.3.2, Table 3.3.2-1, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” Function 6.f, “Auxiliary Feed Water Pumps Train A and B Suction Transfer on Suction Pressure - Low,” by adding “MODE 4 when relying on the SGs for heat removal” to the Applicability.

The licensee proposed to add two notes to LCO 3.4.6 to require two reactor coolant system (RCS) loops be Operable for the initial 7 hours after entry into Mode 3 from Mode 1 or Mode 2 and to require RCS temperature be maintained > [greater than] 200 °F. The added notes also include requirements that two RCS loops be in operation when the rod control system is capable of rod withdrawal or one of the two RCS loops be in operation when the rod control system is not capable of rod withdrawal.

The licensee also proposed to modify Condition A of LCO 3.4.6 when only one RCS loop is Operable and less than 7 hours has elapsed since entry into Mode 3 from Mode 1 or Mode 2. The added required action is to initiate action to restore a second required RCS loop to Operable status immediately.

2.3 Applicable Regulatory Requirements

The NRC staff reviewed the licensee’s request against the regulatory requirements and regulatory guidance documents described below.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," provide the regulatory requirements for the licensing of production and utilization facilities.

Section 50.36 of 10 CFR requires, in part, TSs to include the following categories related to station operation: (1) safety limits, limiting safety systems settings and control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls.

Section 50.36(c)(2) of 10 CFR states that LCOs are the lowest functional capability or performance level of equipment required for safe operation of the facility, and when LCOs are not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the LCO can be met.

Section 50.36(c)(2)(ii)(C) Criterion 3 requires that an LCO be established for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Section 50.36(c)(3) of 10 CFR states that SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, facility operation will be within safety limits, and the LCOs will be met.

10 CFR 50.120, states that each holder of an operating license shall establish, implement, and maintain a training program that provides for the training and qualification of nuclear power plant personnel.

Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50 establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety. According to Section 3.1.1 of the WBN, Unit 1, UFSAR, the WBN plant was designed to meet the intent of the "Proposed General Design Criteria for Nuclear Power Plant Construction Permits" published in July 1967. The WBN construction permit was issued in January 1973. The WBN plant, in general, meets the intent of the NRC GDC published as Appendix A to 10 CFR Part 50 in July 1971, as discussed in UFSAR Section 3.1.2.

> GDC 5, "Sharing of Structures, Systems, and Components," requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
>
> GDC 16, "Containment Design," requires that the containment and associated systems shall be provided to establish a leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

GDC 17, “Electric Power Systems,” requires that an onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety.

GDC 19, “Control room,” requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions.

GDC 34, “Residual Heat Removal,” requires that a system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

GDC 38, “Containment Heat Removal,” requires that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and to maintain them at acceptably low levels.

GDC 44, “Cooling Water,” requires that a system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these SSCs under normal operating and accident conditions. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 50, “Containment Design Basis,” requires, in part, that the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

RG 1.9, “Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units used as Class 1E Onsite Electrical Power Systems at Nuclear Power Plants.”

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-Of-Coolant Accidents (LOCAs).”

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 9.2.1, “Station Service Water.”

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 9.2.2, “Reactor Auxiliary Cooling Water System.”

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 10.4.9, “Auxiliary Feed Water System.”

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 13.2.1, “Reactor Operator Requalification Program; Reactor Operator Training,” Revision 3.

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 13.5.2.1, “Operating and Emergency Operating Procedures,” Revision 2.

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 16, “Technical Specifications,” Revision 3.

NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR (Light Water Reactor) Edition,” Chapter 18, “Human Factors Engineering,” Revision 2.

NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 4.

NUREG-1764, “Guidance for the Review of Changes to Human Actions,” Revision 1.

NUREG-0700, “Human-System Interface Design Review Guidelines,” Revision 2.

NUREG-0711, “Human Factors Engineering Program Review Model,” Revision 3.

NUREG-0737, “Clarification of TMI [Three-Mile Island] Action Plan Requirements.”

3.0 TECHNICAL EVALUATION

For this evaluation, the NRC staff reviewed the information provided in the licensee's amendment request; the UFSAR; details provided by letters dated June 17, 2015 (Reference 8), and August 13, 2015 (Reference 9), to support the WBN, Unit 2, operating license application review; and licensee responses to requests for additional information (RAIs).

The NRC staff conducted an audit of the proposed changes during the periods of July 27 to July 31, August 3 to August 7, and August 25 to August 28, 2015, to verify information in licensee supporting analyses and to identify information to be docketed by the licensee to support the NRC staff's preparation of this safety evaluation (SE).

3.1 Component Cooling System and Essential Raw Cooling Water System

GDC 44 requires a cooling water system to remove heat from SSCs important to safety during normal operating and accident conditions with a loss of offsite power and a single failure. GDC 5 states that if multiple units share SSC, the sharing must not significantly impair their ability to perform their safety functions, including in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. The ERCW and CCS will be shared between units and are intended to satisfy these safety requirements with two trains of equipment for each system, each train capable of performing the systems' safety functions.

The proposed TSs provide extra ERCW flow for both Trains A and B and extra CCS flow for Train B, which serves both units. The NRC staff requested additional information regarding the peak heat removal and flow requirements for CCS and ERCW for the limiting condition of mitigating a DBA (LOCA with a loss of offsite power and single failure of Train A or B) and concurrently maintaining the opposite unit in Mode 4 with maximum decay heat. The licensee responded in its letter dated August 28, 2015, and provided tables of peak heat removal rates (in British Thermal units per hour (Btu/hr)) and required CCS/ERCW flows (in gallons per minute (gpm)) for both loss of Train A and loss of Train B power.

For dual-unit operation, loss of a power train would be the most significant single failure in a DBA scenario. Loss of Train A bounds loss of Train B due to the loss of operability of two of the three CCS heat exchangers during loss of Train A. For the case of a DBA and loss of Train A within 48 hours of an opposite unit shutdown, the minimum required CCS and ERCW flows and maximum heat loads for loss of Train A are reproduced in Tables 1-4.

The licensee developed computer models of its CCS and ERCW to calculate expected flow rates through system cooling loads. The calculated expected flow rates from the computer model must exceed the minimum flow rates for both the loss of Train A (listed in Tables 1 and 2, respectively) and loss of Train B. The computer program modelled the physical layout of the system, including pipe size, length, elevation, and piping components such as fittings, valves, heat exchangers, and pump curves. The program computed the pressure drop throughout the systems while determining expected flow rates through all pipe segments and cooling loads. The limiting parameters considered include loss of offsite power, loss of downstream dam, loss of either A or B power train and one unit in a LOCA and the other unit in hot standby.

Table 1 - CCS Flow Rate Required to Meet Peak Heat Removal Rate
(assuming SGs not in use for heat removal)

Item **per Unit**	Required CCS Flow (gpm)
RHR Heat Exchanger	5,000
Centrifugal Charging Pump	28
RHR Pump	10
Safety Injection Pump	15
Radiation Monitor	6
Containment Spray Pump	2
Total (Each Unit) Supplied by 2 Train B CCS Pumps	5,061
Total (Both Units) Supplied by 2 Train B CCS Pumps*	10,122

* An additional CCS pump powered by Train B, not included in Total CCS flow, is required to supply the CCS HX A for spent fuel pool (SFP) cooling

Table 2 – ERCW Flow Rate to Meet Peak Heat Removal Rate
(assuming SGs not in use for heat removal)

Item	Required ERCW Flow (gpm)
CCS HX C	9,200
CS (containment spray) Heat Exchanger	5,200
2 Diesel Generators	2,600
CCS HX A For SFP Cooling	1,370
* Other Safety-Related loads (LOCA unit)	2,372
* Other Safety-Related loads (Mode 4 unit)	1,267
Total Supplied by 3 Train B ERCW Pumps	22,009

* Other safety-related loads itemized in licensee letter, dated August 28, 2015.

Table 3 - CCS Peak Heat Removal Rate (Unit 1 in Mode 4 with Loss of Train A)

Item for Unit 1, 7 hours after shutdown	Maximum Heat Load (Btu/hr)
RHR HX	89,265,200
Centrifugal Charging Pump	66,760
RHR Pump	100,000
Safety Injection Pump	0
Containment Spray Pump	0
Radiation Monitor	0
Total (Unit 1)	89,431,960

Table 4 - CCS Peak Heat Removal Rate (Unit 2 in LOCA with Loss of Train A)

Item for Unit 2 at LOCA	Maximum Heat Load (Btu/hr)
RHR Heat Exchanger	54,800,000
Centrifugal Charging Pump	66,760
RHR Pump	100,000
Safety Injection Pump	46,000
Radiation Monitor	0
Containment Spray Pump	14,746
Total (Unit 2)	55,027,506

The licensee calculated multiple scenarios for combinations of both units in various modes. In Enclosure 2 to TVA's letter dated July 14, 2015, the licensee provided output tables of its model calculations for the cases associated with this amendment request. Specifically, tabulating the required flow rates for CCS and ERCW for the high core decay heat case associated with Unit 1 in Mode 4 at 7 hours after shutdown and Unit 2 in a LOCA with either a loss of Train A or Train B. The computer model outputs provided in the tables in the July 14, 2015, letter demonstrate that the expected ERCW and CCS flows through safety-related cooling loads will exceed the minimum flows required to perform the required safety functions. The minimum required flows provide acceptance criteria that will be verified by preoperational testing for Unit 2 to ensure compliance with GDC 5, and will be confirmed by inspection prior to the NRC's WBN, Unit 2, operating license decision.

The licensee proposed an applicability note for LCOs 3.7.16 and 3.7.17 to make the LCOs not applicable more than 48 hours after entry into MODE 3 from MODE 1 or 2. The applicability note is part of the LCO as defined in 50.36(c)(2)(i). The NRC staff requested additional information from the licensee to explain the basis for the 48-hour criteria. The requested information and TVA responses are provided in TVA's letter dated August 28, 2015, in response to RAI BOP-17. The licensee responded that after 48 hours, the non-accident unit core decay heat is sufficiently low (~ 56.7 MBtu/hr) that the normal CCS and ERCW, as specified in LCOs 3.7.7 and 3.7.8, can support the accident and non-accident units with any active single failure. The actual time delay before the normal CCS and ERCW can support the heat loads varies depending on system availability and the single active failure postulated. The most limiting time delay is 48 hours due to the availability of only one CCS pump aligned to CCS HX C. The NRC staff finds the applicability limit of 48 hours to be acceptable because the additional heat removal capability provided by the new LCO 3.7.16 and LCO 3.7.17 requirements is not required to manage the combined heat load from a postulated accident on one unit and the core decay heat from a shutdown unit after the initial 48-hour shutdown period.

In its response to RAI BOP-14-1, provided in its letter dated August 28, 2015, the licensee stated that the 89,265,200 Btu/hr heat load associated with the non-accident unit is applicable 7 hours after shutdown. This heat load was used in calculating the minimum required ERCW and CCS flowrate for the case with loss of Train A. However, in its letter dated June 17, 2015, the licensee stated the RHR system is normally placed in service 4 hours after reactor shutdown. Therefore, in BOP-14-1 the NRC staff also requested that the licensee explain how removal of residual heat from the non-accident unit is assured between hours 4 and 7 after shutdown where the means of heat removal could be limited to the RHR system.

The licensee responded in a letter dated September 3, 2015, with changes to LCO 3.4.6, "RCS Loops-MODE 4," as described in Section 2.2 of this SE. The licensee's summary statement of the bases or reasons for the two new notes described in Section 2.2 are stated in the proposed TS Bases that were submitted in the September 3, 2015, letter as described below.

> During the initial seven hours after reactor shutdown, the heat loads are at sufficiently high levels that the requirement of LCO 3.4.6 for one RHR loop in operation may not be sufficient to mitigate a design basis accident on Unit 2 and preclude a heat up of Unit 1. To assure that there would be adequate heat removal capability under all postulated conditions during the initial seven hours after unit shutdown, reliance on heat removal via RCS loops is required. After a unit has been shut down for greater than seven hours, a single RHR loop in operation provides adequate heat removal capability.
>
> Note 2 requires two RCS loops to be OPERABLE during the initial seven hours after entry into MODE 3 from MODE 2 or MODE 1 until decay heat and latent heat are within the capacity of the RHR System.
>
> Note 3 precludes entry into MODE 5 during the initial seven hours after entry into MODE 3 from MODE 2 or MODE 1. This ensures that heat removal capacity via RCS loops is retained until decay heat and latent heat are within the capacity of the RHR System.

As noted in Section 2.2 of this SE, the licensee also added a required action to Condition A of LCO 3.4.6 when only one RCS loop is Operable and less than 7 hours has elapsed since entry into Mode 3 from Mode 1 or Mode 2. The added required action is to initiate action to restore a second required RCS loop to Operable status immediately. The NRC staff finds the proposed changes to LCO 3.4.6 to be acceptable because they will ensure that the plant will not be placed in an operating condition in which the combined heat load due to an accident on Unit 2 during the initial shutdown period for Unit 1 exceeds the heat removal capability of the RHR, CCS, and ERCW systems. Additionally, the NRC staff finds the proposed changes to LCO 3.4.6 to be consistent with the format guidance of the Standard Technical Specifications (STS), NUREG-1431, Revision 4. Lastly, the NRC staff finds the amended TS meets the requirements of 10 CFR 50.36(c)(2)(i).

The licensee initially proposed Applicability Note b. to specify that LCOs 3.7.16 and 3.7.17 would not be applicable when Unit 2 had been shut down for more than 48 hours. The NRC staff requested additional information regarding whether the capabilities of the ERCW and CCS, as defined in LCOs 3.7.7 and 3.7.8 would be sufficient when Unit 2 was shut down for 48 hours and Unit 1 just entered Mode 4, 7 hours after shutdown. Specifically, in RAI BOP-19 the NRC staff requested the licensee to address the heat removal requirements for Unit 2 shutdown for 48 hours and Unit 1 having just reached Mode 4, assuming either a DBA LOCA/loss of offsite power (LOOP) in Unit 2, with loss of a power train, or a DBA LOOP with loss of a power train, 48 hours after Unit 2 shutdown. The licensee responded in its letter dated August 28, 2015, stating the adequacy of two ERCW pumps per train and one CCS pump per train for removal of all decay heat from both units for these scenarios could not be assured under all worst case conditions. Therefore, in its supplement dated September 3, 2015, the licensee revised the proposed LCOs 3.7.16 and 3.7.17 to remove the Applicability Note b. regarding the shutdown status of Unit 2.

LCOs 3.7.16 and 3.7.17 are applicable when Unit 1 has been shut down for 48 hours or less. The additional ERCW flow with a third ERCW pump for each train and additional CCS pump for CCS Train B provides adequate cooling capacity to mitigate an accident in Unit 2 and keep Unit 1 in Mode 4 or 5. However, in a letter dated January 22, 2015 (Reference 10), the licensee identified another event, involving a loss of offsite power with loss of an emergency power train during a simultaneous or near simultaneous shutdown and cooldown of both units that could exceed the capacity of the shared CCS and ERCW. This postulated event could lead to an unanticipated mode change, contrary to the requirements of LCO 3.0.4 which states that "When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time." By letter dated September 21, 2015, the licensee committed to revise the WBN CCS description document and applicable operating procedures to prevent placing both units in Hot Shutdown (Mode 4) simultaneously. This will ensure that one unit will remain in Mode 3, relying on SGs for decay heat removal, until decay heat in both units is reduced to within the cooling capacity of the ERCW and CCS.

The licensee provided its summary statement of the bases or reasons for the LCO 3.7.16 and LCO 3.7.17 Required Actions described in Section 2.2 of this SE in the associated TS Bases as described below.

Condition A:

> In MODE 4, if one CCS [or ERCW] train is inoperable, and the unit is required to be placed in MODE 5 to comply with Required Actions, action must be taken to place the unit in MODE 5 within 24 hours. When the Required Actions of an LCO direct the unit to be placed in MODE 5, either a loss of safety function has occurred or the Required Action and Completion Time for restoring a safety-related component has not been met. Therefore, it is prudent to place the unit in a condition of lower energy with a lower potential for a postulated event. In this Condition, the remaining OPERABLE CCS [or ERCW] train is adequate to perform the heat removal function. The 24 hour Completion Time is consistent with LCO 3.4.6, "RCS Loops - MODE 4," Required Action B.1 for the Condition of one required RHR loop inoperable and no RCS loops OPERABLE.

Condition B:

> In MODE 4, if one CCS [or ERCW] train is inoperable, and the unit is not required to be placed in MODE 5 to comply with Required Actions, actions are taken to verify LCO 3.4.6 is being met with two OPERABLE RCS loops with one loop in operation, and that the unit remains in MODE 4 (T_{avg} > 200°F). These

actions indicate 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. This Action is conservative to the Required Actions of LCO 3.4.6 when there are two OPERABLE 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. The Frequency of once per 12 hours ensures that the systems being relied on for heat removal are operating properly and are maintaining the unit in MODE 4. The 12 hour frequency is reasonable, considering the low probability of a change in system operation during this time period.

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. This Action is consistent with the Required Actions of LCO 3.4.6 Condition B (no OPERABLE RCS loops and one inoperable RHR loop).

Condition C:

In MODE 4, if two CCS [or ERCW] trains are inoperable, immediate action must be taken to restore one of the CCS trains to an OPERABLE status, as no CCS [or ERCW] train is available to support the heat removal function. Required Action C.1 is consistent with LCO 3.4.6, "RCS Loops - MODE 4," Required Action D.1 for the Condition of required RCS or RHR loops inoperable and no required RCS or RHR loop in operation.

Required Action C.1 is modified by two Notes. Note 1 indicates that all required MODE changes or power reductions are suspended until one CCS [or ERCW] train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition. Note 2 indicates that the applicable Conditions and Required Actions of LCO 3.4.6 be entered for RHR loops made inoperable by the inoperable CCS [or ERCW] trains. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

Condition D:

Required Action D.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.7, "RCS Loops - MODE 5, Loops

> Filled," be entered for RHR loops made inoperable by one or more inoperable CCS [or ERCW] train(s). This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.
>
> In MODE 5, if one or more CCS [ERCW] train(s) is inoperable, action must be initiated immediately to restore the CCS [ERCW] train(s) to an OPERABLE status to restore heat removal paths. The immediate Completion Time reflects the importance of maintaining the capability of heat removal.

The Commission's regulations at 10 CFR 50.36(c)(2)(ii)(C) require that a TS LCO must be established for each item meeting, among other things, the following Criterion:

> A structure, system or component [SSC] that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The CCS and ERCW are SSCs that meet this criterion. Accordingly, the regulations require LCOs appropriate for these systems. The licensee determined that additional LCOs were needed to address dual-unit operation and proposed the additional LCOs and Required Actions as described above.

The NRC staff finds the proposed LCOs, Required Actions, and Completion Times to be acceptable because they will ensure that the plant will be placed in an appropriate operating mode to ensure that heat removal capability will be maintained, or that actions will be taken to ensure prompt restoration of heat removal capability in the event required SSCs become inoperable. The NRC staff reviewed the Required Actions and Completion Times of proposed LCOs 3.7.16 and 3.7.17, and finds that the proposed Required Actions and Completion Times are consistent with other existing LCOs for the conditions specified and are consistent in format with the STS guidance. Based on its analysis, the staff concludes that the proposed LCOs 3.7.16 and 3.7.17 meet the requirements of 10 CFR 50.36(c)(2)(ii)(C).

The licensee also proposed SRs for the proposed LCOs. Proposed SRs 3.7.16.1 and 3.7.17.1 require verification that the required pump(s) have correct breaker alignment and power available once per 12 hours. Proposed SR 3.7.16.2 requires verification that two CCS pumps are aligned to the B CCS train once per 12 hours. The NRC staff finds these SRs reasonable because their performance will ensure that the power necessary to run the additional required pumps is available and that the required pumps are correctly aligned to provide the additional flow required through the Train B CCS heat exchanger. The 12 hour frequency is consistent with similar existing SRs and is sufficient to ensure that the required SSCs will be available to perform their required functions. Therefore, the proposed SRs associated with LCOs 3.7.16 and 3.7.17 are acceptable and meet the requirements of 10 CFR 50.36(c)(3) to ensure that the necessary quality of the systems is maintained and the LCOs will continue to be met.

3.2 Auxiliary Feedwater Sources

In its June 17, 2015, letter, the licensee stated that, as an alternative to realigning ERCW and CCS as described above, a second approach to resolving the high heat removal demand during the first 48 hours after a shutdown would be to maintain Unit 1 in Mode 3 or Mode 4 with decay heat being removed through the SGs for at least 48 hours. The NRC staff observed that for this approach to be viable, the feedwater source for the SGs would need to be available for a minimum of 48 hours.

The NRC staff requested in RAI BOP-11, that the licensee assess how clean water would be provided for this extended duration of 48 hours, since the CST has a limited supply of clean water. The licensee responded in its August 28, 2015, letter that the CST is sized to provide 7 hours of clean water (at least 200,000 gallons). This assumes maintaining operation in Mode 3 for 2 hours and then cooling down for 5 hours at 50 degrees per hour until RHR can be placed in service. At 7 hours post-trip, the required rate for AFW flow is 175 gpm. An estimate of the required storage capacity for the remaining 41 hours would result in the conclusion that approximately 430,500 gallons of clean water would be required for the AFW system to function (i.e., total of 630,500 gallons needed for a 48-hour duration). The licensee stated that this estimate is very conservative due to the decreasing demand for AFW as the core heat load decays over the 48-hour period. Specifically, at the 48-hour point, only approximately 110 gpm is required to satisfy the AFW demand. When the CST is intact the inventory available in the CST (at least 200,000 gallons) plus makeup from the make-up water plant and the demineralized water storage tank (DWST) containing approximately 500,000 gallons, maximum, is available for supplying clean water to the AFW system for a minimum of 48 hours. Therefore, the available capacity of the DWST and one CST, at its minimum TS level, equates to approximately 700,000 gallons, which is greater than the conservative estimate of 630,500 gallons needed to supply the AFW system for 48 hours.

In addition to the normal DWST makeup, portable trailers are capable of providing 200 gpm to the CST. No credit is taken for this additional water in the SEs of condensate storage or AFW. The unit specific condensate storage tank is not an ESF and is not seismically qualified, flood protected, or missile protected. Also, the supply from the make-up water tank and the demineralized water storage tank and associated piping are not ESFs and are not seismically qualified.

The capacity of the normal makeup system is sufficient to maintain CST inventory for extended continued operation in Mode 3 or Mode 4. For a normal extended operation in Mode 4, such as one in which the plant cannot rely solely on RHR to remove decay heat 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. The licensee stated that while the normal makeup to the CST is from the demineralized water system, makeup from the opposite unit CST can also be used. The licensee's response demonstrates that a minimum of 48-hour supply of clean makeup water is available for decay heat removal from non-safety-related/non-seismic qualified sources. In the event that these sources become unavailable due to an external event, the engineered safeguards source of makeup water is ERCW.

The NRC staff acknowledges the use of the opposite unit's CST for Unit 1 before Unit 2 is licensed. The NRC staff also acknowledges the use the CSTs, with the DWST and its water supply, can provide an adequate supply of clean water to AFW for a minimum of 48 hours, while the ESF source is the ERCW. The ERCW poor quality feedwater will be used during events when safety is the prime consideration and SG cleanliness is of secondary importance. In addition, the high pressure fire protection 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 SGs in the unlikely event of a flood above plant grade.

The two CSTs would normally be isolated from each other, with one CST dedicated to each unit. Cross-connecting CSTs is not in the licensing basis for dual-unit operation. In NUREG-0847, "Safety Evaluation Report – Related to the Operations of Watts Bar Nuclear Plant Unit 2," Supplement 23 (Reference 11), it is stated that each CST is intended to operate independently in support of one unit, and no credit is taken in the safety analyses for the ability to crosstie the CSTs. At this time, cross connecting CSTs has not been analyzed and, therefore, has not been approved for dual-unit operation. The NRC staff considers use of the opposite unit's CST in a cross-tie configuration for dual-unit operation to be outside the scope of this amendment.

In RAI BOP-11, the NRC staff requested the licensee to provide the design change package and 10 CFR 50.59 analysis for the addition of the AFWST to the plant as an AFW supply. The NRC staff also requested the licensee to provide the analysis for cross connecting the CSTs for dual-unit operation. The licensee was not able to provide the requested information and, accordingly, generated a Condition Report. The NRC staff considers use of the AFWST to augment the AFW supply to be outside the scope of this license amendment request.

TS Table 3.3.2-1, Engineered Safety Feature Actuation System Instrumentation, Item 6.f describes operability requirements for the AFW pump suction transfer on low suction pressure. In RAI BOP-10 the NRC staff noted that the applicable Modes for TS Table 3.3.2-1, Item 6.f are Modes 1, 2, and 3; whereas, TS 3.7.5, Auxiliary Feed Water System, TS 3.7.6, Condensate Storage Tank, and TS 3.7.4, Atmospheric Dump Valves (ADVs), are applicable in Modes 1, 2, and 3, and Mode 4 when the SG is relied upon for heat removal. Because the ERCW is the ESF supply to AFW, the NRC staff requested the licensee in RAI BOP-10 to describe the bases for the difference in the Mode applicabilities, or make necessary changes to the Applicability of TS Table 3.3.2-1, Item 6.f.

The licensee in its September 3, 2015, letter proposed a change to the LCO Applicability for ESFAS instrumentation Function 6.f, "Auxiliary Feed Water Pumps Train A and B Suction Transfer on Suction Pressure-Low," to add "Mode 4 when the SGs are being relied on for heat removal." As a result of this change in Applicability, the licensee also proposed to change the applicable Condition to enter for an inoperable channel from Condition F to Condition B. Currently, Condition F requires an inoperable channel to be restored to an operable status in 48 hours or the unit placed in Mode 4. Condition B will require an inoperable channel to be restored to an operable status in 48 hours or the unit placed in Mode 5. Therefore, application of Condition B will result in removing the unit from the new Mode of Applicability, if an inoperable channel is not restored to an operable status in 48 hours.

The NRC staff finds the proposed changes to TS Table 3.3.2-1 to be acceptable as they conform to the guidelines in SRP 10.4.9, which necessitate automatic switchover to a safety-related water supply without interruption to AFW and meet the requirements of GDC 34 for a system to remove decay heat and other residual heat. Additionally, the NRC staff finds the proposed modification to TS Table 3.3.2-1 to be consistent with the STS guidance. Therefore, the NRC staff concludes the proposed changes to TS Table 3.3.2-1 are acceptable per 10 CFR 50.36(c)(2)(i).

3.3 Containment Performance

The added ERCW and CCS flow requirements associated with the proposed TS and the simultaneous sharing of CCS and ERCW while mitigating an accident in one unit and removing decay heat in the other unit necessitated completion of revised containment performance analyses to verify that the containments would continue to meet design requirements with the shared cooling system flows. A loss of Train A where CCS HX C will be the sole source of CCS to both units is the limiting case. The limiting case has revised values of UA (i.e., the product of the overall heat transfer coefficient of the heat exchanger and the effective heat transfer area) that affected the containment analyses. In completing its updated analyses, the licensee also corrected for other known errors and non-conservatisms in the current analyses of record (AOR) for each unit, as described in the following sections.

3.3.1 Unit 1 Containment Analysis

The LOCA containment mass and energy (M&E) release and containment integrity analysis is performed to confirm that the containment internal pressure, temperature, and the wall temperature do not exceed their design limits. The analysis should ensure that the containment heat removal system can remove the most limiting LOCA or MSLB M&E released into the containment without exceeding the WBN, Unit 1, containment internal design pressure of 15 psig or the administrative acceptance criterion of 13.5 pounds per square inch gauge (psig), and the wall design temperature of 250 °F. The analysis should also ensure that the sump temperature change does not adversely impact the Net Positive Suction Head (NPSH) available for the pumps that draw fluid from the sump during the LOCA recirculation phase.

The M&E release analysis calculated the M&E of the break fluid, which is an input to the containment pressure and temperature response calculation. The main reasons for revising the WBN, Unit 1, containment AOR include:

1. Correction of errors in the WCAP-10325-P-A (Reference 12) methodology reported in Westinghouse Nuclear Safety Advisory Letters (NSALs)-06-6, -11-5, and -14-2.

2. Resolution of an issue reported in Westinghouse InfoGram IG-14-1 (Reference 13), which states that the volumetric heat capacity value used for the RCS metal mass in the current M&E analysis did not bound the values published by the American Society of Mechanical Engineers (ASME). Using the ASME published data for the volumetric heat capacity would increase the stored energy of the RCS metal during normal plant operation, and consequently increases the M&E release into the containment during a design-basis LOCA.

3. Biasing of the containment compartments initial air temperature in the previous AOR for WBN, Unit 2 (FSAR Amendment 112), was non-conservative. It was determined that using the initial conditions specified in TS 3.6.5 resulted in higher containment pressure response and, therefore, is more conservative. A similar situation existed in the Unit 1 containment AOR. Therefore, the licensee revisited the initial containment compartment temperatures stated in assumption number (10) in Section 6.2.1.3.3 of the Unit 1 FSAR, and performed a sensitivity analysis to determine proper biasing of these temperatures, with respect to the TS 3.6.5 maximum limits, for a conservative containment pressure and temperature response.

4. Biasing of the ice bed temperature in the previous AOR for WBN, Unit 2 (FSAR Amendment 112), was determined to be non-conservative. A similar situation existed in the Unit 1 containment AOR. Therefore, the licensee revised the ice bed temperature of 15 °F stated In FSAR Section 6.2.1.3.3, under heading "Containment Pressure Calculation," assumptions (2) and (10) to the maximum value of 27 °F specified in SR 3.6.11.1.

5. The WBN, Unit 1, containment AOR did not consider sharing of the CCS and the ERCW flow with Unit 2 during the most limiting case of normal shutdown cooling in one unit and simultaneous design-basis LOCA in the second unit occurring prior to 48 hours from control rod insertion in the shutdown unit while it is in the hot shutdown mode, concurrent with a LOOP in both units. The sharing of these systems impacts the ERCW and CCS flows to the CCS HX and its cooling capacity during this scenario. Successful operation of the shared CCS and ERCW is required for compliance with GDC 5.

The licensee used the WCAP-17721-P (Reference 14) methodology for the design-basis LOCA M&E release analysis. This methodology, which is identified as WCOBRA/TRAC, has been recently approved by the NRC letter to Westinghouse (Reference 15) and is being applied for the first time for the WBN, Unit 1, LOCA M&E analysis. The licensee stated that the Unit 1 plant specific M&E analysis conforms to all the applicable limiting conditions for WCOBRA/TRAC required in the NRC SE (Reference 15). One of several features of this methodology is that during a LOCA, the initial stored RCS and SG energy is mechanistically released into the containment, which delays the ice bed meltout relative to the current AOR WCAP-10325-P-A (Reference 12) methodology at a given ice mass, and, therefore, results in a reduced calculated peak containment pressure. The licensee included changes in the LOCA M&E analysis assumptions related to: (a) initial RCS inlet temperature to the vessel/core, (b) fuel thermal performance, (c) secondary system volumes, and (d) RCS material properties. The licensee stated that the WCOBRA/TRAC LOCA M&E release analysis has resolved the issue reported in Westinghouse InfoGram 14-1 (Reference 13). The correction consisted of using the RCS metal volumetric specific heat that bounds the ASME published values. The Westinghouse NSALs-06-6, -11-5, and -14-2 were applicable to the current AOR, which used WCAP-10325-P-A methodology (Reference 12). The use of WCOBRA/TRAC eliminates all issues reported in these NSALs.

The containment peak pressure, temperature and sump fluid temperature occur during the long term LOCA post-reflood phase. The licensee used the revised M&E release output data for the analysis. The long term analysis consists of determining the containment pressure and temperature responses during the LOCA post-reflood phase and sump fluid temperature response during the LOCA sump recirculation mode. For this analysis, the licensee used the NRC approved AOR LOTIC1 ice condenser containment analysis methodology documented in WCAP-8354-P-A (Reference 16). The licensee analyzed the most limiting scenario of sharing of the CCS and ERCW flow with Unit 2 during its normal shutdown cooling and a simultaneous design-basis LOCA in Unit 1 occurring prior to 48 hours from control rod insertion in Unit 2 while it is in the hot shutdown mode, concurrent with a LOOP in both units. Sharing of these systems during this scenario impacts the ERCW and CCS flows to the CCS HX and its cooling capacity. Within this scenario, the analysis considered the most limiting case of loss of Train A concurrent with LOOP during which only one CCS HX C is available to be shared between the units. In this case, which has the most limiting heat load on the single CCS HX, the LOCA is assumed to occur in Unit 1 at 7 hours after the initiation of shutdown of Unit 2. At this instant the licensee calculated the CCS HX Unit 2 shutdown heat load of 89.4 MBtu/hr plus the Unit 1 LOCA containment heat load of 54.8 MBtu/hr. Because of sharing of the CCS HX between the units, this scenario results in a reduction of the following parameters of the CCS HX apportioned for containment heat removal in Unit 1 (LOCA Unit): (a) the AOR value of ERCW flow, and (b) the AOR value of UA, which is equal to the product of the overall heat transfer coefficient of the HX and the effective heat transfer area.

The licensee used the revised M&E release analysis results, the AOR ice mass of 2.26×10^6 pounds (lb) and the same assumptions and inputs as in AOR with the exception of conservative changes to: (a) nitrogen mass in the accumulators, (b) ice condenser volume temperature consistent with TS limits, (c) LOCA heat exchanger UA for core cooling, (d) decay heat model, (e) UA value of the CCS HX reduced for containment cooling because the HX is shared with Unit 2 for its shutdown cooling, and (f) ERCW flow to the CCS HX reduced for containment cooling because the flow is shared with Unit 2 for its shutdown cooling.

As stated in Reference 7, Enclosure 1, the licensee performed a sensitivity analysis for proper biasing of the assumed initial containment compartment temperatures to determine if the lower temperatures used in the AOR (lower and dead-ended compartment temperature of 100 °F, and the upper compartment temperature of 80 °F) or the higher values specified in TS 3.6.5 (lower and dead-ended compartment temperature of 120 °F, and the upper compartment temperature of 110 °F) give conservative containment pressure and temperature responses. The sensitivity analysis results show that the lower and dead-ended compartment temperature of 100 °F, and the upper compartment temperature of 80 °F give more limiting responses. The licensee stated that the 80 °F temperature in the upper compartment is a reduction from the TS 3.6.5 lower limit of 85 °F to account for the upper plenum volume of the ice condenser, which is included in the upper compartment volume for the analysis. The upper compartment volume is adjusted to maximize the air mass and the compression ratio. The analysis assumed all containment volumes are at a pressure of 0.3 psig and 10-percent relative humidity, except for 100-percent relative humidity in the ice condenser compartment.

The NRC staff noted during its audit that the licensee split the CCS HX C UA of 6.82×10^6 Btu/hr-°F into two virtual HXs; assigning one to the LOCA unit with a UA of 3.17×10^6 Btu/hr-°F and an ERCW flow of 3,400 gpm, and the second to the shutdown unit. In response to a request for additional information regarding the basis for the determination of the UA for the LOCA unit, the licensee stated that the split was based on approximately an equal CCS mass flow fraction to the virtual HXs.

The analysis results show that the peak containment pressure is 9.36 psig for the limiting design-basis LOCA (double-ended pump suction break) assuming an ice mass of 2.26×10^6 lb. The peak pressure occurs at approximately 8,959 second with ice bed melt-out at approximately 5,875 seconds and the CSS switchover time to sump recirculation mode at 2,718.7 seconds. The results are acceptable because the inputs and assumptions are conservative and the calculated peak containment pressure is less than the design pressure of 15.0 psig as well as the administrative limit of 13.5 psig, and the margin between the ice-bed melt-out time and spray switchover time (5,875 – 2,718.7 = 3,156.3 seconds) is greater than 150 seconds.

The NRC staff noted a significant difference in the heat loads on the two virtual CCS HXs (89.4 MBtu/hr on the shutdown unit versus 54.8 MBtu/hr on the LOCA unit). The NRC staff requested the licensee to re-analyze by splitting the real CCS HX (UA of 6.82×10^6 Btu/hr-°F) into two virtual HXs based on heat load fractions rather than equal CCS mass flow fractions. With the revised split, and with the proposed revised total ERCW and CCS flows of 9200 gpm and 10,000 gpm respectively, the licensee calculated a virtual UA of 2.64×10^6 Btu/hr-°F for the LOCA unit and a virtual UA of 4.12×10^6 Btu/hr-°F for the shutdown unit.

The WBN, Unit 1, containment response was performed with a virtual CCS HX UA of 3.17×10^6 Btu/hr-°F based on CCS mass flow, while based on apportioning by heat load its UA was calculated to be 2.64×10^6 Btu/hr-°F, as noted above. In its response to NRC requests for information (Reference 4), the licensee performed a sensitivity analysis by changing its UA from 3.17×10^6 Btu/hr-°F to 2.00×10^6 Btu/hr-°F (which bounds the calculated UA based on heat load fractions of 2.64×10^6 Btu/hr-°F) and used the same ERCW flow of 3,504 gpm. The sensitivity analysis (Reference 4) showed that lowering the virtual CCS HX UA to 2.00×10^6 Btu/hr-°F assigned for the LOCA unit (Unit 1) has no impact on the containment peak pressure. The RHR sprays do not get credited in the analysis because the calculated peak pressure stays below 9.5 psig, at which the RHR sprays would be initiated. The NRC staff finds the sensitivity study results acceptable since all of the containment heat removal is performed by the CSS and the air return fans.

The NRC staff finds the containment integrity analysis acceptable because: (a) the calculated design-basis LOCA peak containment pressure of 9.36 psig is bounded by the containment design pressure of 15 psig and the administrative limit of 13.5 psig, and is also below the 10 CFR 50 Appendix J Integrated Leak Test Pressure, P_a = 15 psig, which is same as the containment design pressure as stated in TS Section 5.2.7.19, and (b) the calculated maximum containment vapor temperature 234.3 °F (Table 4.5-6 Reference 1, Enclosure 1) is below the containment design temperature of 250 °F.

The licensee calculated 164.5 °F (Table 4.5-6 in Reference 1, Enclosure 1) as the maximum sump fluid temperature after switchover. As per FSAR Section 6.2.2.2, the pump NPSH available from the containment sump is calculated using the maximum credible sump water temperature 190 °F with no credit taken for containment accident pressure and the static height of water in the containment sump. Since the actual calculated maximum sump fluid temperature is less than the temperature assumed in the design analysis of NPSH available, the NPSH margin (NPSH available minus NPSH required) is conservative and bounded by the design margin.

Since the LOCA event requires the greatest amount of ice compared to other accidents and events, the analytical value of initial ice mass 2.26×10^6 lb based on LOCA results is acceptable for all other accidents and events.

3.3.2 Unit 2 Containment Analysis

In a letter dated August 13, 2015 (Reference 9), TVA submitted a revised WBN, Unit 2, containment analysis in support of the operating license review for WBN, Unit 2, to address dual-unit operation of the CCS and ERCW. The revised containment analysis also addressed corrections to the density and specific heat values used for the RCS metal mass.

The Unit 2 containment integrity analysis is performed to confirm that the containment internal pressure, temperature, and the wall temperature do not exceed their design limits. The analysis should ensure that the containment heat removal systems can remove the most limiting LOCA or MSLB M&E released into the containment without exceeding the WBN, Unit 2, containment internal design pressure of 15 psig or the administrative acceptance criterion of 13.5 psig, and the wall design temperature of 250 °F. The analysis should also ensure that the sump temperature change does not adversely impact the NPSH available for the pumps that draw fluid from the sump during the LOCA recirculation phase.

The analysis consists of an M&E release calculation, which is an input to the containment pressure and temperature response calculation. The applicant performed the M&E analysis in accordance with NUREG-0800, SRP 6.2.1.3. The purpose of revising the WBN, Unit 2, containment integrity AOR is due to:

- Correction of errors reported in Westinghouse InfoGram 14-1 (Reference 13) that increase the stored energy of the RCS metal during normal plant operation, and consequently increases the M&E release into the containment during a design-basis LOCA.

- Sharing of the CCS and the ERCW flow by WBN, Units 1 and 2, during a normal shutdown cooling in one unit and simultaneous design-basis LOCA in the second unit occurring prior to 48 hours from control rod insertion in the shutdown unit while it is in the hot shutdown mode, concurrent with a LOOP in both units.

The applicant analyzed the most limiting scenario of loss of Train A concurrent with a LOOP during which only one CCS HX (CCS HX C) is available to be shared between the units. In this scenario, which has the most limiting heat load on the single CCS HX, the LOCA is assumed to occur in the accident unit at 7 hours after the initiation of shutdown of the second unit. At this instant TVA calculated the CCS HX heat load as 89.4 MBtu/hr for the shutdown unit decay heat plus a heat load of 54.8 MBtu/hr for the LOCA unit containment. Because of sharing of the CCS HX between the units, this scenario results in a reduction of the following parameters of the CCS HX apportioned for containment heat removal in the LOCA unit: (a) the AOR value of ERCW flow, and (b) the AOR value of UA.

The applicant included the following other known changes in the containment integrity analysis: (a) initial RCS temperature uncertainty +7 °F, (b) 17x17 Robust Fuel Assembly-2 (RFA-2) fuel (which may, subject to separate future NRC licensing action, incorporate tritium-producing burnable absorber rods) for decay heat, and (c) containment spray flow control valve opening stroke time increased to +13 second. The applicant also considered the impact of ± 0.2 Hertz variation in the DG frequency and concluded it has negligible impact on the LOCA M&E results.

The applicant revised the LOCA M&E release AOR using the NRC approved Westinghouse WCAP-10325-P-A (Reference 12) methodology and resolved the issues reported in Westinghouse InfoGram 14-1 (Reference 13). The corrections consisted of using the revised RCS metal (stainless steel and low alloy carbon steel) density and specific heat that bounds the values given in the current ASME Boiler and Pressure Vessel (B&PV) Code 2010 Edition, Section II, Part D. The applicant used the RCS metal density value of 501 lbm/ft^3, which represents the density of stainless steel 304 and 316 at 70 °F and bounds the density (484 lbm/ft^3) of low alloy carbon steel at 70 °F given in Table PRD of ASME Code Section II, Part D. For conservative M&E release analysis, the applicant used density of stainless steel, (501 lbm/ft^3), which bounds the density of the bulk of the metal mass (carbon steel) in the reactor vessel and the SGs. The applicant stated that the SG tubes are not included because they are treated separately for a conservative LOCA M&E analysis. The ASME Code, Section II, Part D does not directly provide the metal specific heat values as a function of temperature, instead its Table TCD provides thermal diffusivity, which is related to thermal conductivity, density, and specific heat by the following equation:

Thermal Diffusivity = (Thermal Conductivity) / (Specific Heat x Density)

From the above equation, while adding a 10 percent uncertainty, the applicant derived a conservative value of the RCS metal specific heat of 0.145 BTU/lbm °F.

The NRC staff reviewed the ASME B&PV Code, Section II, Part D, and confirmed that the RCS metal density of 501 lbm/ft^3 and specific heat of 0.145 BTU/lbm °F are conservative for the M&E release analysis.

Resolving the issues reported in Westinghouse InfoGram IG-14-1 (Reference 13) impacts the M&E release during the LOCA blowdown phase. The NRC staff finds this change acceptable because an increase in density and specific heat of the RCS metal increases its stored sensible heat during normal plant operation and would, therefore, release greater M&E than the AOR during the LOCA blowdown phase.

The containment peak pressure, temperature and sump fluid temperature occur during the long term LOCA post-reflood phase. The applicant used the revised M&E release output data for the analysis. The long term analysis consists of determining the containment pressure and temperature responses during the LOCA post-reflood phase and sump fluid temperature response during the LOCA sump recirculation mode.

For the M&E analysis, the applicant used the same inputs and assumptions as stated in WBN Unit 2, FSAR Amendment 113, other than the changes noted above. The applicant replaced

Tables 6.2.1-16, 6.2.1-17, 6.2.1-18, 6.2.1-19 of WBN, Unit 2, FSAR Amendment 113 with revised M&E results tables in inserts C, D, E, and F in Reference 3, Enclosure 2.

For the containment response analysis, the applicant used the revised M&E analysis results and used the same assumptions and inputs as in WBN, Unit 2, FSAR Amendment 113 with the following exceptions: (a) the ice mass in the ice condenser was changed from 2.33×10^6 lb to 2.585×10^6 lb, (b) the UA value of the CCS HX was reduced for containment cooling because the HX is shared with Unit 1 for shutdown cooling, and (c) ERCW flow to the CCS HX was reduced for containment cooling because the flow is shared with Unit 1 for shutdown cooling.

The NRC noted during its audit that the applicant split the CCS HX C UA of 6.82×10^6 MBtu/hr-°F into two virtual HXs; assigning one to the LOCA unit with a UA of 3.17×10^6 MBtu/hr-°F and an ERCW flow of 3,400 gpm, and the second to the shutdown unit. In response to an NRC staff question regarding the basis for determination of the UA of 3.17×10^6 MBtu/hr-°F, the applicant stated that the split was based on approximately an equal CCS mass flow fraction to the virtual HXs.

The analysis results show that the peak containment pressure is 11.73 psig for the limiting LOCA (double-ended pump suction break) assuming an ice bed mass of 2.585×10^6 lb. The peak pressure occurs at approximately 3,600 second with ice bed melt-out at approximately 2,959 seconds and the CSS switchover time to sump recirculation mode at 2,718.7 seconds. The results are acceptable because peak calculated pressure is less than the design pressure of 15.0 psig as well as the administrative limit of 13.5 psig, and the margin between the ice-bed melt-out time and spray switchover time (2,959-2,718.7 = 240.3 seconds) is greater than 150 seconds.

The NRC staff noted a significant difference in the heat loads on the two virtual CCS HXs (89.4 MBtu/hr on the shutdown unit versus 54.8 MBtu/hr on the LOCA unit). The NRC staff requested the licensee to re-analyze by splitting the real CCS HX (UA of 6.82×10^6 MBtu/hr-°F) into two virtual HXs based on heat load fractions rather than equal CCS mass flow fractions. With the revised split, and with the proposed revised total ERCW and CCS flows of 9,200 gpm and 10,000 gpm respectively, the applicant calculated a virtual UA of 2.64×10^6 MBtu/hr-°F for the LOCA unit and a virtual UA of 4.12×10^6 MBtu/hr-°F for the shutdown unit.

Since the WBN, Unit 2, containment response described above was performed with a virtual CCS HX UA of 3.17×10^6 MBtu/hr-°F based on CCS mass flow, while based on apportioning by heat load its UA was calculated to be 2.64×10^6 MBtu/hr-°F, the licensee performed a sensitivity study by changing its UA from 3.17×10^6 MBtu/hr-°F to 2.00×10^6 MBtu/hr-°F (which bounds the calculated UA of 2.64×10^6 MBtu/hr-°F) and used the same ERCW flow of 3,504 gpm. The result showed an increase in peak containment pressure from 11.73 to 11.76 psig, which is not considered to be significant. The sensitivity study determined that the peak containment pressure is not sensitive to the UA of the virtual CCS HX assigned for the LOCA unit. The NRC staff finds the sensitivity study results acceptable because most of the containment heat removal is performed by the CSS compared to a small fraction removed by the RHR spray cooled by the CCS HX C.

The NRC staff finds the Unit 2 containment integrity analysis acceptable because: (a) the calculated design-basis LOCA peak containment pressure of 11.73 psig is below the containment design pressure of 15 psig and the administrative limit of 13.5 psig, and also below the 10 CFR 50 Appendix J Integrated Leak Test Pressure, P_a = 15 psig, which is the same as the containment design pressure as stated in TS Section 5.2.7.19, and (b) the calculated maximum containment vapor temperature 234.3 °F is below the containment design temperature of 250 °F.

The licensee calculated 157.5 °F (Table 4.4.3-6 in Reference 9, Attachment to Enclosure 1) as the maximum sump fluid temperature after switchover, which is a small increase from its current approximate value of 155 °F in WBN, Unit 2, FSAR Amendment 113 Figure 6.2.2-3. The increase in the peak sump fluid temperature increases the vapor pressure by 0.262 psi (0.6 ft water), which reduces the NPSH available by 0.6 ft at the inlet of the RHR and CSS pumps during the recirculation mode. The decrease in the NPSH available does not impact the operation of RHR and CSS pumps because the minimum NPSH margin (NPSH available minus NPSH required) for the most limiting pump given in FSAR Amendment 113 Table 6.3-12 is 6.4 ft. The most limiting NPSH margin is conservative because the applicant did not take credit for the sump static water level in the NPSH available calculation as stated in WBN 2 FSAR Amendment 113, Section 6.3.2.14 item (1).

Because the LOCA event requires the greatest amount of ice compared to other accidents and events, the analytical value of initial ice mass 2.585×10^6 lb based on LOCA results is acceptable for all other accidents and events.

3.4 Electric Power Systems

The proposed TS changes require two CCS pumps to be powered from Train B and aligned to the Train B CCS header to support operability of the RHR, such that if a LOOP and a loss of a power train occurs, the unit can continue to be cooled down. Additionally, a third ERCW pump per train would be made available that can be aligned to its respective Unit 1 6.9 kV shutdown board (i.e., 6.9 kV Shutdown Board 1A-A or 1B-B). The proposed changes would be applicable when Unit 1 has been shutdown from Mode 1 or Mode 2 for less than 48 hours.

In its application dated June 17, 2015, the licensee provided the following information regarding addition of interlock bypass switches to support operation of a third ERCW pump on each train:

> The ERCW System controls prevent the automatic loading of two ERCW pumps on a single DG. For each of the pairs of ERCW pumps powered from a 6.9 kV shutdown board, a pump selector switch allows the operations staff to choose which of the two pumps to have in service during normal operation. If one ERCW pump is in operation and powered by a DG, there is a second interlock that prevents the second ERCW pump from starting on that DG. These interlocks prevent the DG from being overloaded should an SI signal occur with the associated loading of the ECCS [Emergency Core Cooling System] pumps on the DG.

> Interlock bypass switches for the ERCW pumps are being added to each 6.9 kV shutdown board. These switches allow the operations staff to start a second ERCW pump on a DG, if necessary. The interlock bypass switches on the Unit 1 6.9 kV shutdown boards would be activated only in the event of a LOCA on Unit 2, concurrent with a LOOP [loss of offsite power] and a single failure that results in the loss of both 6.9 kV shutdown boards on a power train.

3.4.1 Offsite Power System

During normal plant operation, CSSTs C and D are dedicated for providing offsite power to the safety busses. Each CSST has the capacity to support dual-unit shutdown and, therefore, the proposed changes in this license amendment request will not exceed the allowable transformer loads. By letter dated August 1, 2013 (Reference 17), the licensee requested a revision to the WBN, Unit 1, licensing basis for use of CSSTs A and B as qualified offsite power sources. In view of the proposed loads (non-safety auxiliary loads and unit shutdown loads) when CSSTs A or B is used as an offsite power source, the NRC staff requested additional information on the impact of additional loads on CSST A and B. The NRC staff also requested clarification on DG loading and any proposed load shedding.

By letter dated August 3, 2015 (Reference 3), the licensee provided responses to the RAIs. In response to RAI Question Number 1 related to the capacity of CSSTs A and B, the licensee stated that, "With offsite power available, there is no change to the licensing basis documented in SSER 22." The NRC staff has previously reviewed WBN plant CSST A and CSST B loading as part of Supplemental Safety Evaluation Report (SSER) 22 (Reference 18) and found it acceptable. The licensee further clarified that the postulated loading when ERCW and CCS pumps are required in Mode 3, large safety-related loads such as the SI pump, CSS pump, and AFW pump will not be running. As a result, there is considerable margin compared to the limiting case previously analyzed. Therefore, the start of a second ERCW pump on a non-accident shutdown board is acceptable. The NRC staff reviewed the summary of large loads assumed in the analysis and concluded that CSST A and B have adequate capacity to start and run the proposed ERCW and CCS pumps on CSST A or B and maintain compliance with GDC 17 for capacity and capability of offsite power sources. The licensee has previously evaluated the offsite power sources assuming an accident on one unit and a spurious accident signal on the other unit. The loading in that scenario bounds the scenario for GDC 5 and the proposed changes in this amendment do not exceed the evaluated loads. The NRC staff finds that compliance with GDC 5 will be maintained with respect to the offsite power system.

3.4.2 Diesel Generator Loadings

As noted in the WBN, Unit 1, UFSAR Section 8.1.5.3, WBN, Unit 1, complies with RG 1.9, Revision 3, with specified exceptions. In particular regarding the DG loading, the licensee meets Position C1.3 of RG 1.9, Revision 2, which states the predicted loads should not exceed the short time rating of a DG.

The licensee provided details on DG loading for the scenario with Unit 1 being shutdown, Unit 2 in LOCA, LOOP for both units, and single failure of either Train A or Train B (only one train of DGs supporting the scenario).

In the amendment request, the licensee stated that during the postulated scenario, Unit 1 is not in an accident. Therefore, the SI and CSS pumps will not be running and will not be loaded to Unit 1 DGs. The Unit 1 motor-driven AFW pumps are not assumed to be running as a result of the event, because Unit 1 is being cooled by the RHR. In the supplemental information provided by letter dated August 3, 2015, the licensee clarified that before a second ERCW pump can be loaded on its DG, the AFW pump, if running, will be stopped and the main control room (MCR) hand switch placed in pull-to-lock. This action will assure that the AFW pump will not inadvertently start to preclude overloading the DG. In the DG loading calculations, the licensee assumed AFW pumps running for the first 20 minutes, and then stopped. The third ERCW pump is considered loaded after 20 minutes on the DG serving the shutdown unit.

The licensee provided DG loading information in the amendment request and supplements for the scenario: Unit 1 being shutdown, Unit 2 in LOCA, Loss of Train A or Train B. During the audit review process, the licensee also provided web-access to the DG loading analysis (Calculation No. EDQ00099920080014, Revision 31) performed by the licensee. Based on the review of the information provided by the licensee in the supplement dated August 28, 2015, the NRC staff finds that the loading of DGs will be as provided in Table 5 for the scenario under consideration.

The NRC staff finds that the maximum DG loading calculated as 4,313 kilowatt (kW) remains within the continuous service rating of 4,400 kW and within the 2 hours rating of 4,840 kW. Therefore, DG loading will be maintained within the guidelines provided in RG 1.9, Revision 2 and compliance with GDC 17 will also be maintained for capacity and capability of onsite power sources. Also, the addition of ERCW and CCS pumps on one DG will not adversely impact safe shutdown capability of dual units. The NRC staff, therefore, concludes that compliance with GDC 5 is maintained.

Table 5 - Diesel Generator Loading

Pumps	U2 LOCA / U1 Shutdown / Loss of Train A				U2 LOCA / U1 Shutdown / Loss of Train B				Notes
DG	1A	2A	1B (LOOP)	2B (LOCA)	1A (LOOP)	2A (LOCA)	1B	2B	
ERCW (rated HP 800)			1610 (2 pumps#)	805	1610 (2 pumps#)	805			# One pump for first 20 minutes until AFW pumps stopped.
CCS (rated HP 350)			378	720 (2 pumps)	720*** (2 pumps)	378			***378 hp until CCS Pump C-S is manually aligned after 2 hours for spent fuel cooling; then 720 hp.
AFW (motor-driven) (rated HP 300)			**	400* (2 pumps)	**	400* (2 pumps)			* 0 Min-2 Hrs 600 hp until SGs refilled; thereafter 400 hp. ** 0 Min-20 Min 300 hp; then stopped.
Containment Spray (rated HP 700)				596		596			
Centrifugal Charging (rated HP 600)			532	695	532	695			
SI (rated HP 400)				460		460			
RHR (rated HP 400)			370	440	370	440			
Total / Large Motor Load (HP)			2890	4116	3232	3774			
Pressurizer Heaters (kW)			500		500				
Total DG kW Loading: 0 - 20 Min			4188	4283	3984	4165			Based on Appendix N-1 of DG Loading Calc.
Total DG kW Loading: 20 Min – 2 Hours			3941	4313	3738	4201			Based on Appendix N-1 of DG Loading Calc.
Total DG kW Loading: 2 Hours - End			3941	4163	4015	4033			Based on Appendix N-1 of DG Loading Calc.

3.5 Operator Actions

One operator action outside of the MCR is being added. This action is for an Assistant Unit Operator to go to the appropriate 6.9 kilovolt (kV) shutdown board room and actuate one of the new ERCW pump interlock bypass switches.

Depending on the system alignments prior to the event and the electrical train that is lost as a part of the event, three other possible actions may be required of a MCR operator:

a. In the event of a loss of offsite power and the loss of Train A onsite power, a second CCS pump will be started, if two pumps are not already running. If CCS pump 2B-B was the second pump aligned to the CCS Train B header and it was not running, the operator would be required to start the pump. If CCS pump 2B-B was running or CCS pump 1B-B was the second pump aligned to the B header, no operator action would be required.

b. In the event of a loss of offsite power and the loss of either Train A or Train B power, a main control room operator would place the motor driven auxiliary feedwater pump, powered from the shutdown board on which the second ERCW pump was to be started, in pull-to-lock position, to ensure that it was not running and would not start.

c. In the event of a loss of offsite power and the loss of either Train A or Train B power, a main control room operator would start a second ERCW pump.

In accordance with the generic risk categories established in Appendix A to NUREG-1764, these actions are considered "risk-important" due to the fact that their failure could potentially complicate a LOCA by challenging the heat removal capability needed to put both units in a safe shutdown condition. Because of its risk importance, the NRC staff performed a "Level One" review (i.e., the most stringent of the graded reviews possible under the guidance of NUREG-1764).

3.5.1 <u>Operating Experience Review</u>

The licensee stated in its August 28, 2015, submittal that WBN performed an operating experience review for the proposed manual actions. The licensee's operating experience review included a search of the Institute of Nuclear Power Operations Operating Experience (OE) database and TVA's OE database for industry events associated with ERCW, CCS, and dual-unit operations. In addition, WBN benchmarked other TVA plants, Sequoyah and Browns Ferry, for lessons learned on dual-unit operations.

The licensee identified two common OEs associated with dual-unit operation: (1) plant transients due to operation of the wrong unit component and (2) mis-operation of the expected component.

The first OE does not directly apply to the change proposed in this amendment request because ERCW is common unit equipment, thus, there is no potential for operating the correct equipment on the wrong unit. Also, because the alignment of the second Train B pump would be performed from the shutdown unit and operated by the shutdown unit's staff, it is unlikely that this action would be performed on the wrong unit. The CCS pump would be started only in the case of a LOCA accompanied with a loss of Train A electrical power. In this case, the Train B CCS pump on the accident unit will start automatically, so there is no associated manual operator action.

The second OE also does not directly apply to the change proposed in this amendment request because the MCR staff would be alerted that the bypass switch has been placed in the "bypass" position by the corresponding MCR annunciator. If the field operator fails to correctly operate the appropriate bypass switch, the MCR staff would become aware of this fact when attempting to start the ERCW pump. The expected indications, such as breaker position lights, pump current, and pump discharge pressure and flow, would not be observed. This would cue the MCR staff to request that the field operator verify that the correct switch was operated and that it was placed in the "bypass" position.

Based on WBN's operating history of successful implementation of similar manual actions, and licensee's evaluation of relevant operating experience and its applicability to the changes proposed in this amendment request, the NRC staff finds the WBN operating experience review acceptable.

3.5.2 Functional Requirements Analysis and Function Allocation

The functions that must be performed to satisfy WBN's power generation and safety goals have not changed. Therefore, no additional functional requirements analysis or function allocation are necessary. The NRC staff finds this position to be acceptable.

3.5.3 Task Analysis

Task requirements were defined by the Operations team members. The only aspect of previously defined task requirements requiring reanalysis is the increase in operator workload in operating the bypass switch.

The licensee stated in its August 28, 2015, submittal that the new operator action (manual operation of an ERCW pump interlock bypass switch) is considered to be a simple action and the estimated time to perform the task, including switch positioning and access to the board room area, is minimal (i.e., 3 minutes and 35 seconds). If offsite power is supplied to the remaining electrical power train, additional manual operator actions are limited to starting the ERCW pump, Train B CCS pump (if required), and throttling CCS HX flows (if required). These actions can be performed by one operator in 2 minutes or less.

The NRC staff concludes that the additional workload to the operators will not prevent them from accomplishing this task and there is no additional support needed. The NRC staff finds this aspect of the proposed amendment to be acceptable.

3.5.4 Staffing

Staffing and qualifications of operating staff are not affected by the proposed amendment request. No new or additional crew members are required, nor are there any new or additional qualifications required to perform the action sequence within the time constraints established. The NRC staff finds this performance aspect to be acceptable.

3.5.5 Probabilistic Risk Analysis (PRA) and Human Reliability Analyses (HRA)

The licensee is not required to use, and did not use PRA or HRA methods to quantify the risk associated with the installation and use of the new interlock bypass switches. Therefore, there were no insights gained from the use of these analytical methods. The NRC staff finds this position to be acceptable.

3.5.6 Human-System Interface Design

The licensee stated in its July 14, 2015, submittal, as supplemented by the August 28, 2015, submittal that the following changes will be made to the Human-System Interface (HSI) design:

- Four interlock bypass switches for ERCW pumps will be installed, one on each 6.9 kV shutdown board; and
- One annunciator will be added in the MCR, on the common ERCW annunciator panel. The annunciator will alarm when an interlock bypass switch has been activated (i.e., placed in the "bypass" position).

The licensee stated in its August 28, 2015, submittal that each bypass switch will be used to bypass the interlock for the two ERCW pumps powered from that board. All bypass switches are identical in design and appearance, with the exception of identification and labeling. The licensee stated that the switches conform to the requirements for local workstation controls outlined in NUREG-0700. The "normal" (non-bypassed) and "bypass" switch positions are clearly labeled, in accordance with the labeling requirements in Technical Instruction TI-12.14, "Replacement and Upgrade of Plant Component Identification Tagging and Labeling." This labeling is consistent with other controls that the operating staff manipulate during routine evolutions. Standard, easily recognizable abbreviations are used on the switch, as defined in site procedure 0-TI-12.13, "Acronyms/Abbreviations Listing for Labeling."

The licensee has indicated that changes to the HSI are developed in accordance with the plant's design change process, as described in procedure NPG-SPP-09.3, "Plant Modifications and Engineering Change Control." Based on the licensee's statements, the NRC staff finds the proposed HSI design change process used for this plant modification acceptable.

3.5.7 Procedure Design

In its August 28, 2015, submittal, the licensee stated that it has a good understanding of the scope of the procedure revisions required, based on the procedures impact review performed in accordance with procedure NPG-SPP-09.3, "Plant Modifications and Engineering Change Control." The licensee provided a list of procedures that will be revised, based on the changes that need to be implemented as part of this amendment, in its August 28, 2015, submittal. Drafts of the proposed changes have been distributed for review to the appropriate licensee personnel. The procedures needed to implement this change will be issued prior to implementation of the approved amendment.

The guidance needed to realign lineups required in the CCS and ERCW is currently in place. Therefore, no future change to these procedures is required to implement these actions.

The NRC staff finds the plan to revise affected procedures acceptable based on the NRC staff's confirmation that the required actions will be described in controlled procedures, and will be feasible and reliable, based on TVA's validation and verification of the procedures per Technical Instruction TI-12.11, "Emergency Operating Instruction (EOI) Control" (see Section 3.5.9 below).

3.5.8 Training Program and Simulator Design

In its August 28, 2015, submittal, the licensee stated that Training Department representatives are part of the team that reviews Design Changes and amendment request impacts. In addition, one of the responsibilities of the Operations team member is to make a determination whether training is impacted and to inform the appropriate training management representative if training is required.

The training associated with this change will be performed in multiple formats. Relevant impacts to the plant will be included in each licensed operator requalification (LOR) cycle in the "Plant Changes" portion, which covers design changes, procedure changes, relevant industry events, and significant corrective action events. The training changes required for this change will be tracked in the corrective action program. The training needs analysis and changes to lesson plans are required to be complete by the next scheduled LOR training cycle. All training will be complete prior to the implementation of the approved amendment.

The simulator programming will be updated to allow a second ERCW pump to be started when the interlock bypass has been activated. An annunciator window that illuminates when the interlock bypass has been activated, will be added. The NRC staff finds the licensee's plans for training and changes to the plant-specific simulator acceptable.

3.5.9 Human Factors Verification and Validation

The licensee stated in the August 28, 2015, submittal that procedures are walked down 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 perform these walkdowns. The objective is to ensure that the procedure content provides the level of detail that is needed to successfully perform the action(s).

For the emergency operating network, the licensee uses Technical Instruction TI-12.11, "Emergency Operating Instruction (EOI) Control," which describes a more specific validation process. This instruction defines what validation method should be used, based on the change, the persons who should make up the validation team, how the validation is to be conducted, and how the validation is to be documented. In all cases involving the EOI network, a determination is made on whether the task can be performed by the minimum shift compliment.

The NRC staff finds that this process for verification and validation of operator actions is acceptable.

3.5.10 Human Performance Monitoring Strategy

The licensee stated in the August 28, 2015, submittal that existing plant processes would require review in accordance with 10 CFR 50.59, "Changes, test, and experiments," if a future modification required a change to the operation of the bypass switches. In addition, the licensee stated that it intends to perform routine monitoring of the bypass switches. 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. As a result, this will require the switches and their positions to be viewed locally by Nuclear Assistant Unit Operators (NAUOs) eight times each month. This frequent check, coupled with standard labeling and similarity of this task to those already performed by the NAUOs, ensure that this task can be successfully performed by the NAUOs. The NRC staff finds this long-term monitoring strategy to be acceptable.

3.6 Occupational Doses to Plant Operators

In its letter dated June 17, 2015, the licensee identified that during a DBA during dual-unit operations, operators may have to access the 6.9 kV shutdown boards located in close proximity to the MCR, in order to operate an ERCW pump interlock bypass switch. In its letter dated July 14, 2015, the licensee provided calculations, including plant layouts and access/egress paths, estimating the mission dose for the operators completing this vital mission. These calculations demonstrate that the operators can complete the actions necessary to mitigate the consequences of a DBA at WBN Unit 1 without exceeding a radiation dose of 5 rem.

3.7 Summary

The licensee has calculated the required heat removal rates and cooling water flow rates for the loads cooled by ERCW and CCS during a DBA in one unit and simultaneously maintaining cooldown in Mode 4 or 5 for the non-accident unit with the decay and sensible heat at 7 hours after shutdown. The calculations for the required ERCW and CCS cooling water flow rates considered separately a loss of both Train A and Train B power during the DBA.

Prior to 48 hours after shutdown of the non-accident unit, the required cooling water flow rates are greater than the current design flow and current TS LCOs for CCS and ERCW as specified in LCOs 3.7.7 and 3.7.8, respectively. To achieve the required ERCW flow, the licensee implemented a design change to make it possible to run a second ERCW pump on each emergency DG when needed, which would allow three ERCW pumps to be operable for each train.

A new LCO was proposed for ERCW requiring three operable ERCW pumps per train when shutdown 48 hours or less and in Mode 4 or 5. To achieve the required CCS flow, a new LCO was also proposed for CCS requiring a second operable CCS pump for Train B when shutdown 48 hours or less and in Mode 4 or 5. The licensee performed CCS and ERCW system pressure drop calculations to determine that the proposed TS LCOs provide at least the required extra

CCS and ERCW flowrates. The actual flows will be verified to meet the required flows by Unit 2 preoperational testing of ERCW and CCS.

Since a unit can be on RHR cooling within 4 hours after shutdown but the licensee used 7 hours after shutdown for computing the maximum decay heat load for the proposed TS LCOs, a revised TS LCO 3.4.6, "RCS Loops – MODE 4 was proposed that requires two operable RCS loops for the initial 7 hours after entry into Mode 3 from Mode 1 or Mode 2. The RCS loops assure adequate decay heat removal until RHR can assume the full decay heat load. The NRC staff finds that the addition of TS LCOs 3.7.16 and 3.7.17 and the changes to TS LCO 3.4.6 meet the requirements of GDC 5 and GDC 44 and the guidelines of SRP 9.2.1 and 9.2.2 for dual-unit operation.

The licensee could avert the need to enter the new proposed TS LCOs for ERCW and CCS by keeping a shutdown unit in Mode 3 for 48 hours when SGs are relied upon for heat removal. This would necessitate the need for makeup water to the SGs for 48 hours. The licensee has sufficient clean feedwater for at least 48 hours, but the sources are not seismically qualified, flood protected, or missile protected. However, if clean water sources were unavailable, the ERCW system would provide an adequate safety-related backup water supply to AFW. TS LCO 3.3.2 "ESFAS Instrumentation," did not require the automatic switchover to ERCW on low AFW suction pressure in Mode 4 when SGs are relied on for heat removal. Accordingly, TS Table 3.3.2-1, Engineered Safety Feature Actuation System Instrumentation, Item 6.f, which describes the AFW pump suction transfer on suction low pressure, was changed to include Mode 4 applicability when SGs are relied on for heat removal. The NRC staff finds that this change meets the requirements of GDC 34 and the guidelines of SRP 10.2.9 for dual-unit operation.

With respect to containment performance, the NRC staff concludes that the proposed changes continue to meet the requirements of: (1) GDC 5, because the sharing of the systems will not significantly impair their ability to perform their safety function of mitigating the increase in LOCA containment pressure and temperature in the LOCA unit; (2) GDC 16, because the containment design conditions important to safety are not exceeded during a design-basis LOCA in either unit; (3) GDC 38, because the containment heat removal system would reduce the containment pressure and temperature rapidly, following a design-basis LOCA and would maintain them at acceptable levels; and (4) GDC 50, because the containment heat removal system is designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from design-basis LOCA.

The NRC staff has reviewed the licensee's electrical system load analysis for offsite and onsite power systems. The NRC staff also reviewed the analysis for specific scenarios postulated in the amendment request and information provided in response to RAIs. Based on these reviews, the NRC staff concludes that there is reasonable assurance that the requirements of GDC 5 and GDC 17 will continue to be met.

The NRC staff has reviewed the licensee's proposed operator actions associated with the proposed changes. Based on the statements provided by the licensee that the proposed operator actions will be described in controlled procedures, that training will be updated, the HSI will be appropriately modified in accordance with the licensee's and industry human factors standards and guidance, these changes will be verified and validated, and they will be protected from inadvertent change by long-term monitoring, the NRC staff finds the licensee's proposed changes acceptable from the human performance perspective.

The NRC staff has reviewed the effect of the proposed changes on occupational doses to plant operators. Based on the statements and calculations provided by the licensee demonstrating that the WBN Unit 1 radiation shielding is sufficient to continue to allow operators to access important areas of the plant during accident conditions, the NRC staff finds the proposed changes acceptable.

The NRC staff concludes that the proposed TS LCOs, Required Actions, Completion Times, and SRs are consistent with STS guidance and meet the applicable requirements of 10 CFR Part 50, Appendix A, and 10 CFR 50.36. Therefore, the proposed changes provide reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The comment period for the NRC staff's proposed no significant hazards consideration determination expired on October 15, 2015. The NRC's regulations in 10 CFR 50.92 state that the NRC may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration (brackets [...] indicate portions of the licensee's analysis related to proposed changes consistent with TSTF-273-A that were withdrawn by the licensee, as described in Section 1.0 of this SE):

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The likelihood of a malfunction of any systems, structures or components (SSCs) supported by containment cooling system (CCS) and essential raw cooling water (ERCW) is not significantly increased by adding new technical specification (TS) for ERCW and CCS that require alternate CCS and ERCW system alignments during the first 48 hours after shut down of a unit when the steam generators are not available for heat removal. CCS and ERCW provide the means for transferring residual and decay heat to the Residual Heat Removal (RHR) System for process and operating heat from safety related components during a transient or accident, as well as during normal operation. Although the proposed change includes a design change to allow two ERCW pumps to be powered from one diesel generator (DG), the additional ERCW pump is only aligned to the DG on a non-accident unit during a design basis event on the other unit, and does not result in overloading the DG due to the reduced loading on the non-accident DG. The CCS and ERCW are not initiators of any analyzed accident. All equipment supported by CCS and ERCW has been evaluated to demonstrate that their performance and operation remains as described in the FSAR with no increase in probability of failure or malfunction.

The SSCs credited to mitigate the consequences of postulated design basis accidents remain capable of performing their design basis function. The change in CCS and ERCW system alignments has been evaluated to ensure the RHR System remains capable of removing normal operating and post-accident heat. Additionally, all the CCS and ERCW supported equipment, credited in the accident analysis to mitigate an accident, has been shown to continue to perform their design function as described in the FSAR.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

[...]

The proposed change to require the Reactor Coolant System (RCS) loops to be operable for the initial seven hours after shutdown and for the automatic switching of the auxiliary feedwater (AFW) pumps suction from the condensate storage tank (CST) to the ERCW System to be operable in Mode 4 when relying on steam generators for heat removal does not increase the probability or consequences of an accident that has been previously evaluated at WBN. The RCS loops are currently required to be operable to remove decay heat until plant conditions allow the RHR System to be placed in service. Specifying that the RCS loops are required to be operable for the initial seven hours after shutdown is consistent with the heat load assumptions at the specified time after shutdown described in the Updated Final Safety Analysis Report (UFSAR). The suction piping to the AFW pumps from either the CST or ERCW is not an initiator of any analyzed accident. The equipment supported by AFW and ERCW as described in the UFSAR has not been changed.

The SSCs credited to mitigate the consequences of postulated design basis accidents remain capable of performing their design basis function. The change requiring the RCS loops to be operable for the initial seven hours after shutdown does not affect heat removal capability. It ensures the RHR System is not solely relied on for decay heat removal before the decay heat load is within the capability of the RHR System. The change requiring the pressure switches in the AFW pump suction piping to remain in service in Mode 4 when steam generators are relied on to remove heat from the RCS does not affect heat removal capability. It retains the same automatic action required by the instruments in Modes 1, 2, and 3, consistent with the TS Applicability requirements for the AFW System.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not introduce any new modes of plant operation, change the design function of any SSC, or change the mode of operation of any SSC. There are no new equipment failure modes or malfunctions created as the affected SSCs continue to operate in the same manner as previously evaluated and have been evaluated to perform their safety functions when in the alternate alignments as assumed in the accident analysis. Additionally, accident initiators remain as described in the FSAR and no new accident initiators are postulated as a result of the alternate CCS and ERCW alignments.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

[...]

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not introduce any new modes of plant operation, change the design function of any SSC, or change the mode of operation of any SSC. There are no new equipment failure modes or malfunctions created as the affected SSCs continue to operate in the same manner as previously evaluated. Additionally, accident initiators remain as described in the UFSAR and no new accident initiators are postulated as a result of requiring the RCS loops to be operable for a specified duration after plant shutdown or by extending the Mode of Applicability of the AFW pump suction swap over from the CST to ERCW.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change continues to ensure that the cooling capability of RHR during normal operation and during the mitigation of a design basis event remains within the evaluated equipment limits and capabilities assumed in the accident analysis. The proposed change does not result in any changes to plant equipment functions, including setpoints and actuations. The proposed change does not alter existing limiting conditions for operation, limiting safety system settings, or safety limits specified in the Technical Specifications. The proposed change to add a new TS for ERCW and CCS assures the ability of these systems to support post-accident residual heat removal.

   Therefore, since there is no adverse impact of this change on the Watts Bar Nuclear Plant safety analysis, there is no significant reduction in the margin of safety of the plant.

   [...]

   The proposed change does not result in any changes to plant equipment functions, including setpoints and actuations. The proposed change does not alter limiting safety system settings or safety limits specified in the TS for these instruments. The proposed change ensures the decay heat load of the plant is within the capability of the RHR System prior to allowing sole use of the RHR loops for decay heat removal. In addition, the proposed change ensures the same automatic action to align ERCW as a supply source to AFW that occurs in Modes 1, 2, and 3 will remain available in Mode 4 when relying on the steam generators for decay heat removal. Thus, the proposed change does not reduce the margin of safety.

   Therefore, since there is no adverse impact of this change on the safety analysis, there is no significant reduction in the margin of safety of the plant.

The NRC staff reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). Based on this review, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a determination that no significant hazards consideration is involved for the proposed amendment.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on October 14, 2015. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in10 CFR Part 20. The NRC staff has determined that no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (80 FR 55383). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

8.0 REFERENCES

1. Letter from TVA to NRC, dated June 17, 2015, "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)" (ADAMS Accession No. ML15170A474).

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

3. Letter from TVA to NRC, dated August 3, 2015, "Watts Bar Nuclear Plant, Unit 1 - Response to Request for Additional Information Related to Application to Revise Technical Specifications for Component Cooling Water and Essential Raw Cooling Water to Support Dual Unit Operation" (ADAMS Accession No. ML15215A649).

4. Letter from TVA to NRC, dated August 28, 2015, "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" (ADAMS Accession No. ML15243A044).

5. Letter from TVA to NRC, dated September 3, 2015, "Revision to Watts Bar Nuclear Plant, Unit 1 Essential Raw Cooling Water and Component Cooling System License Amendment Request, Including Proposed Changes to Auxiliary Feedwater Pump Suction Transfer Instrumentation and Reactor Coolant System Loops – Mode 4" (ADAMS Accession No. ML15246A638).

6. Letter from TVA to NRC, dated September 21, 2015, "Response to Question Regarding GDC 5 Guidance for Dual Unit Shutdown" (ADAMS Accession No. ML15265A100).

7. Letter from TVA to NRC, dated September 21, 2015, "Response to NRC Question Regarding the Watts Bar Nuclear Plant, Unit 1 Revised Containment Analysis" (ADAMS Accession No. ML15266A026).

8. Letter from TVA to NRC, dated June 17, 2015, "Watts Bar Nuclear Plant, Unit 2 - Shutdown Technical Specifications for Component Cooling System and Essential Raw Cooling Water System to Support Dual Unit Operation" (ADAMS Accession No. ML15170A473).

9. Letter from TVA to NRC, dated August 13, 2015, "Revised FSAR Section 6.2.1, Containment Functional Design" (ADAMS Accession No. ML15225A382).

10. Letter from TVA to NRC, dated January 22, 2015, "Watts Bar Nuclear Plant, Unit 2 – Technical Specification Sections 3.0 and 3.10.1" (ADAMS Accession No. ML15023A187).

11. NUREG-0847, "Safety Evaluation Report – Related to the Operation of Watts Bar Nuclear Plant, Unit 2," Supplement 24 (ADAMS Accession No. ML11277A148).

12. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design," March 1979 Version (non-public).

13. Westinghouse InfoGram IG-14-1, dated November 5, 2014, "Material Properties for Loss-of-Coolant Accident Mass and Energy Release Analyses."

14. WCAP-17721-NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR LOCA Mass and Energy Release Calculation Methodology," April 2013 (ADAMS Accession No. ML13133A064).

15. Letter from NRC to Westinghouse Electric Company, dated August 24, 2105 , "Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report (TR) WCAP-17721-P, Revision 0, and WCAP-17721-NP, Revision 0, "Westinghouse Containment Analysis Methodology – PWR [Pressurized Water Reactor] LOCA [Loss-of-Coolant Accident] Mass and Energy Release Calculation Methodology" (ADAMS Accession No. ML15221A005).

16. WCAP-8354-P-A (Proprietary) and WCAP-8355-A (Non-Proprietary), "Long Term Ice Condenser Containment Code – LOTIC Code," April 1976.

17. Letter from TVA to NRC, dated August 1, 2013, "Application to Modify Watts Bar Nuclear Plant, Unit I Technical Specifications Regarding AC Sources - Operating (TS-WBN-13-02)" (ADAMS Accession No. ML13220A103).

18. NUREG-0847, "Safety Evaluation Report – Related to the Operations of Watts Bar Nuclear Plant Unit 2," Supplement 22.

Principal Contributors: G. Purciarello, A. Sallman, G. Matharu, V. Huckabay, D. Woodyatt, L. Wheeler, V. Goel, G. Lapinsky, R. Pedersen, P. Snyder

Date: October 20, 2015

October 20, 2015

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING APPLICATION TO REVISE TECHNICAL SPECIFICATIONS FOR COMPONENT COOLING WATER AND ESSENTIAL RAW COOLING WATER TO SUPPORT DUAL UNIT OPERATION (TAC NO. MF6376)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. NPF-90 for the Watts Bar Nuclear Plant, Unit 1. This amendment consists of changes to Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," and TS 3.4.6, "Reactor Coolant System Loops – MODE 4"; and the addition of new TS 3.7.16, "Component Cooling System – Shutdown," and new TS 3.7.17, "Essential Raw Cooling Water System – Shutdown," in response to your application dated June 17, 2015, as supplemented by letters dated July 14, August 3, August 28, September 3, and September 21, 2015.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Jeanne A. Dion, Project Manager
Watts Bar Special Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-390

Enclosures:
1. Amendment No. 104 to NPF-90
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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ADAMS Accession No.: ML15275A042 ***by Memo **by E-mail**

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DATE	10/15/2015	10/16/2015	10/07/2015	10/07/2015	10/13/2015	9/15 & 28/2015
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