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

September 14, 2017

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

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – ISSUANCE OF AMENDMENT REGARDING CHANGE IN TECHNICAL SPECIFICATION ICE MASS LIMITS (CAC NO. MF8993)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 14 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment consists of changes to the Technical Specification (TS) ice mass surveillance requirements (SRs) in response to your license amendment request dated December 21, 2016, as supplemented by your letter dated May 19, 2017.

The amendment revises TS SRs 3.6.11.2 and 3.6.11.3 to modify the requirements for the total weight of stored ice, minimum weight of each ice basket, and average ice weight of sample baskets. The amendment also makes conforming changes to TS Table SR 3.0.2-1.

A copy of the safety evaluation is also enclosed. A Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

The NRC staff has determined that its documented safety evaluation (Enclosure 2) contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has prepared a redacted, nonproprietary version (Enclosure 3). However, the NRC will delay placing the nonproprietary safety evaluation in the public document room for a period of 10-working days from the date of this letter to provide Tennessee Valley Authority the opportunity to comment on any proprietary aspects. If you believe that any information in Enclosure 3 is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10-working days, the nonproprietary safety evaluation will be made publicly available.

Document transmitted herewith contains Sensitive Unclassified Non-Safeguard Information in its Enclosure 2. When Separated from Enclosure 2, this document is decontrolled.

If you have any questions, please contact me at 301-415-6020 or Robert.Schaaf@nrc.gov.

Sincerely,

Robert G. Schaaf, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosures:
1. Amendment No. 14 to NPF-96
2. Safety Evaluation (Proprietary)
3. Safety Evaluation (Nonproprietary)

cc with Enclosures 1 and 3: Listserv (10 days after issuance of the amendment)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

WATTS BAR NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. NPF-96

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

   A. The application for amendment by Tennessee Valley Authority (TVA) dated December 21, 2016, as supplemented May 19, 2017, 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 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment, and paragraph 2.C.(2) of Facility Operating License No. NPF-96 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. 14 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 no later than 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Undine Shoop

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: September 14, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 14

WATTS BAR NUCLEAR PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. NPF-96

DOCKET NO. 50-391

Replace page 3 of Facility Operating License No. NPF-96 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.0-9	3.0-9
3.6-26	3.6-26

C. The 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 3411 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 14 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) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) PAD4TCD may be used to establish core operating limits for Cycles 1 and 2 only. PAD4TCD may not be used to establish core operating limits for subsequent reload cycles.

(5) By December 31, 2017, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved by License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

3.0 SR APPLICABILITY (continued)

Table SR 3.0.2-1

Surveillance Requirement (SR)	Description of SR Requirement	Frequency Extension Limit
3.6.6.4	Verify each containment spray pump starts automatically on an actual or simulated actuation signal	10/31/17
3.6.9.3	Verify each Emergency Gas Treatment System (EGTS) train actuates on an actual or simulated actuation signal	10/31/17
3.6.11.2	Verify total weight of stored ice is greater than or equal to 2,404,500 lb by: a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains greater than or equal to 1237 lb of ice; and b. Calculating total weight of stored ice, at a 95 percent confidence level, using all ice basket weights determined in SR 3.6.11.2.a.	10/31/17
3.6.11.3	Verify azimuthal distribution of ice at a 95 percent confidence level by subdividing weights, as determined by SR 3.6.11.2.a, into the following groups: a. Group 1-bays 1 through 8; b. Group 2-bays 9 through 16; and c. Group 3-bays 17 through 24. The average ice weight of the sample baskets in each group from radial rows 1, 2, 4, 6, 8, and 9 shall be greater than or equal to 1237 lb.	10/31/17
3.6.13.5	Visually inspect ≥ 95% of the divider barrier seal length, and verify: a. Seal and seal mounting bolts are properly installed; and b. Seal material shows no evidence of deterioration due to holes, ruptures, chemical attack, abrasion, radiation damage, or changes in physical appearance	10/31/17
3.7.7.3	Verify each Component Cooling System (CCS) automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal	10/31/17
3.7.7.4	Verify each CCS pump starts automatically on an actual or simulated actuation signal	10/31/17
3.7.8.2	Verify each Essential Raw Cooling Water (ERCW) automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal	10/31/17
3.7.8.3	Verify each ERCW pump starts automatically on an actual or simulated actuation signal	10/31/17

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.11.2	Verify total weight of stored ice is greater than or equal to 2,404,500 lb by: a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains greater than or equal to 1237 lb of ice; and b. Calculating total weight of stored ice, at a 95 percent confidence level, using all ice basket weights determined in SR 3.6.11.2.a.	18 months
SR 3.6.11.3	Verify azimuthal distribution of ice at a 95 percent confidence level by subdividing weights, as determined by SR 3.6.11.2.a, into the following groups: a. Group 1-bays 1 through 8; b. Group 2-bays 9 through 16; and c. Group 3-bays 17 through 24. The average ice weight of the sample baskets in each group from radial rows 1, 2, 4, 6, 8, and 9 shall be greater than or equal to 1237 lb.	18 months
SR 3.6.11.4	Verify, by visual inspection, accumulation of ice on structural members comprising flow channels through the ice bed is less than or equal to 15 percent blockage of the total flow area for each safety analysis section.	18 months

(continued)

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

1.0 INTRODUCTION

By application dated December 21, 2016 (Reference 1), as supplemented by a letter dated May 19, 2017 (Reference 2), the Tennessee Valley Authority (TVA or the licensee) requested changes to the Technical Specifications (TSs) for Watts Bar Nuclear Plant (Watts Bar or WBN), Unit 2. The requested changes would revise TS surveillance requirements (SRs) 3.6.11.2 and 3.6.11.3 to modify the requirements for the total weight of stored ice, minimum weight of each ice basket, and average ice weight of sample baskets. The amendment also makes conforming changes to TS Table SR 3.0.2-1.

The supplement dated May 19, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 28, 2017 (82 FR 15388).

2.0 REGULATORY EVALUATION

The U. S. Nuclear Regulatory Commission (NRC or the Commission) staff based its review of the effects of the proposed change on the containment analyses on the regulatory requirements, including Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50 Appendix A, General Design Criteria (GDC), described below.

Section 50.36 of 10 CFR requires, in part, that the TSs contain SRs. Section 50.36(c)(3) states that SRs to be included in the TSs are those relating to test, calibration, or inspection which assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the TS limiting conditions for operation will be met.

GDC 16, "Containment design," as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and assuring that the containment design conditions important to safety are not exceeded for as long as the postulated accident requires.

GDC 38, "Containment heat removal," as it relates to the containment heat removal system safety function which shall reduce rapidly, consistent with the functioning of other associated

Enclosure 3

systems, the containment pressure and temperature following any Loss-of-Coolant Accident (LOCA) and to maintain them at acceptably low levels.

GDC 39, "Inspection of containment heat removal system," as it relates to the containment heat removal system that its design shall permit appropriate periodic inspection of important components to assure the integrity and capability of the system.

GDC 40, "Testing of containment heat removal system," as it relates to the containment heat removal system that its design shall permit appropriate periodic functional testing to assure performance of the active components of the system, and the operability of the system as a whole, and, under conditions as close to the design as practical.

GDC 50, "Containment design basis," as it relates to the containment heat removal system which 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.

3.0 TECHNICAL EVALUATION

3.1 Description of Containment

Watts Bar Units 1 and 2 are Westinghouse designed pressurized water reactors with an ice condenser containment type. The following description of the primary containment applies to both Watts Bar Units 1 and 2.

The containment vessel is a freestanding steel structure made up of a vertical cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. The vessel 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, Steam Generators (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 is an insulated enclosure in which ice is stored.

3.2 Brief Description and Functions of the Ice Condenser

The ice condenser is an annular compartment enclosing approximately 300-degrees of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The lower portion has a series of hinged doors exposed to the atmosphere of the lower containment compartment, which, for normal plant operation, are designed to remain closed. At the top of the ice condenser is another set of doors exposed to the atmosphere of the upper compartment, which also remain closed during normal plant operation. Intermediate deck doors, located below the top deck doors, form the floor of a plenum at the upper part of the ice condenser. These doors also remain closed during normal plant operation. The upper plenum area is used to facilitate surveillance and maintenance of the ice bed. The ice condenser floor is a concrete structure containing embedded refrigeration system piping. Ice is maintained in an array of vertical cylindrical columns formed by perforated metal baskets with the space between columns forming the flow channels for steam and air. The ice inventory is contained in 1944 ice baskets in the ice condenser arranged in 24 bays.

The primary function of the ice condenser is to provide a heat sink in the event of release of energy from a postulated design basis 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. The ice baskets are arranged to enhance the ice condenser's function of condensing steam released to the containment during the accident. Another function of the ice bed is to become a large source of borated water, after it melts and appears in the containment sump, for the long term Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) heat removal functions during the recirculation phase of LOCA or MSLB. The third function of the ice condenser and the melted ice is to remove fission product iodine released during a LOCA. The iodine removal starts during the ice melt phase of the accident and continues as the sump water, obtained from the melted ice, is sprayed into the containment atmosphere by the CSS. After ice-melt, the containment pressure control is provided by the air return fan system, CSS, and the Residual Heat Removal (RHR) system spray train. The ice condenser limits the containment pressure below the design pressure of 15 pounds per square inch gauge (psig) for all reactor coolant pipe break sizes up to and including the largest double-ended guillotine pipe break of the Reactor Coolant System (RCS).

3.3 Current Analytical and Technical Specification Limits of Ice Mass

The current analytical (minimum) limit of ice mass stated in the Watts Bar dual unit Updated Final Safety Analysis Report (UFSAR) Section 6.7.6.1 is 2,585,000 pounds (lb) in the Unit 2 ice condenser. This limit is based on the following: (a) Mass and Energy (M&E) analysis using the NRC-approved Westinghouse Commercial Atomic Power (WCAP) Report WCAP-10325-P-A (Reference 3) methodology after correcting the errors reported in Westinghouse Nuclear Safety Advisory Letters (NSALs) 06-6, 11-5, 14-2 and InfoGram (IG)-14-1, and (b) the LOCA long-term containment integrity analysis, using the NRC-approved WCAP-8354-P-A (Reference 4) based on the LOTIC1 computer code. The current Watts Bar Unit 2 TS SR 3.6.11.2 ice mass of 2,750,700 lb includes additional ice mass to conservatively account for ice loss due to sublimation effects.

The current analytical (minimum) limit of ice mass stated in Watts Bar dual unit UFSAR Section 6.7.6.1 is 2,260,000 lb in the Unit 1 ice condenser. This limit was initially established in the license amendment request (Reference 5) for the replacement steam generators (RSGs) and was approved by the NRC in Reference 6. The analytical limit was calculated based on the following: (a) M&E analysis using the NRC approved WCAP-10325-P-A (Reference 3) methodology without correcting the errors reported in NSALs 06-6, 11-5, 14-2 and IG-14-1, and (b) the LOCA long-term containment integrity analysis, using the WCAP-8354-P-A (Reference 4) based on the LOTIC1 computer code. By Reference 7, the licensee superseded the Reference 5 analysis by using the NRC-approved Westinghouse WCOBRA/TRAC WCAP-17721-P-A methodology (Reference 8) and the LOTIC1 computer code. The Reference 7 analysis used 2,260,000 lb as the input ice mass, which is the same as in the Reference 5 analysis. The revised analysis resulted in a containment peak pressure and vapor temperature of 9.36 psig and 234.3 °F, respectively. Both of these values are bounded by the values of containment pressure and vapor temperature of 11.03 psig and 236.3 °F, respectively, obtained in the Reference 5 analysis using the WCAP-10325-P-A methodology. Because of the bounding values of the results, the licensee retained the current (Reference 5) analytical and TS limits 2,260,000 lb and 2,404,500 lb, respectively. The difference between the current TSs and analytical limits conservatively accounts for ice loss due to sublimation effects.

3.4 Proposed Changes

The licensee has proposed to revise the Watts Bar Unit 2 ice mass limits in TS SRs 3.6.11.2 and 3.6.11.3 to be identical with the ice mass limits in the Watts Bar Unit 1 TS SRs 3.6.11.2 and 3.6.11.3. The purpose of SR 3.6.11.2 is to verify the total weight of ice to ensure that it bounds the required TS value. To perform this SR, a representative sample of six baskets from each of the 24 ice condenser bays is picked with one basket from each of the radial rows 1, 2, 4, 6, 8, and 9. In case a basket from a designated row cannot be obtained for weighing, a basket from the same row of an adjacent bay is weighed. The licensee stated that the rows chosen include the rows nearest the inside and outside walls of the ice condenser (rows 1 and 2, and rows 8 and 9, respectively), where heat transfer into the ice condenser is most likely to influence melting or sublimation. The purpose of SR 3.6.11.3 is to verify that the azimuthal distribution of ice is reasonably uniform and that the average ice weight in each of three azimuthal groups of ice condenser bays is within the limit. The licensee also proposed to make conforming changes to TS Table SR 3.0.2-1.

3.5 NRC Staff Evaluation

The licensee has proposed to apply the Watts Bar Unit 1 LOCA containment integrity analysis to Watts Bar Unit 2 by performing a detailed comparison of the Unit 2 primary system, operating parameters, reactor protection features, ECCS capabilities, fuel design, containment design and analytical inputs to those modeled in the Unit 1 analyses. In Tables 1 and 2 of the Enclosure to Reference 1, the licensee provided a comparison of the input parameters for LOCA M&E and containment integrity analyses respectively. The NRC staff evaluation is provided below.

3.5.1 Comparison of Units 1 and 2 M&E Analysis Input Parameters

The following input parameters for Units 1 and 2 have the same values:

- core thermal power,
- RCS flow rate,
- RCS pressure,
- reactor vessel average temperature,
- low pressurizer pressure reactor trip,
- low pressurizer pressure safety injection setpoint,
- safety injection signal delay,
- accumulator water volume,
- accumulator nitrogen pressure,
- accumulator temperature, and
- pressurizer level

The following input parameters for Units 1 and 2 have minor differences that would not result in any significant difference in the M&E release:

- RCS pressure uncertainty,
- RCS temperature uncertainty,
- pressurizer level uncertainty,
- total injection flow, and
- total recirculation flow

Regarding the comparison of Unit 1 and Unit 2 fuel temperatures, the licensee stated:

> *The fuel temperatures used in the WBN Unit 1 analysis of record are based on PAD4+TCD (i.e., fuel thermal conductivity degradation was accounted for), and were calculated at a conservative burnup to maximize core stored energy. This method is consistent with NRC Condition 1 in the WCAP-17721-P safety evaluation (Reference 5 [included in Reference 9 of this safety evaluation]). Because WBN Unit 1 and WBN Unit 2 share the same fuel design, the WBN Unit 1 fuel modeling, with respect to fuel temperatures, bounds WBN Unit 2.*

The most significant difference is in the Watts Bar Unit 1 RSGs and the Watts Bar Unit 2 SGs. The following table provides a comparison of the input parameters related to the SGs:

Parameter	WBN Unit 1 RSG	WBN Unit 2 SG
SG design	Model 68AXP	Model D3
SG dry mass (lb)	760,187	704,500
Heat transfer area (ft^2)	68,000	48,000
SG water side mass including 10-percent conservatism (lb)	139,162	122,474
SG outlet pressure (psia*)	1090	1030

*pounds per square inch absolute

The input parameters of the SGs that could potentially affect the containment integrity analysis are the total dry metal mass, heat transfer area, and water side mass. For the Watts Bar Unit 1 RSGs, these parameters bound those for the Watts Bar Unit 2 SGs from the standpoint of M&E release.

Tables 1 and 2 of the Enclosure to Reference 1 did not provide a comparison of the mass and material properties of the following parameters for Watts Bar Units 1 and 2 that could affect the M&E release to the containment: (a) reactor internals, (b) reactor vessel, (c) RCS piping, and (d) RCS fluid. In an NRC staff request for additional information (RAI) (Reference 10), the licensee was requested to provide a quantitative comparison of the mass and material properties (density, specific heat, and thermal conductivity) of items (a) through (c), and the mass of item (d) for Watts Bar Units 1 and 2, and justify that the Watts Bar Unit 1 LOCA containment M&E release from these items bounds the Watts Bar Unit 2 M&E release.

In its response to the RAI (Reference 2), the licensee provided a quantitative comparison of the material mass and fluid volumes in the reactor vessel, reactor internals, RCS piping, and RCS pump (Table 1 in Reference 2). The licensee stated that the materials of construction for the reactor internals, reactor vessel, and RCS piping for Watts Bar Units 1 and 2 are identical and made of stainless steel, with the exception of the reactor vessel shell that is carbon steel. The small differences in the material mass and fluid volumes between Watts Bar Units 1 and 2 do not affect the model, with the exception of the reactor internals mass of Watts Bar Unit 2, which is higher than Watts Bar Unit 1 by 861 lb. However, because the reactor vessel shells are the same, this results in 1.72 cubic feet (ft^3) less fluid volume in the Watts Bar Unit 2 reactor vessel. These differences, excluding the SGs, represent small fractions of the RCS material mass of 1.72×10^6 lb and fluid mass of 339,000 lb (approximately 7850 ft^3).

The licensee also provided the values of material properties (density, specific heat, and thermal conductivity (Tables 2, 3, and 4 in Reference 2) used in the Watts Bar Unit 1 WCOBRA/TRAC model for the M&E analysis. The material properties are based on the 2010 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section II - Materials, Part D Properties. To account for uncertainty, the licensee increased both heat capacity and thermal conductivity by 10 percent, which is reasonable and conservative. The licensee obtained the density values from Table PRD of the 2010 ASME Boiler and Pressure Vessel Code, Section II - Materials, Part D Properties.

The licensee stated that Watts Bar Units 1 and 2 have the same RCS pipe diameters and thicknesses. The differences between the as-built piping configurations (such as in the allowances for field welds and about a 1-inch difference between the lengths of the crossover leg piping) are negligible because they are small fractions of the over 280 ft length of the RCS piping. The licensee stated that the metal energy and the fluid volume in the reactor coolant pumps for Watts Bar Units 1 and 2 is equal because their casing and the impeller are of the same material and design.

Based on the above information, the NRC staff finds that, excluding the SGs' metal material mass and fluid volume, the differences in Unit 1 and Unit 2 RCS material mass and fluid volume are negligible, and the conservative material properties used in the WCOBRA/TRAC LOCA M&E analysis for Watts Bar Unit 1 are bounding and applicable to Watts Bar Unit 2.

In a staff RAI (Reference 10), the licensee was requested to describe the modelling of the Main Steam Isolation Valves (MSIVs) for Watts Bar Unit 1 with respect to: (a) the starting time to close after event initiation, and (b) valve-specific time to fully close. In the event that items (a) and (b) differ between Watts Bar Units 1 and 2 due to differences in valve characteristics, different valve manufacturers, or both, the licensee was requested to provide a comparison of items (a) and (b) for Watts Bar Units 1 and 2, and justify that the Watts Bar Unit 1 M&E release based on these items would be bounding for Watts Bar Unit 2.

In its response to the RAI (Reference 2), the licensee stated that the MSIVs are assumed to **[[**

]].[1] The NRC staff finds the licensee's response acceptable because it is consistent with MSIV modelling documented in item 15 of Section 4.1 in the WCAP-17721-P-A (Reference 8) methodology and Sections 3.2.3 and 3.3.7.4 of the NRC Safety Evaluation (Reference 11).

Based on the overall comparison of the input parameters for the M&E release analysis, the NRC staff determined that the Unit 1 LOCA M&E release would bound the Unit 2 M&E release from the standpoint of calculating the analytical limit of the ice mass in the ice condenser.

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[1] The text between bolded brackets **[[ ]]** contains proprietary information.

3.5.2 Comparison of Input Parameters for Watts Bar Units 1 and 2 Containment Integrity Analysis

The licensee stated that Watts Bar Unit 2 has the same values of all of the input parameters used in the containment integrity analysis for Watts Bar Unit 1. These parameters are as follows:

- upper compartment volume,
- lower compartment volume,
- ice condenser volume,
- dead ended compartment volume,
- active sump volume,
- ice bed initial temperature,
- range of upper containment compartment air temperature,
- range of lower containment compartment air temperature,
- relative humidity for non-ice compartments,
- upper compartment return flow,
- containment air return fan delay time,
- Emergency Raw Cooling Water (ERCW) flow to the component cooling system (CCS) heat exchanger (single train),
- Component cooling water flow to RHR heat exchanger,
- Containment Spray (CS) flow rate (single train) in upper compartment during injection and recirculation phases,
- RHR spray flow rate (single train),
- ERCW flow to the CS heat exchanger (single train),
- RHR heat exchanger coefficient,
- CS heat exchanger coefficient,
- CCS heat exchanger coefficient,
- ultimate heat sink temperature,
- RHR containment spray actuation time,
- delay time for CS initiation,
- maximum refueling water storage tank temperature, and
- structural heat sink data and material properties

<u>Westinghouse Letter Regarding LOTIC1 Code Error</u>

As stated in Section 3.3 above, the current LOCA long-term containment integrity analysis for Watts Bar Units 1 and 2 is performed using the LOTIC1 computer code (WCAP-8354-P-A, Reference 4) with the M&E release input derived from the WCOBRA/TRAC methodology (Reference 8) for Unit 1 and WCAP-10325-P-A (Reference 3) for Unit 2. In a letter from Westinghouse (Reference 12), the NRC received the following information regarding an error in WCAP-8354-P-A (proprietary) and WCAP-8355-A (non-proprietary), "Long Term Ice Condenser Containment Code - LOTIC Code" (Reference 4):

> *A source code inspection revealed that the lower compartment conditions are not included until the end of the depressurization period. It has been determined that the affected portion of the transient is very short, and including the lower compartment conditions in the calculation would have a negligible impact on calculated containment*

conditions. Code updates regarding this issue would provide no improved transient behavior or influence on the limiting time of the event nor increase in nuclear safety.

The Westinghouse letter (Reference 12) provided the following change in Section 5.2, page 5.2-6 of Reference 4 for the calculation of the containment pressure profile during the phase between the end of blowdown and the establishment of recirculation:

<u>FROM</u>

[1] If the expanding volume is smaller than the lower compartment volume, the system pressure calculation is based on the upper compartment and the ice-filled part of the ice compartment.
[2] If the expanding volume occupies the lower compartment, the pressure calculation then includes the lower compartment conditions.
[3] If the expanding volume fills the lower compartment and the ice-empty part of the ice compartment, this calculation period is terminated.

<u>TO</u>

[1] If the expanding volume is smaller than the lower compartment volume <u>and ice-empty part of the ice compartment</u>, the system pressure calculation is based on the upper compartment and the ice-filled part of the ice compartment.
[2] ~~If the expanding volume occupies the lower compartment, the pressure calculation then includes the lower compartment conditions.~~
[3] If the expanding volume fills the lower compartment and the ice-empty part of the ice compartment, this calculation period is terminated.

The Westinghouse letter (Reference 12) did not provide a quantitative basis for the conclusion that the changes to correct the discrepancy between the description of the methodology and its implementation in the LOTIC code would have a negligible impact on the calculated containment response. In order to quantitatively confirm this conclusion, the NRC staff, in an RAI (Reference 10), requested the licensee to provide the following additional information for Watts Bar Units 1 and 2:

(a) The addition of "ice-empty part of the ice compartment" in item [1] above appears to be a significant change in the total volume (lower compartment volume + ice-empty volume) which is compared with the expanding volume, because the ice-empty volume varies from zero (or a small volume) to the full volume of the ice-compartment during the depressurization phase. Provide a quantitative impact of this change on the entire pressure response, including the peak pressure, by performing a sensitivity analysis for WBN Unit 1, and provide justification that the results apply to Unit 2.

(b) Describe the impact on the WBN, Units 1 and 2, containment pressure, containment temperature, and sump temperature responses and their peak values due to the deletion of item [2] "If the expanding volume occupies the lower compartment, the pressure calculation then includes the lower compartment conditions," and provide quantitative results."

The licensee provided the following response (Reference 2) to the RAI:

> *A temporary version of the LOTIC1 code was created to model the treatment of the lower compartment volume as it is currently described in WCAP-8354-P-A. The lower compartment conditions were considered in the system pressure calculation of this temporary LOTIC1 code version after the air bubble had expanded to fill the lower compartment. The current WBN Unit 1 containment model input deck for the peak pressure case was used with this temporary LOTIC1 code version to generate a transient pressure response for a sensitivity comparison with the current base case analysis. The following paragraph responds to the information requested in items (a) and (b) of the RAI.*
>
> *The calculated peak pressure from the base case analysis is 9.36 psig. The calculated peak pressure from the sensitivity case is 9.39 psig, which is only 0.03 psi higher. TVA considers this delta in the peak pressure to be negligible with respect to the entire pressure response. There were no changes to the peak lower compartment and sump temperatures due to the temporary code version. The peak upper compartment air temperature decreased by less than 0.5 °F.*

The licensee also provided Figures 1 through 4 in Enclosure 1 of Reference 2, which shows a comparison of the transient containment pressure, temperature, and sump temperature results for the current base case in the analysis of record (AOR) and the sensitivity case for Watts Bar Unit 1.

The NRC staff reviewed the comparison of the results of the AOR with the sensitivity case for the LOCA containment pressure, temperature, and sump temperature responses and finds that the difference is negligible. Because of the negligible difference, the Watts Bar Unit 1 AOR for containment response using the current version of the LOTIC1 code is acceptable. Also, because the input parameters and assumptions for Watts Bar Unit 1 for determining the containment pressure, temperature, and sump temperature responses for Watts Bar Units 1 and 2 are the same, the NRC staff accepts that the Watts Bar Unit 1 LOCA containment AOR is applicable to Watts Bar Unit 2.

3.6 Technical Conclusion

The NRC staff finds the licensee request for the proposed change to revise the Watts Bar Unit 2 ice mass limits in TS SRs 3.6.11.2 and 3.6.11.3 to be identical with the current ice mass limits in the Watts Bar Unit 1 TS SRs 3.6.11.2 and 3.6.11.3 to be acceptable. The following are the NRC staff conclusions for Watts Bar Unit 2 related to compliance with the applicable regulatory requirements:

GDC 16 requirements are met because the evaluation demonstrated that the performance of the containment in serving as a barrier to fission product release following an accident is not compromised.

GDC 38 requirements are met because the Ice Condenser System will continue to perform its design function of containment heat removal in response to accident conditions.

GDC 39 requirements are met because the proposed change will not impact the ability to inspect the containment heat removal system. These inspections will continue to be performed as required by the TSs in accordance with plant procedures.

GDC 40 requirements are met because the proposed change will not impact the ability for periodic functional testing of the containment heat removal system to assure performance of the active components of the system, and the operability of the system as a whole, and under conditions as close to the design as practical.

GDC 50 requirements are met because the Ice Condenser System will perform its design functions 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 accident.

The requirements of 10 CFR 50.36(c)(3) are met because the SRs will continue to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the TS limiting conditions for operation will be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on August 25, 2017. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on this finding published in the *Federal Register* on March 28, 2017 (82 FR 15388). 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.

6.0 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 to the health and safety of the public.

7.0 REFERENCES

1. Letter from TVA to NRC, "Application to Modify the Watts Bar Nuclear Plant Unit 2 Ice Mass Limit in Technical Specification Surveillance Requirements 3.6.11.2 and 3.6.11.3

(WBN-TS-16-026)," December 21, 2016 (Agency Documents Access and Management System (ADAMS) Accession No. ML16356A673).

2. Letter from TVA to NRC, "Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specifications Regarding Ice Condenser Containment Ice Mass Requirements for Watts Bar Nuclear Plant, Unit 2 (CAC No. MF8993)," May 19, 2017 (ADAMS Accession No. ML17139C939 (non-Proprietary), ML17139C940 (Proprietary, not publicly available)).

3. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version" (ADAMS Accession No. ML080640615, (Proprietary, not publicly available)).

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

5. Letter from TVA to NRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 – Technical Specification (TS) Change No. TVA-WBN-TS-05-09 – Ice Condenser Ice Weight Increase Due to Replacement Steam Generators," December 15, 2005 (ADAMS Accession No. ML053540054).

6. Letter from NRC to TVA, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Technical Specification Change to Increase Containment Ice Condenser Ice Weight to Support Replacement Steam Generators (TAC No. MC9270)," July 25, 2006 (ADAMS Accession No. ML061830005).

7. Letter from TVA to NRC, "Notification of Revised Westinghouse Containment Integrity Analysis," September 11, 2015 (ADAMS Accession No. ML15254A564).

8. WCAP-17721-P-A, Revision 0 (Proprietary) and WCAP-17721-NP-A, Revision 0 (non-Proprietary), "Westinghouse Containment Analysis Methodology - PWR [Pressurized-Water Reactor] LOCA Mass and Energy Release Calculation Methodology," September 24, 2015 (ADAMS Package Accession No. ML15272A050).

9. Letter from NRC to TVA, "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)," October 20, 2015 (ADAMS Accession No. ML15275A042).

10. E-mail from NRC to TVA, "Watts Bar, Unit 2 - Final Request for Additional Information Concerning Request to Amend Ice Mass Surveillance Requirements (CAC No. MF8993)," April 7, 2017 (Accession No. ML17100A107).

11. Final Safety Evaluation by the Office of Nuclear Reactor Regulation, 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" Westinghouse Electric Company (Westinghouse) Project No. 700, August 24, 2015 (ADAMS Accession No. ML15221A008 (non-Proprietary), ML15221A009 (Proprietary, not publicly available)).

12. Letter from Westinghouse to NRC, "Errata for WCAP-8354-P-A (Proprietary) and WCAP-8355-A (Non-Proprietary), "Long Term Ice Condenser Containment Code - LOTIC Code," February 1, 2017 (ADAMS Accession No. ML17034A376).

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Date: September 14, 2017

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – ISSUANCE OF AMENDMENT REGARDING CHANGE IN TECHNICAL SPECIFICATION ICE MASS LIMITS (CAC NO. MF8993) DATED SEPTEMBER 14, 2017

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