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

July 11, 2017

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 2 – ISSUANCE OF AMENDMENT REGARDING ONE-TIME EXTENSION OF INTERVALS FOR SPECIFIED SURVEILLANCE REQUIREMENTS (CAC NO. MF8895)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 13 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment consists of changes to the surveillance requirements in response to your license amendment request (LAR) dated November 23, 2016, as supplemented by letters dated February 16, 2017, and June 9, 2017.

The amendment revises the Technical Specification Surveillance Requirement (SR) 3.0.2 to allow a one-time interval extension for SRs listed in Attachments 5, 6, 7, 9, 12, 13, 14, 15, 16, and 17, to Enclosure 1 of your LAR. These SRs are normally performed on an 18-month frequency in conjunction with a refueling outage. The interval extensions of SRs listed in LAR Enclosure 1, Attachments 8, 10, and 11, were approved in Amendment No. 10 to Facility Operating License No. NPF-96 on April 7, 2017, and are not affected by this amendment.

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

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

Sincerely,

for

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. 13 to NPF-96
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

WATTS BAR NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
License No. NPF-96

1. The U.S. Nuclear Regulatory Commission (the Commission) has found, with respect to surveillance requirements listed in Attachments 5, 6, 7, 9, 12, 13, 14, 15, 16, and 17, to Enclosure 1 of the application, that:

   A. The application for amendment by Tennessee Valley Authority (TVA) dated November 23, 2016, as supplemented by letters dated February 16, 2017, and June 9, 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. 13 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 7 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

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

Attachment:
Changes to the Facility Operating
License and Technical Specifications

Date of Issuance: July 11, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 13

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
3.0-6	3.0-6
3.0-7	3.0-7
3.0-8	3.0-8
3.0-9	3.0-9
---	3.0-10
---	3.0-11

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. 13 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
3.3.1.13, Table 3.3.1-1, Function 15	Perform TADOT of the Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS) to Reactor Trip Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 1.b	Perform SLAVE RELAY TEST of the Safety Injection Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 2.b	Perform SLAVE RELAY TEST of the Containment Spray Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 3.a(2)	Perform SLAVE RELAY TEST of the Containment Isolation Phase A Isolation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 3.b(2)	Perform SLAVE RELAY TEST of the Containment Isolation Phase B Isolation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 4.b	Perform SLAVE RELAY TEST of the Steam Line Isolation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 5.a	Perform SLAVE RELAY TEST of the Turbine Trip and Feedwater Isolation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 6.a	Perform SLAVE RELAY TEST of the Auxiliary Feedwater Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.5, Table 3.3.2-1, Function 7.a	Perform SLAVE RELAY TEST of the Automatic Switchover to Containment Sump Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.7, Table 3.3.2-1, Function 1.b	Perform SLAVE RELAY TEST of the Safety Injection Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.7, Table 3.3.2-1, Function 3.a(2)	Perform SLAVE RELAY TEST of the Containment Isolation Phase A Isolation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.7, Table 3.3.2-1, Function 3.b(2)	Perform SLAVE RELAY TEST of the Containment Isolation Phase B Isolation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.2.8, Table 3.3.2-1, Function 1.a	Perform TADOT of the Safety Injection Manual Initiation Function	10/31/17
3.3.2.8, Table 3.3.2-1, Function 2.a	Perform TADOT of the Containment Spray Manual Initiation Function	10/31/17
3.3.2.8, Table 3.3.2-1, Function 3.a(1)	Perform TADOT of the Containment Isolation Phase A Isolation Manual Initiation Function	10/31/17
3.3.2.8, Table 3.3.2-1, Function 3.b(1)	Perform TADOT of the Containment Isolation Phase B Isolation Manual Initiation Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 1.c	Verify ESFAS RESPONSE TIMES are within limit for the Safety Injection Containment Pressure – High Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 1.d	Verify ESFAS RESPONSE TIMES are within limit for the Safety Injection Pressurizer Pressure – Low Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 1.e	Verify ESFAS RESPONSE TIMES are within limit for the Safety Injection Steam Line Pressure - Low Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 2.c	Verify ESFAS RESPONSE TIMES are within limit for the Containment Pressure – High High Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 3.b(3)	Verify ESFAS RESPONSE TIMES are within limit for the Containment Isolation Phase B Isolation Containment Pressure – High High Function	10/31/17

(continued)

3.0 SR APPLICABILITY (continued)

Table SR 3.0.2-1		
Surveillance Requirement (SR)	Description of SR Requirement	Frequency Extension Limit
3.3.2.10, Table 3.3.2-1, Function 6.b	Verify ESFAS RESPONSE TIMES are within limit for the Auxiliary Feedwater SG Water Level – Low Low Coincident with: 1) Vessel ΔT Equivalent to power ≤ 50% RTP With a time delay (Ts) if one SG is affected or A time delay (Tm) if two or more SGs are affected OR 2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0) Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 6.e	Verify ESFAS RESPONSE TIMES are within limit for the Auxiliary Feedwater Trip of all Turbine Driven Main Feedwater Pumps Function	10/31/17
3.3.2.10, Table 3.3.2-1, Function 7.b	Verify ESFAS RESPONSE TIMES are within limit for the Automatic Switchover to Containment Sump Refueling Water Storage Tank (RWST) Level - Low Coincident with Safety Injection and Coincident with Containment Sump Level - High Function	10/31/17
3.3.3.2, Table 3.3.3-1, Function 5	Perform CHANNEL CALIBRATION of the RCS Pressure (Wide Range) Function	10/31/17
3.3.3.2, Table 3.3.3-1, Function 6	Perform CHANNEL CALIBRATION of the Reactor Vessel Water Level Function	10/31/17
3.3.3.3, Table 3.3.3-1, Function 11	Perform TADOT of the Containment Isolation Valve Position Function	10/31/17
3.3.4.2, Table 3.3.4-1, Function 2.b	Verify each required control circuit and transfer switch is capable of performing the intended function for the Reactor Coolant System (RCS) Pressure Control Pressurizer Power Operated Relief Valve (PORV) Control and Pressurizer Block Valve Control Function	10/31/17
3.3.4.2, Table 3.3.4-1, Function 2.c	Verify each required control circuit and transfer switch is capable of performing the intended function for the Reactor Coolant System (RCS) Pressure Control Pressurizer Heater Control Function	10/31/17
3.3.4.2, Table 3.3.4-1, Function 3.b	Verify each required control circuit and transfer switch is capable of performing the intended function for the Reactor Coolant System (RCS) Pressure Control Pressurizer Heater Control Function	10/31/17

3.0 SR APPLICABILITY (continued)

Table SR 3.0.2-1

Surveillance Requirement (SR)	Description of SR Requirement	Frequency Extension Limit
3.3.4.2, Table 3.3.4-1, Function 4.b	Verify each required control circuit and transfer switch is capable of performing the intended function for the Decay Heat Removal via Steam Generators (SGs) AFW Controls Function	10/31/17
3.3.4.2, Table 3.3.4-1, Function 4.c	Verify each required control circuit and transfer switch is capable of performing the intended function for the Decay Heat Removal via Steam Generators (SGs) SG Pressure Indication and Control Function	10/31/17
3.3.4.2, Table 3.3.4-1, Function 5.a	Verify each required control circuit and transfer switch is capable of performing the intended function for the Decay Heat Removal via RHR System RHR Flow Control Function	10/31/17
3.3.4.3, Table 3.3.4-1, Function 2.b	Perform CHANNEL CALIBRATION for each required instrumentation channel for the Reactor Coolant System (RCS) Pressure Control Pressurizer Power Operated Relief Valve (PORV) Control and Pressurizer Block Valve Control Function	10/31/17
3.3.4.3, Table 3.3.4-1, Function 2.c	Perform CHANNEL CALIBRATION for each required instrumentation channel for the Reactor Coolant System (RCS) Pressure Control Pressurizer Heater Control Function	10/31/17
3.3.4.3, Table 3.3.4-1, Function 4.c	Perform CHANNEL CALIBRATION for each required instrumentation channel for the Decay Heat Removal via Steam Generators (SGs) SG Pressure Indication and Control Function	10/31/17
3.3.4.3, Table 3.3.4-1, Function 4.e	Perform CHANNEL CALIBRATION for each required instrumentation channel for the Decay Heat Removal via Steam Generators (SGs) SG Tsat Indication Function	10/31/17
3.3.6.5, Table 3.3.6-1, Function 2	Perform SLAVE RELAY TEST of the Containment Vent Isolation Instrumentation Automatic Actuation Logic and Actuation Relays Function	10/31/17
3.3.6.6, Table 3.3.6-1, Function 1	Perform TADOT of the Containment Vent Isolation Instrumentation Manual Initiation Function	10/31/17
3.4.12.8	Perform CHANNEL CALIBRATION for each required PORV actuation channel	10/31/17
3.5.2.5	Verify each ECCS 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
3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	10/31/17
3.6.3.6	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal	10/31/17
3.6.6.3	Verify each automatic containment spray 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

(continued)

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 2,750,700 lb by: a. Weighing a representative sample of ≥ 144 ice baskets and verifying each basket contains greater than or equal to 1415 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 1415 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

3.0 SR APPLICABILITY (continued)

Table SR 3.0.2-1		
Surveillance Requirement (SR)	Description of SR Requirement	Frequency Extension Limit
3.8.1.9	Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 66.75 Hz; b. Within 3 seconds following load rejection, the voltage is ≥ 6555 V and ≤ 7260 V; and c. Within 4 seconds following load rejection, the frequency is ≥ 59.8 Hz and ≤ 60.1 Hz.	11/30/17
3.8.1.10	Verify each DG operating at a power factor ≥ 0.8 and ≤ 0.9 does not trip and voltage is maintained ≤ 8880 V during and following a load rejection of ≥ 3960 kW and ≤ 4400 kW and ≥ 2970 kVAR and ≤ 3300 kVAR	11/30/17
3.8.1.11	Verify on an actual or simulated loss of offsite power signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected shutdown loads through automatic load sequencer, 3. maintains steady state voltage ≥ 6800 V and ≤ 7260 V, 4. maintains steady state frequency ≥ 59.8 Hz and ≤ 60.1 Hz, and 5. supplies permanently connected and auto connected shutdown loads for ≥ 5 minutes	11/30/17
3.8.1.12	Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each Unit 2 DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start and during tests, achieves voltage ≥ 6800 V and frequency ≥ 58.8 Hz; b. After DG fast start from standby conditions the DG achieves steady state voltage ≥ 6800 V and ≤ 7260 V, and frequency ≥ 59.8 Hz and ≤ 60.1 Hz. c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized from the offsite power system.	11/30/17
3.8.1.13	Verify each DG's automatic trips are bypassed on automatic or emergency start signal except: a. Engine overspeed; and b. Generator differential current	11/30/17

(continued)

3.0 SR APPLICABILITY (continued)

Table SR 3.0.2-1

Surveillance Requirement (SR)	Description of SR Requirement	Frequency Extension Limit
3.8.1.16	Verify each DG: a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation	11/30/17
3.8.1.17	Verify, DG 2A-A and 2B-B operating in test mode and connected to its bus, an actual or simulated ESF actuation signal overrides the test mode by: a. Returning DG to ready-to-load operation; and b. Automatically energizing the emergency load from offsite power.	11/30/17
3.8.1.18	Verify the time delay setting for each sequenced load block is within limits for each accident condition and non-accident condition load sequence.	11/30/17
5.7.2.4b	Perform integrated leak test for each system at least once per 18 months. Specifically, only the centrifugal charging pump injection portion of the safety injection system.	10/31/17

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

1.0 INTRODUCTION

By letter dated November 23, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16333A250), as supplemented by letters dated February 16, 2017 (ADAMS Accession No. ML17055B168), and June 9, 2017 (ADAMS Accession No. ML17163A231), Tennessee Valley Authority (TVA, the licensee), submitted a license amendment request (LAR) for Watts Bar Nuclear Plant (Watts Bar or WBN), Unit 2.

The LAR proposes to revise Technical Specification (TS) Surveillance Requirement (SR) 3.0.2 to extend, on a one-time basis, 52 specified SRs that are normally performed on an 18-month frequency in conjunction with a refueling outage. The proposed change extends the due dates for these SRs from their current due dates (i.e., 18 months from the last surveillance plus the 25 percent extension allowed by SR 3.0.2) to October 31, 2017, which would allow these SRs to be performed during the first refueling outage for Watts Bar, Unit 2. This will be accomplished by adding entries to existing TS Table SR 3.0.2-1 for each function of each extended SR.

Due to the impending due dates for certain SRs, the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff split its review into two separate groups. Group 1, which includes SRs from Attachments 8, 10, and 11 of LAR Enclosure 1, was reviewed in an April 7, 2017, safety evaluation (ADAMS Accession No. ML17074A501). Group 2 is comprised of SRs from Attachments 5, 6, 7, 9, 12, 13, 14, 15, 16, and 17 of Enclosure 1 to the LAR, and is reviewed in this safety evaluation.

By electronic mail dated May 11, 2017 (ADAMS Accession No. ML17135A028), the NRC staff sent the licensee a request for additional information (RAI) related to the Group 2 SRs. The licensee's June 9, 2017, response provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 17, 2017 (82 FR 4932), with respect to the Group 2 amendment.

2.0 REGULATORY EVALUATION

2.1 Background Related to the Proposed Amendment

TVA received the NRC-approved Facility Operating License No. NPF-96 for Watts Bar, Unit 2, on October 22, 2015. After receiving the operating license, startup began, and included completion of required TS SRs, and additional power ascension testing (PAT) to confirm the unit operated as designed. However, delays in startup caused a delay in commercial operation until October 19, 2016. This resulted in a delay of the first refueling outage to October 14, 2017.

2.2 Licensee's Proposed Changes

The licensee is proposing a one-time change to Watts Bar, Unit 2, TS SR 3.0.2, extending the test intervals for SR 3.3.1.13, Function 15; SR 3.3.2.5, Functions 1.b, 2.b, 3.a(2), 3.b(2), 4.b, 5.a, 6.a, 7.a; SR 3.3.2.7, Functions 1.b, 3.a(2), 3.b(2); SR 3.3.2.8, Functions 1.a, 2.a, 3.a(1), 3.b(1); SR 3.3.3.2, Function 5 and Function 6; SR 3.3.6.5, Function 2; SR 3.3.6.6, Function 1; SR 3.4.12.8; SR 3.5.2.5; SR 3.5.2.6; SR 3.6.3.6; SR 3.6.6.3; SR3.6.6.4; SR 3.6.9.3; SR 3.6.13.5; SR 3.7.7.3; SR 3.7.7.4; 3.7.8.2; SR3.7.8.3; and SR 5.7.2.4b to expire on October 31, 2017, so they may be accomplished during the rescheduled first refueling outage. This will be accomplished by including the affected SRs in an update to TS SR Table 3.0.2-1 with an SR due date of October 31, 2017. Changes to SR 3.0.2 will not be made specifically as proposed in the LAR because a recent Watts Bar, Unit 2, license amendment has resulted in acceptable similar SR 3.0.2 changes.

2.3 Regulatory Review

The regulatory requirements and guidance that the NRC staff considered in its review of the application are as follows:

- Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

- General Design Criteria (GDC) 13, "Instrumentation and control," requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate, to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

- GDC 16, "Containment design," requires that the reactor containment and associated systems shall be provided to establish an essentially leaktight 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 postulated accident conditions require.

- GDC 20, "Protective system functions," requires the protection system be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC 21, "Protection system reliability and testability," requires that the system be designed for high functional reliability and inservice testability, with redundancy and independence sufficient to preclude loss of the protection function from a single failure and preservation of minimum redundancy, despite removal from service of any component or channel.

- GDC 22, "Protection system independence," requires that the system be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions do not result in loss of the protection function.

- GDC 37, "Testing of Emergency Core Cooling System [ECCS]," requires that the ECCS be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components; (2) the operability and performance of the active components of the system; and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system; the transfer between normal and emergency power sources; and the operation of the associated cooling water system.

- GDC 40, "Testing of containment heat removal system," requires that *a* containment heat removal system be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components; (2) the operability and performance of the active components of the system; and (3) the operability of the system as a whole and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system; the transfer between normal and emergency power sources; and the operation of the associated cooling water system.

- GDC 44, "Cooling water," requires a system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components 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 46, "Testing of cooling water system," requires that the cooling water system be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

- GDC 54, "Piping systems penetrating containment," requires that piping systems penetrating primary reactor containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities, which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

- GDC 55, "Reactor coolant pressure boundary penetrating containment," requires each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

  (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

  (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

  (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

  (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

  Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

  Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

- GDC 56, "Primary containment isolation," requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

  (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

  (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

- GDC 57, "Closed system isolation valves," requires that each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve, which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

- 10 CFR 50.36 sets forth the requirements for the content of the TSs. This regulation requires, in part, that the TSs contain SRs. Specifically, 10 CFR 50.36(c)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

3.0 TECHNICAL EVALUATION

The NRC staff reviewed each of the proposed TS SR changes (from LAR Enclosure 1, Attachments 5, 6, 7, 9, 12, 13, 14, 15, 16, and 17) to ensure they do not impact compliance with regulations listed in Section 2.0 of this safety evaluation. The following are evaluations for each of these TS SR changes in the order they were presented in the LAR.

3.1 SR 3.3.1.13, Function 15 – Perform trip actuating device operational test (TADOT) on the reactor trip system instrumentation for the safety injection (SI) input from the engineered safety feature actuation system (ESFAS)

Attachment 5 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.3.1.13, Function 15. The manual SI reactor trip is tested during refueling outages with the integrated emergency diesel generator testing. The hand switch has both normally open and normally closed contacts that change state on operation to initiate a reactor trip. The normally closed contacts input to the reactor trip breaker (RTB) undervoltage coils and the normally open contacts input to the RTB shunt coils.

The SR extension applies to hand switches 2-HS-63-133A and 2-HS-63-133B.

SR 3.3.1.13, Function 15, has a due date of July 25, 2017 (including the 25 percent allowance of SR 3.0.2), which is 98 days before the requested surveillance extension date of October 31, 2017.

The licensee provided the following justification for allowing this interval extension. The licensee states that the design for the hand switch application using both normally closed and normally open contacts for end device operation provides an inherent reliability due to its simplicity of operation. The reliability is also assured with robust design of the breakers having both undervoltage and shunt trip coils. The licensee also performed an equipment history review of similar type hand switches used for ESFAS manual initiation on both units. For this review, the licensee examined its work orders for eight hand switches and determined there was only one actual failure (occurring on Unit 1), which occurred in 1997. This failure involved a containment vent isolation (CVI) Phase B hand switch, which is of similar design to that of the Unit 2 hand switches for which the one-time surveillance extension is requested. The specific failure was that contacts were found to be open with the switch in the actuate position. The hand switch was replaced and the function was re-tested satisfactorily.

Two other surveillance tests, SR 3.3.1.4 and SR 3.3.1.5, were cited as providing assurance of the operability of the RTBs, reactor trip breaker undervoltage and shunt trip mechanisms, and the reactor trip system automatic trip logic. However, neither SR 3.3.1.4 nor SR 3.3.1.5 actually test the hand switches themselves.

The first of these tests, SR 3.3.1.4, requires a TADOT to be performed on the RTBs, including the undervoltage and shunt trip mechanisms, every 62 days on a staggered test basis. The Watts Bar, Unit 2, SR 3.3.1.4 TS Bases state:

> The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. A note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The second test, SR 3.3.1.5, requires an actuation logic test to be performed on the reactor trip system every 92 days on a staggered test basis. The Watts Bar, Unit 2, SR 3.3.1.5 TS Bases state:

> The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through a semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection Function.

Based on the simplicity of the hand switch design and on the identified successful past performance of similar components, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.2 <u>SR 3.3.2.5 – Perform slave relay test;</u>
<u>SR 3.3.2.7 – Perform slave relay test on slave relays K603A, K603B, K604A, K604B, K607A, K607B, K609A, K609B, K612A, K625A, and K625B</u>

Attachment 6 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.3.2.5 and SR 3.3.2.7. The slave relay test involves energizing the slave relays and verifying operation of the end device. The slave relay tests verify the operability of the ESFAS instrumentation for the TS Table 3.3.2-1 functions identified below. This SR extension applies to the 18-month test frequency for Westinghouse type AR relays and Potter and Brumfield motor-driven relay (MDR) series relays identified in the above SRs. SR 3.3.2.5

also contains a note excluding those relays listed in SR 3.3.2.7 from the testing required by SR 3.3.2.5.

The slave relay tests verify the operability of the ESFAS instrumentation for the following functions of TS Table 3.3.2-1:

SR	Function	Description	Date Last SR Performed
3.3.2.5	1.b	SI automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	2.b	Containment spray (CS) automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	3.a(2)	Containment isolation (CI) Phase A isolation automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	3.b(2)	CI Phase B isolation automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	4.b	Steam line isolation automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	5.a	Turbine trip and feedwater isolation automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	6.a	Auxiliary feedwater automatic actuation logic and actuation relays	9/06/2015
3.3.2.5	7.a	Automatic switchover to containment sump automatic actuation logic and actuation relays	9/06/2015
3.3.2.7	1.b	SI automatic actuation logic and actuation relays	9/06/2015
3.3.2.7	3.a(2)	CI Phase A isolation automatic actuation logic and actuation relays	9/06/2015
3.3.2.7	3.b(2)	CI Phase B isolation automatic actuation logic and actuation relays	9/06/2015

The SR 3.3.2.5 and SR 3.3.2.7 functions listed above have a due date of July 24, 2017 (including the 25 percent allowance of SR 3.0.2), which is 99 days before the requested surveillance extension date of October 31, 2017.

Two other tests, SR 3.3.2.2 and SR 3.3.2.3, were cited as providing assurance of the operability of the ESFAS instrumentation.

The first of these tests, SR 3.3.2.2, requires an actuation logic test to be performed on the ESFAS instrumentation. The Watts Bar, Unit 2, SR 3.3.2.2 TS Bases state:

> The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils.

This surveillance is performed every 92 days on a staggered test basis.

The second test, SR 3.3.2.3, requires a master relay test to be performed on the ESFAS instrumentation. The Watts Bar, Unit 2, SR 3.3.2.3 TS Bases state:

> The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity.

This surveillance is performed every 92 days on a staggered test basis.

In response to RAI-MF8895-EICB-1.a, the licensee confirmed that SR 3.3.2.2 and SR 3.3.2.3 do not satisfy the requirements of SR 3.3.2.5 or SR 3.3.2.7, but do confirm the slave relay coils have not open circuited. The licensee provided approximately 5 years of Unit 1 data for SR 3.3.2.5 and SR 3.3.2.7 in response to RAI-MF8895-EICB-1.c. The data identified no failures for the performances of SR 3.3.2.5 and three documented failures for SR 3.3.2.7, which were then re-tested satisfactorily.

Based on the identified successful past performance of similar components, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.3 <u>SR 3.3.2.8 – Perform TADOT</u>

Attachment 7 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.3.2.8. A TADOT consists of operating the trip-actuating device and verifying the operability of required alarm, interlock, display, and trip functions. SR 3.3.2.8 is the performance of a TADOT utilizing manual hand switches in the main control room. This test is a check of the manual actuation functions. This SR is required to be performed every 18 months. The SR is modified by a note that excludes verification of setpoints during the TADOT for manual initiation functions. The manual initiation functions have no associated setpoints.

The licensee is requesting a one-time surveillance extension of SR 3.3.2.8 for the following TS Table 3.3.2-1 functions:

Function	**Description**	**Date of Last SR 3.3.2.8 Performed**
1.a	SI Manual Initiation	9/07/2015
2.a	CS Manual Initiation	9/06/2015
3.a.1	CI Phase A Isolation Manual Initiation	9/06/2015
3.b.1	CI Phase B Isolation Manual Initiation	9/06/2015

The SR 3.3.2.8 functions listed above have due dates of July 24, 2017, and July 25, 2017 (including the 25 percent allowance of SR 3.0.2), which are 98-99 days before the requested surveillance extension date of October 31, 2017.

The manual SI actuation is tested during refueling outages with the integrated emergency diesel generator testing. The manual CS actuation, Phase A actuation, and Phase B actuation are tested during refueling outages when both trains of the solid state protection system are removed from service and during the integrated emergency diesel testing.

This SR extension applies to the following hand switches:

- WBN-2-HS-063-0133A – SI Actuate
- WBN-2-HS-063-0133B – SI Actuate
- WBN-2-HS-30-64A – Phase B & CVI
- WBN-2-HS-30-64B – Phase B & CVI
- WBN-2-HS-30-68A – Phase B & CVI
- WBN-2-HS-30-68B – Phase B & CVI
- WBN-2-HS-30-63A – Phase A & CVI
- WBN-2-HS-30-63B – Phase A & CVI

Two other surveillance tests, SR 3.3.2.2 and SR 3.3.2.3, were cited as providing assurance of the operability of the ESFAS instrumentation. However, in response to RAI-MF8895-EICB-2.a, the licensee confirmed the hand switches are not tested under SR 3.3.2.2 or SR 3.3.2.3.

The first of these tests, SR 3.3.2.2, requires an actuation logic test to be performed on the ESFAS instrumentation. The Watts Bar, Unit 2, SR 3.3.2.2 TS Bases state:

> The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils.

This surveillance is performed every 92 days on a staggered test basis.

The second test, SR 3.3.2.3, requires a master relay test to be performed on the ESFAS instrumentation. The Watts Bar, Unit 2, SR 3.3.2.3 TS Bases state:

> The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity.

This surveillance is performed every 92 days on a staggered test basis.

As justification for allowing this interval extension, the licensee states that the design for the hand switch application using both normally closed and normally open contacts for end device operation provides an inherent reliability due to its simplicity of operation. The licensee also performed an equipment history review of similar type hand switches used for ESFAS manual initiation on both units. For this review, the licensee examined its work orders for eight hand switches and determined there was only one actual failure (occurring on Unit 1), which occurred in 1997. This failure involved a CVI Phase B hand switch, which is of similar design to that of the Unit 2 hand switches for which the one-time surveillance extension is requested. The specific failure was that contacts were found to be open with the switch in the actuate position. The hand switch was replaced and the function was re-tested satisfactorily.

Based on the simplicity of the hand switch design and on the identified successful past performance of similar components, the NRC staff determines there is reasonable assurance the Unit 2 components will remain operable during the extended surveillance interval period.

3.4 SR 3.3.3.2, Table 3.3.3-1, Function 5 – Perform channel calibration of the reactor coolant system (RCS) pressure (wide range) function;
SR 3.3.3.2, Table 3.3.3-1, Function 6 – Perform channel calibration of the reactor vessel water level function

Attachment 9 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for: SR 3.3.3.2, Table 3.3.3-1, Function 5, and SR 3.3.3.2, Table 3.3.3-1, Function 6. Function 5, RCS pressure (wide range), consists of three channels of instrumentation. Each channel consists of a pressure transmitter, Eagle 21 reactor protection system inputs and outputs, and pressure indication in the control room. This SR extension applies to the pressure transmitter in each channel. Function 6, reactor vessel water level, consists of two redundant trains of instrumentation. Each reactor vessel water level channel consists of one pressure transmitter, three level transmitters, Eagle 21 inputs and outputs, and a common-Q panel that serves as the display for all reactor vessel level instrumentation system (RVLIS) parameters. The rack portion of the system calibration is performed online. This SR extension applies to the pressure transmitter (included in Function 5), level transmitter, and output from Eagle 21 into the common-Q panel.

SR 3.3.3.2 was last performed on September 26, 2015, for the functions listed above and has a due date of August 13, 2017 (including the 25 percent allowance of SR 3.0.2), which is 79 days before the requested surveillance extension date of October 31, 2017.

One other surveillance test, SR 3.3.3.1, was cited as providing assurance of the operability of the RCS pressure (wide range) and reactor vessel water level instrumentation.

SR 3.3.3.1 requires a monthly channel check to be performed on the post-accident monitoring system instrumentation channels to verify deviations between redundant parameters are within allowable acceptance criteria. The Watts Bar, Unit 2, SR 3.3.3.1 TS Bases state:

> Performance of the channel check ensures that a gross instrumentation failure has not occurred. A channel check is normally a comparison of the parameter indicated on one channel to a similar parameter on another channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A channel check will detect gross channel failure; this it is key to verifying the instrumentation continues to operate properly between each channel calibration.

This surveillance is performed every 31 days.

In the LAR, the licensee provided the following justification for allowing this interval extension:

> A review of the most recent calibration history for WBN Unit 1 RVLIS level transmitters in 2015 indicated that the transmitters were found within the as-left tolerances. These level transmitters are the same make and model (Barton 764) for both units and demonstrate a high degree of reliability. Two of the three wide range pressure transmitters are the same make and model (Barton 763) for both units. The WBN Unit 1 transmitters were found within the as-left tolerance during the most recent calibration in 2015. The third pressure transmitter is a Weed transmitter that was last calibrated on October 26, 2015. The manufacturer's drift specification of 0.4% of the range (6000 psig) over 24 months (i.e., a drift of 24 psig) provides a significant margin over the TVA allowable drift, which is 35.6 psig over 22.5 months. This provides assurance that the transmitter will remain within drift limits during the surveillance extension period.

In response to RAI MF8895-EICB-3, the licensee provided Unit 1 calibration data for SR 3.3.3.2 Function 5 and Function 6 for approximately the last 3 years. The licensee noted the as-found data was either the same as the as-left data or was within their allowable tolerance criteria for the dates provided. In addition, the licensee provided Unit 1 and Unit 2 channel check results for the last 2 years, which were all completed satisfactorily.

Based on the demonstrated successful past performance of the affected components, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.5 <u>SR 3.3.6.5, Function 2 – Perform slave relay test of the CVI instrumentation automatic actuation logic and actuation relays</u>

Attachment 12 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.3.6.5, Function 2. SR 3.3.6.5 is the 18-month slave relay test for Westinghouse type AR and Potter and Brumfield MDR series relays. The slave relay test involves energizing the slave relays and verifying operation of the end device. The slave relay tests verify the operability of the ESFAS instrumentation for Function 2 of TS Table 3.3.6-1. This SR extension applies to the 18-month slave relay tests for slave relays K615A, K615B, K622A, and K622B.

SR 3.3.6.5, Function 2, was last performed on September 7, 2015, and has a due date of July 25, 2017 (including the 25 percent allowance of SR 3.0.2), which is 98 days before the requested surveillance extension date of October 31, 2017.

The licensee identified the following additional testing as providing assurance that the slave relays are functioning correctly through the extended surveillance interval:

- On February 8, 2016, TVA performed procedure 2-SI-90-8, "18 Month Channel Calibration Containment Purge Air Exhaust Rad Monitor Loop 2-LPR-90-131-B," which actuated a CVI on high radiation as part of the surveillance testing. This high radiation signal also actuated the K615 and K622 relays. This surveillance test verified CVI actuation via main control room alarms.

- On February 28, 2016, TVA performed procedure 2-SI-90-6, "18 Month Channel Calibration Containment Purge Air Exhaust Rad Monitor Loop 2-LPR-90-130-A," which actuated a CVI on high radiation as part of the surveillance testing. This high radiation signal also actuated the K615 and K622 relays. This surveillance test verified CVI actuation via main control room alarms.

- On June 5, 2016, an automatic SI and CI Phase A and CVI occurred on Watts Bar, Unit 2. This occurrence functionally tested the automatic actuation logic for the CVI SR via slave relays K615 and K622.

Based on the identified additional testing and the plant actuation that successfully exercised the automatic actuation logic via the slave relays, the NRC staff determined there is reasonable assurance these components are currently operable and will remain operable during the extended surveillance interval period.

3.6 SR 3.3.6.6, Table 3.3.6-1, Function 1 – Perform TADOT of the CVI instrumentation manual initiation

Attachment 13 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.3.6.6, Table 3.3.6-1, Function 1. The manual actuation hand switches are tested during refueling outages as part of the integrated Phase A, Phase B, CVI, and CS manual initiation test. The hand switches are redundant to the automatic initiation logic. The TADOT consists of operating the trip actuating device and verifying the operability of required alarm, interlock, display, and trip functions. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles). The SR is required to be performed every 18 months.

The SR extension applies to the following hand switches:

- 2-HS-30-63A
- 2-HS-30-63B
- 2-HS-30-64A
- 2-HS-30-64B
- 2-HS-30-68A
- 2-HS-30-68B

SR 3.3.6.6, Function 1, has a due date of August 24, 2017 (including the 25 percent allowance of SR 3.0.2), which is 68 days before the requested surveillance extension date of October 31, 2017.

As justification for allowing this interval extension, the licensee stated in the LAR that the design for the hand switch application using both normally closed and normally open contacts for end device operation provides an inherent reliability due to its simplicity of operation. The licensee also performed an equipment history review of similar type hand switches used for ESFAS manual initiation on both units. For this review, the licensee examined its work orders contained in its files for eight hand switches and determined there was only one actual failure (occurring on Unit 1), which occurred in 1997. This failure involved a CVI Phase B hand switch, which is of similar design to that of the Unit 2 hand switches for which the one-time surveillance extension is requested. This specific failure was that contacts were found to be open with the switch in the actuate position. The hand switch was replaced and the function was re-tested satisfactorily.

The licensee stated that no additional testing has been performed since SR 3.3.6.6 was last performed for Table 3.3.6-1, Function 1.

Based on the simplicity of the hand switch design and on the identified successful past performance of similar components, the NRC staff determined there is reasonable assurance that these components will remain operable during the extended surveillance interval period.

3.7 SR 3.4.12.8 – Perform channel calibration for each required power-operated relief valve (PORV) actuation channel

Attachment 14 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.4.12.8. This SR requires performance of a channel calibration on each required PORV actuation channel in order to adjust the entire channel so that the valve opens within the required range and accuracy. This SR extension applies to the channel calibration of each required PORV actuation channel. The SR is performed every 18 months to adjust the entire channel so that the valve opens within the required range and accuracy to known input.

SR 3.4.12.8 was last performed on September 27, 2015, and has a due date of August 14, 2017 (including the 25 percent allowance of SR 3.0.2), which is 78 days before the requested surveillance extension date of October 31, 2017.

No other surveillance test was cited as providing an alternate means of partially satisfying the TS SRs.

The licensee provided approximately 5 years of Unit 1 data for SR 3.4.12.8 in response to RAI-MF8895-EICB-4. The data identified no failures for the performances of SR 3.4.12.8 and identified that as-left data was either the same as the as-found data or was within the tolerance criteria.

Based on the identified successful past performance of similar components, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.8 SR 3.5.2.5 – Verify each ECCS 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;
SR 3.5.2.6 – Verify each ECCS pump starts automatically on an actual or simulated actuation signal

Attachment 15 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SRs 3.5.2.5 and 3.5.2.6. SRs 3.5.2.5 and 3.5.2.6 demonstrate every 18 months that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. The performance of SRs 3.5.2.5 and 3.5.2.6 can only be accomplished in Modes 5 and 6. The actuation logic is tested as part of ESFAS testing, and equipment performance is monitored as part of the inservice testing program. These SRs are not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control.

SRs 3.5.2.5 and 3.5.2.6 were last performed on September 6, 2015, and September 7, 2015, respectively. With the 25 percent SR 3.0.2 allowance, the SRs 3.5.2.5 and 3.5.2.6 due dates are July 24, 2017, and July 25, 2017, respectively, which are 99 and 98 days before the requested surveillance extension date of October 31, 2017.

The NRC staff reviewed the licensee's request and notes that the ECCS operated properly during the latest surveillance performances. In addition, related to SRs 3.5.2.5 and 3.5.2.6, the licensee stated in an enclosure to the LAR that on June 5, 2016, an automatic SI and subsequent reactor trip occurred. This functionally tested the automatic actuation logic for ECCS, containment isolation valve, containment spray system (CSS), emergency gas treatment system (EGTS), CCS, and emergency raw cooling water systems. All safety systems functioned as designed and as previously evaluated in the updated final safety analysis report. ECCS responded as expected to the automatic SI signal. The automatic actuation signal to ECCS was functionally tested on June 5, 2016. All other functions of the circuit were tested and were within frequency, which includes the end device operation of ECCS.

Based on the demonstrated successful past performance of the ECCS subsystems during the SI actuation, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.9 SR 3.6.3.6 – Verify each automatic CIV that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal;
SR 3.6.6.3 – Verify each automatic CS 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;
SR 3.6.6.4 – Verify each CS pump starts automatically on an actual or simulated actuation signal;
SR 3.6.9.3 – Verify each EGTS train actuates on an actual or simulated actuation signal;
SR 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;
SR 3.7.8.2 – Verify each essential raw cooling water 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;
SR 3.7.8.3 – Verify each essential raw cooling water pump starts automatically on an actual or simulated actuation signal

Attachment 15 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SRs 3.6.3.6, 3.6.6.3, 3.6.6.4, 3.6.9.3, 3.7.7.3, 3.7.7.4, 3.7.8.2, and 3.7.8.3. These SRs are intended to verify that the associated valves, dampers, pumps, and fans will start or reposition when an automatic initiation signal occurs (i.e., SI, containment isolation, or containment high-high pressure).

These SRs have due dates of either July 24, 2017, or July 25, 2017 (including the 25 percent allowance of SR 3.0.2), which are 98 or 99 days before the requested surveillance extension date of October 31, 2017.

Some of the components are in normally operating systems. Many of these components are subject to quarterly or more frequent operational exercise, which combine to demonstrate that the components and operating circuitry are largely intact and functional except for the automatic actuation relays and connecting wiring segments to the component control circuitry.

The June 9, 2017, response to NRC RAI-MF8895-SBPB-01 identifies components subject to these SR tests that automatically started or repositioned during the June 5, 2016, automatic SI actuation. Many of the components subject to these SRs effectively accomplished the intent of the SRs by their response to this event. In addition, the LAR indicated that none of the components subject to these SRs were noted as not performing as designed.

Based on the functionality of the associated components being checked by more frequent testing or exercise, and that many of these components had their proper response to an automatic initiation signal confirmed by the June 5, 2016, actual SI signal and subsequent reactor trip, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.10 SR 3.6.13.5 – Visually inspect ≥ 95 percent 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.

Attachment 16 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 3.6.13.5. The SR 3.6.13.5 inspection of the divider barrier seal has a normal interval of 18 months and is performed during refueling outages, since significant additional personnel dose and industrial hazard potential are present during plant operation at power. Barrier seal condition deterioration due to normal operating environmental conditions is slow, and disturbance by work activities or improper restoration after maintenance is almost always outage related.

SR 3.6.13.5 has a due date of September 19, 2017 (including the 25 percent allowance of SR 3.0.2), which is 42 days before the requested surveillance extension date of October 31, 2017.

The interval extension for SR 3.6.13.5 is acceptable since deterioration of barrier seal condition is expected to be minimal for such a short duration of plant operation.

The NRC staff determines there is reasonable assurance that deterioration of barrier seal condition will be minimal for such a short duration of normal plant operation at power.

3.11 SR 5.7.2.4b – Perform integrated leak test for each system at least once per 18 months. Specifically, only the centrifugal charging pump (CCP) injection portion of the SI system.

Attachment 17 of Enclosure 1 of the LAR discusses the proposed one-time SR frequency extension for SR 5.7.2.4b. SR 5.7.2.4b is associated with the integrated leak test of the primary coolant sources outside containment associated with CS, SI, RHR, chemical volume control system (CVCS), RCS sampling, and waste gas systems, and is performed to establish that diverting the CCP flow from the normal (or alternate) charging flow path to the high-head SI flow path can be performed as necessary or as required.

The last surveillance performance of SR 5.7.2.4b was on November 23, 2015, and has a due date of October 10, 2017 (including the 25 percent allowance of SR 3.0.2), which is 21 days prior to the surveillance extension date of October 31, 2017.

SR 5.7.2.4b testing is normally performed during Mode 6 (or defueled) in conjunction with similar tests where conditions have been established to allow diverting CCPs flow from the normal (or alternate) charging flow path to the high-head SI flow path.

The NRC staff reviewed the licensee's request and notes that the impacted systems including CS, RHR, and CCP operated properly during the latest performances for the surveillance. In addition, related to SR 5.7.2.4b, during the June 5, 2016, actual SI signal and subsequent reactor trip, the plant performed as designed. On receipt of the SI signal, the CCPs aligned for suction from the RWST, discharging through the boron injection tank (BIT) and into the RCS. Therefore, the affected section of pipe was pressurized to charging header pressure. The licensee stated in the LAR that the monthly ECCS venting had been conducted inside the BIT room that contains piping immediately upstream of the piping impacted by this test extension. Operations personnel access the area around the top of the BIT tank during the venting and there has been no leakage reported since the June 5, 2016, SI actuation.

Based on the demonstrated past leak integrity of the primary coolant sources outside containment associated with CS, SI, RHR, CVCS, RCS sampling, and waste gas systems, and specifically the performance of impacted systems including CS, RHR, and CCP during the SI actuation, the NRC staff determines there is reasonable assurance these components will remain operable during the extended surveillance interval period.

3.12 Technical Conclusion

Based on demonstrated successful past performance of the affected components, identified instrument performance data, performance of monthly channel checks, and additional testing or triggering data performed, the NRC staff concludes that the proposed one-time extension of test intervals for SR 3.3.1.13, Function 15; SR 3.3.2.5, Functions 1.b, 2.b, 3.a(2), 3.b(2), 4.b, 5.a, 6.a, 7.a; SR 3.3.2.7, Functions 1.b, 3.a(2), 3.b(2); SR 3.3.2.8, Functions 1.a, 2.a, 3.a(1), 3.b(1); SR 3.3.3.2, Function 5 and Function 6; SR 3.3.6.5, Function 2; SR 3.3.6.6, Function 1; SR 3.4.12.8; SR 3.5.2.5; SR 3.5.2.6; SR 3.6.3.6; SR 3.6.6.3; SR3.6.6.4; SR 3.6.9.3; SR 3.6.13.5; SR 3.7.7.3; SR 3.7.7.4; 3.7.8.2; SR3.7.8.3; and SR 5.7.2.4b, meets the requirements of 10 CFR Part 50, Appendix A, GDC 13, GDC 16, GDC 20, GDC 21, GDC 22, GDC 37, GDC 40, GDC 44, GDC 46, GDC 54, GDC 55, GDC 57, and 10 CFR 50.36(c)(3). Therefore, the NRC staff finds these one-time changes acceptable.

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 June 27, 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 determines 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 January 17, 2017 (82 FR 4932). 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.

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Date: July 11, 2017

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 – ISSUANCE OF AMENDMENT REGARDING ONE-TIME EXTENSION OF INTERVALS FOR SPECIFIED SURVEILLANCE REQUIREMENTS (CAC NO. MF8895) DATED JULY 11, 2017

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