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

April 15, 2024

James Barstow
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 165 AND 72 REGARDING INCREASE IN THE MAXIMUM NUMBER OF TRITIUM PRODUCING BURNABLE ABSORBER RODS AND SUPPORTING CHANGES, AND REVISION TO THE UPDATED FINAL SAFETY ANALYSIS REPORT (EPID L-2023-LLA-0039)

Dear James Barstow:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 165 to Facility Operating License No. NPF-90 and Amendment No. 72 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant (Watts Bar), Units 1 and 2, respectively. These amendments are in response to your application dated March 20, 2023.

The amendments revise Watts Bar, Units 1 and 2, Technical Specification 4.2.1, “Fuel Assemblies,” to increase the maximum number of tritium producing burnable absorber rods from 1,792 to 2,496. The amendments also revise Watts Bar, Unit 1 Technical Specification 5.9.6.b to be consistent with Watts Bar, Unit 2. The amendments revise both units’ Technical Specification 5.9.6.b to add a supporting reference. Lastly, the amendments revise the Watts Bar Updated Final Safety Analysis Report to modify the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an updated version of the ORIGEN code.

The request to revise the Watts Bar, Units 1 and 2, reactor vessel surveillance capsule removal schedule was previously reviewed and approved by the NRC staff by letter dated January 18, 2024.

A copy of our related safety evaluation for the amendments identified above is enclosed. A notice of issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Kimberly J. Green, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-390 and 50-391

Enclosures:
1. Amendment No. 165 to NPF-90
2. Amendment No. 72 to NPF-96
3. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
License No. NPF-90

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

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated March 20, 2023, 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 to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 165 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. Accordingly, the license is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated March 20, 2023. The licensee shall update the UFSAR to modify the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an updated version of the ORIGEN code, as described in the licensee's application dated March 20, 2023, and the NRC staff's safety evaluation attached to this amendment, and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance. The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

DAVID WRONA

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Date: 2024.04.15 10:10:33 -04'00'

David Wrona, 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: April 15, 2024

ATTACHMENT TO AMENDMENT NO. 165

WATTS BAR NUCLEAR PLANT, UNIT 1

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Replace page 3 of Facility Operating License No. NPF-90 with the attached revised page 3. The revised page is identified by amendment number and contain marginal lines indicating the area of change.

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

Remove Pages	Insert Pages
4.0-1	4.0-1
5.0-31	5.0-31

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

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

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 165 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

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

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

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

4.0 DESIGN FEATURES

4.1 Site

The Watts Bar Nuclear Plant is located on a tract of approximately 1770 acres in Rhea County on the west bank of the Tennessee River at river mile 528. The site is approximately 1-1/4 miles south of the Watts Bar Dam. The 1770 acres reservation is owned by the United States and is in the custody of TVA. The exclusion area is determined by a circle of radius 1200 meters centered on a point 20 feet from the north wall of the turbine building along the building centerline. The distance to the low population zone is a radius of 3 miles.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. For Unit 1, Watts Bar is authorized to place a maximum of 2496 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be either silver-indium-cadmium or boron carbide with silver indium cadmium tips as approved by the NRC.

(continued)

5.9 Reporting Requirements (continued)

5.9.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation (power operated relief valve lift settings required to support the Cold Overpressure Mitigation System (COMS) and the COMS arming temperature), criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

b. The analytical methods used to determine the RCS pressure and temperature limits and COMS setpoints shall be those previously reviewed and approved by the NRC, specifically, the analytical methods are described in the following references:

1. WCAP-14040-A, Rev. 4 “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.”

2. WCAP-18124-NP-A, Rev. 0, “Fluence Determination with RAPTOR-M3G and FERRET,” and WCAP-18124-NP-A Rev. 0 Supplement 1-NP-A, Rev. 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials," may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.

3. The PTLR will contain the complete identification for each of the TS reference Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

(continued)

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. 72
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, the licensee) dated March 20, 2023, 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 to this license amendment, 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. 72 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. Accordingly, the license is amended to authorize revision to the Updated Final Safety Analysis Report (UFSAR), as set forth in the application dated March 20, 2023. The licensee shall update the UFSAR to modify the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an updated version of the ORIGEN code, as described in the licensee's application dated March 20, 2023, and the NRC staff's safety evaluation attached to this amendment, and shall submit the revised description authorized by this amendment with the next update of the UFSAR.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance. The UFSAR changes shall be implemented in the next periodic update to the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

DAVID WRONA

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Date: 2024.04.15 10:11:20 -04'00'

David Wrona, 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: April 15, 2024

ATTACHMENT TO AMENDMENT NO. 72

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 revised page 3. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

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

Remove Pages	Insert Pages
4.0-1	4.0-1
5.0-34	5.0-34

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 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 72 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) FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, 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 in 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:

4.0 DESIGN FEATURES

4.1 Site

The Watts Bar Nuclear Plant is located on a tract of approximately 1770 acres in Rhea County on the west bank of the Tennessee River at river mile 528. The site is approximately 1-1/4 miles south of the Watts Bar Dam. The 1770 acre reservation is owned by the United States and is in the custody of TVA. The exclusion area is determined by a circle of radius 1200 meters centered on a point 20 feet from the north wall of the turbine building along the building centerline. The distance to the low population zone is a radius of 3 miles.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of ZIRLO® or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. For Unit 2, Watts Bar is authorized to place a maximum of 2496 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

4.2.2 Control Rod Assemblies

The reactor core shall contain 57 control rod assemblies. The control material shall be silver indium cadmium as approved by the NRC.

(continued)

5.9 Reporting Requirements (continued)

5.9.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation (power operated relief valve lift settings required to support the Cold Overpressure Mitigation System (COMS) and the COMS arming temperature), criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
   LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

b. The analytical methods used to determine the RCS pressure and temperature limits and COMS setpoints shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. WCAP-14040-A, Rev. 4 “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves.”

   2. WCAP-18124-NP-A, Rev. 0, “Fluence Determination with RAPTOR-M3G and FERRET,” and WCAP-18124-NP-A Rev. 0 Supplement 1-NP-A, Rev. 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials," may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.

   3. The PTLR will contain the complete identification for each of the TS reference Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements).

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 165 AND 72

TO FACILITY OPERATING LICENSE NOS. NPF-90 AND NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-390 AND 50-391

1.0 INTRODUCTION

By letter dated March 20, 2023 (Reference 1), the Tennessee Valley Authority (TVA or the licensee), submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC or Commission) for Watts Bar Nuclear Plant (Watts Bar), Units 1 and 2. The requested changes would revise Watts Bar Technical Specification (TS) 4.2.1, "Fuel Assemblies," to increase the maximum number of tritium producing burnable absorber rods (TPBARs) from 1,792 to 2,496. The requested changes also would revise Watts Bar, Unit 1, TS 5.9.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to be consistent with Watts Bar, Unit 2, TS 5.9.6. Additionally, the proposed changes would revise both units' TS 5.9.6.b to add WCAP-18124-NP-A, Revision 0 Supplement 1-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials," as an analytical method. Lastly, the requested changes would revise the Watts Bar Dual-Unit Updated Final Safety Analysis Report (UFSAR) to modify the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an updated version of the ORIGEN (Oak Ridge Isotope Generation) code.

A regulatory audit was conducted to examine non-docketed information generated by the licensee to support the LAR (Reference 2). The audit was conducted to gain an understanding, to verify certain information used to support the licensee's LAR, and to identify information that would require docketing to support the basis of a licensing or regulatory decision.

2.0 REGULATORY EVALUATION

2.1 Background

The U.S. Department of Energy (DOE) initially chose Watts Bar and Sequoyah Nuclear Plants to produce tritium for the replenishment of the National Security Stockpile by irradiating TPBARs installed in the core. TPBARs are similar to standard burnable poison rod assemblies (BPRAs) inserted into fuel assemblies. The BPRAs absorb excess neutrons and help control the power in the reactor to ensure an even power distribution and extend the time between refueling outages.

TPBARs function in a manner similar to a BPRA, but TPBARs absorb neutrons using lithium aluminate instead of boron. Tritium is produced when the neutrons strike the lithium material. A solid zirconium material in the TPBAR, called a getter, captures the tritium as it is produced. Most of the tritium is trapped in the getter material. However, a small fraction of the tritium will permeate through the TPBAR cladding into the RCS. After one cycle of exposure, the TPBARs are removed from the fuel assemblies and shipped to a DOE extraction facility.

The NRC staff reviewed DOE's safety assessments submitted in its "Tritium Production Core (TPC) Topical Report," NPD-98-181 dated July 30, 1998. In response to requests for additional information, DOE subsequently submitted Revision 1 by letter dated February 10, 1999 (Reference 3). The topical report systematically evaluates the impact of irradiating up to approximately 3,300 TPBARs in a reactor core on all areas covered by the Standard Review Plan (NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition"). The NRC staff's review of the DOE topical report is documented in NUREG-1672, "Safety Evaluation Report related to the Department of Energy's topical report on the tritium production core," (Reference 4). NUREG-1672 represents the staff's generic acceptance of resolution of the technical issues with TPCs except 17 plant-specific issues and any emergent issues that may not have been considered. The staff identified 17 interface items that must be addressed by a licensee referencing the DOE topical report in its plant-specific LAR.

The topical report is expected to be referenced by licensees participating in DOE's commercial light-water reactor (CLWR) tritium program and to form the basis for a plant-specific application for an amendment to the facility operating license authorizing irradiation of TPBARs for the production of tritium.

The history of TPBAR loading in the Watts Bar, Unit 1, core is documented in the July 29, 2016, safety evaluation (SE) for Amendment No. 107 to the Watts Bar, Unit 1, Facility Operating License, which authorized the insertion of up to 1,792 TPBARs in the Unit 1 core (Reference 5). On May 22, 2019, the NRC issued Amendment No. 27 to the Watts Bar, Unit 2 Facility Operating License to also allow the insertion of up to 1,792 TPBARs in the Unit 2 reactor (Reference 6). The current LAR seeks to increase the maximum number of TPBARs allowed to be inserted into the Watts Bar, Units 1 and 2, cores from 1,792 to 2,496.

2.2 Requested Changes

The licensee requested the following changes for the respective unit's TSs. Markups of the proposed TS changes for Watts Bar, Units 1 and 2, are provided in Attachments 1 and 2, respectively, to Enclosure 1 of the LAR.

Unit 1:

- Revise TS 4.2.1, to change the maximum number of TPBARs from 1,792 to 2,496
- Revise 5.9.6.a as shown below (new text shown in **bold** and deleted text shown in ~~strikeout~~):

  RCS pressure and temperature limits for heatup, cooldown, low temperature operation **(power operated relief valve lift settings required to support the Cold Overpressure Mitigation System (COMS) and the COMS arming temperature),** criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

- Revise TS 5.9.6.b as shown below (new text shown in **bold** and deleted text shown in ~~strikeout~~):

The analytical methods used to determine the RCS pressure and temperature limits **and COMS setpoints** shall be those previously reviewed and approved by the NRC~~. The acceptability of the analytical methods is documented in NRC letter, "WATTS BAR UNIT 1 - ACCEPTANCE FOR REFERENCING OF PRESSURE TEMPERATURE LIMITS METHODOLOGY AND PRESSURE TEMPERATURE LIMITS REPORT (TAC M89048)", September 22, 1995 and "EXEMPTION FROM THE REQUIREMENTS OF 10 CFR Part 50.60, ACCEPTANCE CRITERIA FOR FRACTURE PREVENTION MEASURES FOR LIGHTWATER NUCLEAR POWER REACTORS FOR NORMAL OPERATION - WATTS BAR NUCLEAR PLANT (TAC NO. M99063)." September 29, 1997. S~~, specifically, the analytical methods are described in the following references:

1. **WCAP-14040-A, Rev. 4 "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."** ~~Letter, W. J. Museler to NRC, regarding request for exemption from 10 CFR 50.60, March 10, 1994.~~

2. **WCAP-18124-NP-A, Rev. 0, "Fluence Determination with RAPTOR-M3G and FERRET," and WCAP-18124-NP-A Rev. 0 Supplement 1-NP-A, Rev. 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials," may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.**~~Letter, D. E. Nunn to NRC, regarding heatup and cooldown curves for normal operation (submitting WCAP-14176 and WCAP-14040, Rev. 1), December 23, 1994.~~

3. **The PTLR will contain the complete identification for each of the TS reference Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements).**~~Letter, R. R. Baron to NRC, responding to NRC July 11, 1995, request for additional information, July 31, 1995.~~

4. ~~Letter, R. R. Baron to NRC providing more information regarding cold overpressure mitigating system setpoints, September 8, 1995.~~

5. ~~Letter, J. A. Scalice to NRC, regarding request for exemption from 10 CFR 50.60, concerning use of Code Case N-514 to determine LTOP setpoints, dated June 20, 1997~~

Unit 2:

- Revise TS 4.2.1 to change the maximum number of TPBARs from 1,792 to 2,496

- Revise the list of analytical methods used to determine the RCS pressure and temperature limits and COMS setpoints described in TS 5.9.6.b.2 as shown below (new text shown in **bold**):

  2. WCAP-18124-NP-A, Rev. 0, "Fluence Determination with RAPTOR- M3G and FERRET," **and WCAP-18124-NP-A Rev. 0 Supplement 1-NP-A, Rev. 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials,"** may be used as an alternative to Section 2.2 of WCAP-14040-A Rev. 4.

The licensee also requested to revise section 15.1.7 of Watts Bar Dual-Unit UFSAR and the corresponding references to modify the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an updated version of the ORIGEN code. A markup of the proposed changes is in Attachment 6 to Enclosure 1 of the LAR.

2.3 Regulations and Guidance

Appendix G, "Fracture Toughness Requirements," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," specifies pressure-temperature (P-T) limits for operation and requirements for the Charpy upper-shelf energy (USE) of the reactor vessel. Appendix G to 10 CFR Part 50 requires that the initial unirradiated USE at the start of the vessel life be no less than 102 joules (75 foot-pounds (ft-lb)), and that the vessel maintain a USE level no less than 68 joules (50 ft-lb) throughout the service life. If it is anticipated that a vessel might fall below 68 joules (50 ft-lb) before license expiration, an analysis must be submitted that demonstrates "margins of safety against fracture equivalent to those required by Appendix G of the [American Society of Mechanical Engineers] ASME Code." This analysis is subject to the approval of the director of the Office of Nuclear Reactor Regulation.

Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 specifies the requirements for the material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. The data are used as described in Section IV of Appendix G to 10 CFR Part 50.

The provisions of 10 CFR Section 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," establish screening criteria for the pressurized thermal shock event that define a level of embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature, RT_{PTS} (i.e., the reference temperature evaluated for the end of life fluence for each of the vessel beltline materials), and must be determined by using the fluence received by the reactor vessel shell material on the expiration date of the operating license. The regulation establishes a methodology for calculating the RT_{PTS} for reactor vessel beltline plates, forgings, and welds. This assessment must be updated whenever there is a significant change in projected values of RT_{PTS}, or upon request for a change in the expiration date for operation of the facility. The pressurized thermal shock screening criterion is 270 degrees Fahrenheit (°F) for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials.

Pursuant to 10 CFR 50.36, "Technical specifications," TSs for operating reactors are required, in part, to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

Section 50.36(a)(1) of 10 CFR requires each applicant for a license authorizing operation of a production or utilization facility to include a summary statement of the bases or reasons for proposed TSs, other than those covering administrative controls; however, the bases shall not become part of the TSs.

Section 50.36(b) of 10 CFR requires that each license authorizing reactor operation include TSs derived from the analyses and evaluation included in the safety analysis report and amendments thereto.

Section 50.36(c)(4) of 10 CFR requires that design features that are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety be included in the technical specifications.

Section 50.36(c)(5) of 10 CFR requires that TSs include administrative controls, which are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

Section 50.34(b)(3) of 10 CFR specifies "the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in [10 CFR] Part 20..."

Part 20, "Standards for protection against radiation," of 10 CFR contains regulations to ensure that radiation doses are within the dose limits for occupational workers and members of the public and are as low as is reasonably achievable (ALARA).

Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50, contains regulations to ensure that the routine radioactive effluent releases are within the design objectives to meet the ALARA criterion.

Section 100.11, "Determination of exclusion area, low population zone, and population center distance," of 10 CFR Part 100, "Reactor site criteria," requires, in part, that the licensee determine:

> (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem [roentgen equivalent man] or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

Section 50.67, “Accident source term,” of 10 CFR, requires, in part, that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

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

The NRC staff determined that the following GDC are relevant to the review:

GDC 10, “Reactor design,” states, in part, that the reactor core and associated reactor coolant, control, and protection systems are to be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

GDC 19, “Control room,” states, in part, that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, and that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem [0.05 Sv] whole body, or its equivalent to any part of the body, for the duration of the accident.

GDC 61, “Fuel storage and handling and radioactivity control,” states, in part, that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

The following regulatory guidance is relevant:

NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition,” provides guidance to NRC staff in performing safety reviews of construction permit or operating license applications, including requests for amendments under 10 CFR Part 50. The following sections of the SRP are relevant to the NRC staff’s review:

Chapter 4, “Reactor,” provides guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and the thermal and hydraulic design of the core.

Chapter 4, section 4.2, “Fuel System Design,” Revision 3 (Reference 7), provides guidance for the review to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained.

Chapter 11, “Radioactive Waste Management,” provides guidance and acceptance criteria for determining the impact of the proposed change on plant effluent treatment systems and whether the ALARA design criteria of 10 CFR Part 50, Appendix I are met.

Chapter 12, “Radiation Protection,” provides guidance and acceptance criteria for determining whether radiation protection design feature, and programs, are sufficient to ensure that requirements of 10 CFR Part 20 are met such that there is reasonable assurance that occupational doses, and doses to members of the public, will be maintained within the limits, and will be ALARA.

Chapter 13, “Conduct of Operations,” section 13.5.2.1, Revision 2, “Operating and Emergency Operating Procedures” (Reference 8), provides guidance for reviewing the impact of the proposed change on operating and emergency operating procedures.

Chapter 15, “Transient and Accident Analysis,” provides guidance for verifying that the proposed change is bounded by previous license amendments approved by the NRC with respect to the remaining analysis acceptance criteria for transient and accident analyses.

Chapter 16.0, “Technical Specifications,” Revision 3 (Reference 9). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard TSs for each of the LWR nuclear designs. Accordingly, for Westinghouse Electric Company’s (Westinghouse) plant designs, the NRC staff’s review includes consideration of whether the proposed changes are consistent with NUREG-1431, “Standard Technical Specifications, Westinghouse Plants” (Reference 10). NUREG-1431, section 5.0, “Administrative Controls,” specifies reporting requirements, including an RCS PTLR. It also specifies that the analytical methods used to determine the RCS P-T limits and other setpoints shall be those previously reviewed and approved by the NRC and described in the reports listed in the TSs.

NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report [(TR)] on the Tritium Production Core," documents the NRC staff's review and conclusions for DOE Topical Report NDP-98-153, "Tritium Production Core (TPC) Topical Report," and Revision 1 to the report regarding the acceptability of irradiating up to approximately 3,300 TPBARs in a core.

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes general procedures acceptable to the NRC for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Specifically, there are two measures of radiation embrittlement used in this guide that are obtained from the results of the Charpy V-notch impact test (i.e., a shift in the Charpy curve for the irradiated material relative to that for the unirradiated material measured at the 30-foot-pound energy level and the decrease in the Charpy USE level).

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Revision 0.

RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Revision 0.

RG 8.32, "Criteria for Establishing a Tritium Bioassay Program," Revision 0.

American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1-1984, "Radioactive Source Term for Normal Operation of Light Water Rectors," American Nuclear Society, La Grange Park, Illinois.

3.0 TECHNICAL EVALUATION

In its LAR, the licensee requested to increase the maximum number of TPBARs allowed to be irradiated in the core from 1,792 to 2,496. The licensee addressed the 17 plant-specific interface items from NUREG-1672 in section 4.0 of the LAR. The staff's review of the current LAR is focused on plant-specific changes or new TPBAR-related information. Any changes from previously approved designs or methodology or any new relevant information are discussed in section 3.1 of this SE. The remainder of the staff's previous SE on NUREG-1672 remains applicable.

In support of TPBAR Interface Issue 5, "Control Room Habitability Systems," the licensee requested a revision to the Watts Bar Dual-Unit UFSAR to modify the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an updated version of the ORIGEN code. The staff's evaluation of the proposed UFSAR revision is evaluated in SE section 3.1.5.

The licensee requested changes to Watts Bar, Units 1 and 2, TS 4.2.1, and supporting changes to TS 5.9.6. The staff's evaluation of the proposed TS changes is documented in SE section 3.2.

3.1 Evaluation of TPBAR Interface Issues

3.1.1 Evaluation of TPBAR Interface Issue 1 – Handling of TPBARs

In NUREG-1672, the NRC staff identified three main issues pertaining to TPBAR handling. These issues included:

- activities required to remove the TPBARs from the fuel assemblies,
- activities required to prepare the TPBARs for shipment, and
- post-irradiation movement of the TPBARs outside of the fuel assemblies.

The applicant indicated that the information related to the handling of TPBARs provided in Reference 11 is still applicable, with exceptions. Section 4.1 of the LAR described these exceptions as initiatives to streamline and reduce durations of work on the refuel floor, and an exception related to the increase in weight of the completed TPBAR consolidation fixture (TCF) from 9,000 pounds to 9,050 pounds.

The staff evaluated the exception related to the increase in weight of the completed TCF. The LAR indicates that the TCF weight has increased to 9,050 pounds, (i.e., 3,800 upper TCF plus 5,250 lower TCF). In Reference 11, the licensee described that the TCF would be handled in two parts, each of which was less than half the rated capacity of the 10-ton auxiliary hoist used to move them. The staff finds that the TCF weight is still bounded by the capacity of the auxiliary hoist and the staff's safety conclusion documented in Reference 6, section 3.2.19.1, "Handling of TPBARs," is still applicable. Therefore, the NRC staff finds the additional heavy load handling activities associated with the proposed increase in the maximum number of tritium producing burnable absorber rods from 1,792 to 2,496 for Watts Bar, Units 1 and 2, to be acceptable. The NRC staff finds that TPBAR Interface Issue 1 has been satisfactorily addressed by the licensee.

3.1.2 Evaluation of TPBAR Interface Issue 2 – Procurement and Fabrication Issues

In NUREG-1672, sections 1.3 and 2.17.1, the NRC staff identified vendor-related activities with respect to quality assurance (QA) plans and fabrication inspections in order to determine compliance with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 and with 10 CFR Part 21, "Reporting of Defects and Noncompliance," and procurement processes performed on behalf of DOE for production core TPBAR components by contractors other than the production core TPBAR fabricator as an interface issue.

In section 4.0, "Technical Evaluation," of the LAR, TVA stated that there are no substantive changes to TPBAR Interface Issue 2, as described in its 2017 application to revise the Watts Bar, Units 1 and 2, TS 4.2.1, "Fuel Assemblies" (Reference 11), and it remains applicable to this LAR. Therefore, TPBAR Interface Issue 2 is not discussed.

In section 3.2.16, "Quality Assurance," of the NRC staff's SE (Reference 6), the NRC staff found that the TPBARs suppliers' Quality Assurance programs meet the requirements of Appendix B to 10 CFR Part 50, and the reporting requirements of 10 CFR Part 21.

As documented in the NRC's audit report (Reference 12), the NRC staff verified that there have been no changes made in how TVA provides the required quality oversight of its two suppliers for the TPBARs. The only change the NRC staff identified was the change in the name of one of the suppliers from WesDyne International, LLC to WGS.

Because there have been no substantive changes since the NRC staff's approval of the previously issued amendments, the staff finds that TPBAR Interface Issue 2 continues to be satisfactorily addressed by the licensee.

3.1.3 Evaluation of TPBAR Interface Issue 3 – Compliance with DNB Criterion

NUREG-1672, section 2.4.4 describes the thermal-hydraulic (T-H) design evaluation of a generic core with TPBARs using the acceptance criteria outlined in section 4.4 of the SRP (Reference 13).

During the review of NUREG-1672, the NRC staff identified compliance with the departure from nucleate boiling (DNB) criterion as an interface issue for which plant-specific information would be required in the licensee's submittal to support an amendment to the facility operating license for authorization to operate a TPC. This criterion requires a demonstration that DNB will not occur on the most limiting fuel rod on at least a 95 percent probability at a 95 percent confidence level. For the Watts Bar, Units 1 and 2, 2,492 TPBAR tritium production equilibrium cycle, the normal T-H DNB-related reload analyses were performed using the VIPRE-01 code (Reference 14).

As noted in section 2.1.2 of the NRC staff's audit summary (Reference 12), the staff audited Calculation CN-WATTS-209, which documents the T-H evaluation of a 2,496 TPBAR core. The staff verified that the reload analyses in CN-WATTS-209 included the following T-H evaluations, which were previously included and summarized in the December 20, 2017, LAR for the 1,792 TPBAR core (Reference 11):

1. An axial power shape study was performed to assure that the power distributions used in design would still be valid in the presence of the TPBAR. This study compares power shapes resulting from depletion during operation of the cycles to reference shapes used as the basis for Thermal-hydraulic design analyses.

2. The steamline break with rod withdrawal at power transient was analyzed to demonstrate the continued acceptability of the DNB ratio (DNBR) design basis for this transient.

3. The zero power hypothetical steamline break was analyzed to demonstrate that the DNBR design basis was met.

The NRC staff noted that Calculation CN-WATTS-209 concluded the axial power shape comparisons in the presence of TPBARs during the T-H safety analyses for Watts Bar, Units 1 and 2, have shown that the TPBARs do not present any excessive power distribution changes beyond those which are already bounded within the thermal-hydraulic design bases.

The T-H analysis also showed that the DNB criterion will not be challenged. To support this, an explicit check of the DNB criterion is included in the cycle-specific reload safety evaluation performed for each Watts Bar, Units 1 and 2, reload core.

The conclusions reached in Calculation CN-WATTS-209 are the same as the ones reached for the evaluation of the 1,792 TBPAR core and were previously documented in the licensee's LAR (Reference 11). The NRC staff finds the licensee's approach to evaluate DNB for the proposed core operation to be acceptable based on use of an NRC-approved methodology which remains applicable for the increased number of TPBARs. Therefore, the staff finds that TPBAR Interface Issue 3 has been satisfactorily addressed.

3.1.4 Evaluation of TPBAR Interface Issue 4 – Reactor Vessel Integrity

In NUREG-1672, section 2.5.3, the NRC staff determined that a licensee participating in DOE's program for the CLWR production of tritium must present the material properties for its reactor vessel and perform analyses that demonstrate it will meet the requirements of Appendices G and H to 10 CFR Part 50 and of 10 CFR 50.61.

3.1.4.1 Evaluation of Neutron Fluence Determination

Neutron fluence determination is addressed as part of TPBAR Interface Issue 4 for Watts Bar, Units 1 and 2, with loaded TPBARs. The reactor pressure vessel (RPV) beltline and extended beltline neutron fluence through 40- and 60-year license periods are calculated for Watts Bar, Units 1 and 2. The methodology used to calculate the neutron fluence is a discrete ordinates transport analysis based on RG 1.190 and the NRC staff-approved TR WCAP-18124-NP-A, Revision 0 (Reference 15). The analyses consist of neutron exposure parameters in terms of fast neutron (E > 1.0 million electron volts (MeV)) fluence and iron atom displacements (dpa) on a plant-specific and fuel cycle-specific bases. The fast fluence results in an energy-dependent damage function. ASTM International Standard Practice E853-18, "Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Neutron Exposure Results," recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a database for the future. The calculations and dosimetry evaluations described in are based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF), ENDF/B-VI.

Highlights of Methodology for Neutron Fluence Determination

The methodology used to determine the pressure vessel fluence is consistent with NRC-approved topical reports, WCAP-18124-NP-A, Revision 0 and WCAP-18124-NP-A, Revision 0 Supplement 1-NP-A (Reference 16). This TR describes the application of RAPTOR-M3G code to perform neutron transport calculations of beltline neutron exposure for LWRs and subsequent application of the FERRET code to perform a least squares analysis of LWR dosimetry. The methodology follows the guidance prescribed by Reg. Guide 1.190.

The reactor vessel fluence was calculated using RAPTOR-M3G which is a three dimensional (3D) parallel processing discrete ordinates radiation transport code. The geometric mesh description of the reactor model is accomplished using 150 to 250 radial intervals, 80 to 150 azimuthal intervals, and 100 to 200 axial intervals. The dimension of the mesh depends on the overall size of the reactor and on the complexity required to model the core periphery, the in-vessel surveillance capsules, and the details of the reactor cavity. The mesh size is chosen to assure proper convergence of the inner iterations of the transport calculations at a value of 0.001. The transport calculations are performed using the BUGLE-96 cross-section library which consists of a 67-group coupled neutron-gamma ray cross-section data set developed at Oak Ridge National Laboratory (ORNL), produced specifically for LWR applications. BUGLE-96: Coupled Cross Section Library is derived from ENDF/B-IV for LWR shielding and pressure vessel dosimetry applications. The spatial variation of the neutron source is obtained from a burnup-weighted average of the respective power distributions from individual fuel cycles.

Discrete Ordinates Transport Calculations with RAPTOR-M3G

The RAPTOR-M3G code solves the time-independent Linear Boltzmann Equation in the absence of fission in three dimensions via the discrete ordinates approximation. RAPTOR-M3G calculations are performed with an S8 (or higher) level-symmetric angular quadrature set. Neutron fluence values are determined directly from the results of this radiation transport calculations. The complete theory that constitutes Boltzmann Transport equation is provided in WCAP-18124, sections 2.5.1 through 2.5.5.

The FERRET code is used to perform a least squares adjustment that provides the capability to combine measurement data with the results of neutron transport calculations to establish a best estimate neutron energy spectrum with associated uncertainties at the measurement locations. This adjusts the spectrum best estimates for key exposure parameters such as neutron fluence rate, (E > 1.0 MeV), or iron atom displacement rate, dpa/s, along with their uncertainties. The theory, uncertainty calculations, and results for least square adjustment are provided in WCAP-18124, sections 3.2 through 3.4.

Watts Bar, Units 1 and 2,Transport Analyses

<u>Watts Bar, Unit 1</u>

The results from the neutron transport analyses for Watts Bar, Unit 1, are provided in table 2-1 through table 2-13 of WCAP-18769, “Watts Bar Units 1 & 2 Reactor Vessel Integrity Evaluations for the 2,496 TPBAR Implementation Project” (Enclosure 2 to LAR). In table 2-1, the fast neutron fluence is listed at the radial and azimuthal center of the surveillance capsule at the core midplane. Table 2-2 lists the integrated fast neutron fluence at the center of the surveillance capsule. Tables 2-4 and 2-5 lists iron dpa rates at the center of the surveillance capsules and the integrated iron dpa, respectively. These results establish the calculated exposure of the surveillance capsules to-date and projected into the future based upon the equilibrium 2,496 TPBAR core. Iron dpa rates at the center of the surveillance capsules are shown in table 2-22, with the integrated iron dpa presented in table 2-23. These results establish the calculated exposure of the surveillance capsules to-date and projected into the future based on the equilibrium 2,496 TPBAR core.

<u>Watts Bar, Unit 2</u>

Tables 2-19 through 2-31 of WCAP-18769 present selected results from the neutron transport analyses for Watts Bar, Unit 2. Table 2-19 presents the calculated fast neutron (E > 1.0 MeV) fluence rates at the radial and azimuthal center of the surveillance capsule positions at the core midplane. The integrated fast neutron fluence at the center of the surveillance capsules is presented in table 2-20.

Proposed Revision to Technical Specification 5.9.6

The licensee proposed to revise the Watts Bar, Unit 1, TS 5.9.6, to expand applicability of the WCAP-18124-NP-A, Revision 0, methodology to Unit 1. The licensee stated in the LAR that the justification for use of WCAP-18124-NP-A, Revision 0, that was previously reviewed and approved for Unit 2 is also applicable to Unit 1. As documented in the sections below, the NRC staff reviewed relevant sections of WCAP-18769-NP, which was included as Enclosure 2 to the LAR, and confirmed that the fluence calculations and results performed for Units 1 and 2 are consistent, which is expected given the similar nature of the units. The disposition of the first limitation and condition on WCAP-18124-NP-A, Revision 0, for Unit 2 was based on verifying that materials far from the active fuel region did not exceed the threshold fluence defined in Regulatory Issue Summary 2014-11 (Reference 17). Based on WCAP-18769-NP, the proposed operation for Unit 1 with the increased number of TPBARs and extended period of operation does not increase the accumulated fluence on these materials enough to invalidate this justification for Unit 1. The second limitation and condition on WCAP-18124-NP-A, Revision 0, for Unit 2 was dispositioned by simply stating that the least squares analysis was not used to modify the calculated surveillance capsule or RPV neutron exposure, and as stated above, the licensee said that this was also true for Unit 1. Therefore, the NRC staff finds that the licensee has satisfied the limitations and conditions for use of WCAP-18124-NP-A, Revision 0, with Unit 1, and that the proposed use of the methodology to calculate fluence for Unit 1 is acceptable.

In addition, the licensee proposes adding WCAP-18124-NP-A, Supplement 1-NP-A, Revision 0, to both units' TS. This supplement was submitted by Westinghouse to provide justification for expanding the use of the WCAP-18124-NP-A methodology to the extended beltline region, thus addressing in part Limitation and Condition 1 in the original topical report and providing a generic quantification of the uncertainty that may be used. The NRC staff finds the proposed use of the supplement to justify the uncertainties used for the extended beltline region to be acceptable because the licensee utilized the modeling defined in the supplement and the validation in the supplement is applicable to the Watts Bar plant design.

Summary

The NRC staff reviewed the use of RAPTOR-M3G for neutron fluence determination and the FERRET code for the least squares evaluation of LWR surveillance dosimetry. The fluence determination methodology is used to derive calculated values of neutron exposure at reactor vessel material locations of interest. The staff reviewed the application of RAPTOR-M3G code, and its solution procedure as applied to Watts Bar, Units 1 and 2. The staff reviewed the least square procedure that combines neutron spectrum calculations with available measurements to determine reduced uncertainties at measurement locations.

The NRC staff determined that the fluence determination process included the uncertainty criterion as discussed in the RG 1.190 for the beltline region of the Watts Bar, Units 1 and 2, with the increased number of TPBARs. This uncertainty was also previously found by the staff to be acceptable for the extended beltline region for previous Watts Bar, Units 1 and 2, TPBAR cores, and this finding continues to be applicable for the 2,496 TPBAR cores. The staff determined that the input parameters, calculated neutron spectrum, measured reaction rates, and dosimetry reaction cross-sections and their associated uncertainties are appropriate for use in Watts Bar, Units 1 and 2, dosimetry evaluations.

3.1.4.2 Evaluation of Reactor Vessel

The NRC staff noted that the TS 5.9.6 for Watts Bar, Units 1 and 2, govern the PTLR, which is a unit-specific document that provides the RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates for the current reactor vessel fluence period. Specifically, the analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC that are specified in TS 5.9.6. The latest revision of the PTLR for Watts Bar, Units 1 and 2, were submitted by letters dated December 8, 2021 (Reference 18), and April 10, 2023 (Reference 19), respectively.

The staff notes that updates to the PTLR, as a result of the licensee's 2,496 TPBAR Implementation Project, are governed by TS 5.9.6 for Watts Bar, Units 1 and 2. The staff's evaluation of the licensee's compliance to Charpy USE requirements contained in Appendix G to 10 CFR Part 50 and the pressurized thermal shock requirements contained in 10 CFR 50.61 are documented below.

3.1.4.2.1 Charpy Upper-Shelf Energy

The licensee stated that TPBAR Interface Issue 4 is addressed in WCAP-18769-NP, Revision 1, "Watts Bar Units 1 & 2 Reactor Vessel Integrity Evaluations for the 2,496 TPBAR Implementation Project," and the report presents the evaluation of the Watts Bar, Units 1 and 2, RPVs with respect to reactor vessel integrity.

Section 7 of WCAP-18769-NP, Revision 1, provides the licensee's evaluation of Charpy USE of the RPV. The NRC staff notes that WCAP-18769-NP, Revision 1, contains information and assessments of reactor pressure materials that are considered "Reactor Vessel Non-Beltline Materials" (i.e., materials that are not exposed to a neutron fluence of 1×10^{17} n/cm^2 through 32 effective full-power years (EFPY)), and reactor vessel integrity evaluations through 48 EFPY for Watts Bar, Unit 1. The licensee did not request review of, nor did the staff review, these portions of WCAP-18769-NP, Revision 1; thus, the NRC staff does not make any determinations or conclusions regarding this information in any potential future licensing applications or license periods. Additionally, unless specifically identified in the staff's SE below related to Charpy USE, the staff does not make any determinations or conclusions regarding this information for any potential future licensing applications or license periods.

The NRC staff's assessment of Watts Bar, Units 1 and 2, USE is documented separately below.

Watts Bar, Unit 1

During its review, the NRC staff assessed the material property values (e.g., initial USE and weight percent (%) copper (Cu)) for the "reactor vessel beltline materials" contained in table 3-1 of WCAP-18769-NP to confirm (1) these values were consistent with the corresponding values in the current licensing basis (CLB), or (2) revisions to the CLB values are justified and appropriate. Based on its review, the staff confirmed that the material property values are consistent with the licensee's CLB (e.g., UFSAR, PTLR, and other relevant license amendments) and, therefore, appropriate for use in determining USE values for 32 EFPY as part of the licensee's 2,496 TPBAR Implementation Project.

During its review, the NRC staff assessed the material property values (e.g., initial USE and weight % Cu) for the "reactor vessel extended beltline materials" contained in table 3-1 of WCAP-18769-NP to (1) confirm these values were consistent with the CLB, (2) confirm revisions to the CLB values are justified and appropriate, or (3) determine if these values are justified and appropriate if the RPV materials were not previously addressed in the CLB. The licensee stated that the initial USE values for the upper to intermediate shell circumferential weld seam W06 (Heat #899680) and Lower Shell to Bottom Head Ring Circumferential Weld Seam W04 (Heat #899680) were determined based on weld Heat #895075, which does have USE data and is a Rotterdam weld of the same flux type (Grau L.O., LW 320). The licensee explained that the initial USE value for these two RPV materials were determined in this manner because the certified material test reports (CMTRs) for weld Heat #899680 only reported a limited set of data at a single test temperature that did not reach greater than 55-percent shear. Thus, the NRC staff noted that in absence of USE data for weld Heat #899680, the licensee used the test results from weld Heat #895075 from the first surveillance capsule for Watts Bar, Unit 2, as discussed in WCAP-15046, "Analysis of Capsule U from the Tennessee Valley Authority Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program" (Reference 20), to estimate the initial USE value for weld Heat #899680. Given the lack of information from the CMTR for Heat #899680, the NRC staff finds the licensee's approach to be reasonable because both welds utilized the same flux type (i.e., Grau L.O., LW 320) and use of test data from the first capsule to determine initial USE values is consistent with NRC Branch Technical Position (BTP) 5-3, "Fracture Toughness Requirements" (Reference 21). The NRC staff noted that the test data from Capsule U indicated an increase in average energy absorption at full shear for Heat #895075, which is not expected following irradiation of test specimens; thus, staff noted that the reduction of 25 percent was conservatively based on the unirradiated USE value for Heat #895075 (i.e., 131 ft-lb), which is less than the capsule test data (i.e., 143 ft-lb). Additionally, the staff finds the 25-percent reduction in unirradiated USE value provides a conservative estimate for initial USE values (i.e., 98 ft-lb) for Heat #899680 given the available information.

Based on its review, the NRC staff determined that the material property values were based on information from plant-specific CMTRs, fabrication records, or determined to be in accordance with or more conservative than BTP 5-3 for the specific RPV material. The staff noted that BTP 5-3 provides guidance and a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with Appendix G to 10 CFR Part 50 and 10 CFR 50.61. Based on its review, the NRC staff finds the material property values for the reactor vessel extended beltline materials are acceptable and appropriate for use in determining USE values for 32 EFPY as part of the licensee's 2,496 TPBAR Implementation Project.

The NRC staff noted that the licensee assessed relevant surveillance data to determine its credibility per the criteria in RG 1.99, Revision 2, and potential consideration as to whether it is appropriate to use the surveillance data when calculating USE values. Specifically, the licensee indicated that USE values for the following RPV materials in table 7-1 of WCAP18769-NP were determined based on surveillance data:

- Unit 1
  - Intermediate Shell Forging 05
  - Intermediate to Lower Shell Circumferential Weld Seam W05

During its review, the NRC staff noted that section 4, "Surveillance Data," section 5, "Chemistry Factor," and appendix B, "Credibility Evaluation of the Watts Bar Unit 1 Surveillance Program," of WCAP-18769-NP, provide the licensee's assessment of surveillance data. Based on its review, the NRC staff verified the licensee's use and assessment of its credible surveillance data for the evaluation of USE values for these two RPV materials are appropriate and consistent with RG 1.99, Revision 2.

Based on its review, as described above related to the material property information and surveillance data, the NRC staff also verified that the licensee calculated the projected USE values for the "reactor vessel beltline materials" and "reactor vessel extended beltline materials," contained in table 7-1 of WCAP-18769-NP, including those that took into consideration credible surveillance data, in accordance with RG 1.99, Revision 2. The NRC staff finds that the projected USE values for these RPV materials, except the intermediate shell forging 05, are greater than the screening criterion of 50 ft-lb per Appendix G of 10 CFR Part 50 through 32

EFPY as part of the licensee’s 2,496 TPBAR Implementation Project. The NRC staff’s review of the intermediate shell forging 05 is provided below.

The NRC staff noted that the projected USE values for 32 EFPY for Intermediate Shell Forging 05 for Unit 1 (i.e., 46 ft-lb) is less than the criteria of no less than 68 joules (50 ft-lb) throughout the service life specified in Appendix G to 10 CFR Part 50. However, an equivalent margins analysis was performed by the licensee that demonstrated margins of safety exist against fracture for intermediate shell forging 05 that are equivalent to those required by Appendix G of the ASME Code at the 43-ft-lb level (i.e., USE at 32 EFPY). The NRC staff’s evaluation of this equivalent margins analysis is documented in letter dated May 11, 1994 (Reference 22). Since the projected USE value at 32 EFPY, as part of the licensee’s 2,496 TPBAR Implementation Project, for intermediate shell forging 05 for Unit 1 (i.e., 46 ft-lb) is greater than the 43 ft-lb (i.e., level at which it was demonstrated that margins of safety against fracture equivalent to those required by Appendix G of the ASME Code exist), the NRC staff finds the equivalent margins analysis performed for the intermediate shell forging 05 remains valid and the licensee has demonstrated compliance with Appendix G to 10 CFR Part 50.

Watts Bar, Unit 2

During its review, the NRC staff assessed the material property values (e.g., initial USE and weight % Cu) for the “reactor vessel beltline materials” contained in table 3-2 of WCAP-18769-NP to confirm (1) these values were consistent with the corresponding values in the CLB, or (2) revisions to the CLB values are justified and appropriate. The licensee explained that the chemistry values (e.g., weight % Cu) for the intermediate to Lower Shell Circumferential Weld Seam W05 (Heat #895075) was updated to be based on the average of all available data for this weld wire heat. The NRC staff finds this revision to the chemistry values for intermediate to Lower Shell Circumferential Weld Seam W05 (Heat #895075) to be acceptable and reasonable because all available data for this weld wire heat number was used in determining the best-estimate values for the material, which is consistent with the guidance in RG 1.99, Revision 2. Based on its review, the NRC staff confirmed that the material property values, except for that noted above for the chemistry values for Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat #895075), are consistent with the licensee’s CLB and, therefore, are appropriate for use in determining USE values for 32 EFPY as part of the licensee’s 2,496 TPBAR Implementation Project.

During its review, the NRC staff assessed the material property values (e.g., initial USE and weight % Cu) for the “reactor vessel extended beltline materials” contained in table 3-2 of WCAP18769-NP to (1) confirm these values were consistent with the CLB, (2) confirm revisions to the CLB values are justified and appropriate, or (3) determine if these values are justified and appropriate if the RPV materials were not previously addressed in the CLB. The licensee stated that the initial USE values for the upper to intermediate shell circumferential weld seam W06 (Heat #899680) and Lower Shell to Bottom Head Ring Circumferential Weld Seam W04 (Heat #899680) were determined based on weld Heat #895075, which does have the USE data and is a Rotterdam weld of the same flux type (Grau L.O., LW 320). The licensee explained that this was necessary because the CMTRs for weld Heat #899680 only reported a limited set of data at a single test temperature that did not reach greater than 55-percent shear and there was no additional information available for this weld heat. The NRC staff noted that in absence of USE data for weld Heat #899680, the licensee used the rest results from weld Heat #895075 from the first surveillance capsule for Watts Bar, Unit 2 (Reference 23), to estimate the initial USE value for weld Heat # 899680 and to ensure conservatism, the licensee reduced the USE value from the first surveillance capsule by 25 percent and result in an initial USE of 101 ft-lb.

Given the lack of information from the CMTR for Heat #899680, the NRC staff finds this approach to be reasonable because both welds utilized the same flux type (i.e., LW 320) and use of test data from the first capsule to determine initial USE values in accordance with BTP 5-3. Additionally, the staff finds the 25-percent reduction in test data for weld Heat #895075 provides a conservative estimate for the initial USE value (i.e., 101 ft-lb) for Heat #899680 given the available information.

Based on its review, the NRC staff determined that the material property values were based on information from plant-specific CMTRs, fabrication records, or determined in accordance with or more conservative than BTP 5-3 for the specific RPV material. The staff noted that BTP 5-3 provides guidance and a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with Appendix G to 10 CFR Part 50 and 10 CFR 50.61. Based on its review, the NRC staff finds the material property values for the “reactor vessel extended beltline materials” are acceptable and appropriate for use in determining USE values for 32 EFPY as part of the licensee’s 2,496 TPBAR Implementation Project.

The NRC staff noted that at the time of its review only one surveillance capsule from Watts Bar, Unit 2, had been withdrawn and tested in accordance with its approved capsule withdrawal schedule. Therefore, the guidance in Position 2.2 of RG 1.99, Revision 2, regarding the assessment of surveillance data is not currently applicable since two or more credible surveillance data sets are not available at the time of its review. As such, the licensee used Position 1.2 of RG 1.99, Revision 2 for the determining the decrease in Charpy USE of the reactor vessel materials at Watts Bar, Unit 2.

Based on its review, as described above related to the material property information and surveillance data, the NRC staff also verified that the licensee appropriately calculated the projected USE values for the “reactor vessel beltline materials” and “reactor vessel extended beltline materials” contained in table 7-3 of WCAP-18769-NP in accordance with RG 1.99, Revision 2. Additionally, the staff finds that the projected USE values for these RPV materials are greater than the screening criterion of 50 ft-lb per Appendix G of 10 CFR Part 50 through 32 EFPY as part of the licensee’s 2,496 TPBAR Implementation Project.

3.1.4.2.2 Pressurized Thermal Shock

The licensee stated that the TPBAR Interface Issue 4 is addressed in WCAP-18769-NP, Revision 1, and this report presents the evaluation of the Watts Bar, Units 1 and 2, RPVs with respect to RV integrity.

Section 6 of WCAP-18769-NP, Revision 1, provides the licensee’s evaluation of pressurized thermal shock of the RPV. The NRC staff noted that WCAP-18769-NP, Revision 1, contains information and assessments of reactor pressure materials that are considered “Reactor Vessel Non-Beltline Materials” (i.e., materials that are not exposed to a neutron fluence of 1×10^{17} n/cm^2 through 32 EFPY), and reactor vessel integrity evaluations through 48 EFPY for Watts Bar, Unit 1. The licensee did not request review of, nor did the staff review, these portions of WCAP-18769-NP, Revision 1; thus, the NRC staff does not make any determinations or conclusions regarding this information for any potential future licensing applications or license periods. Additionally, unless specifically identified in the staff’s SE below related to pressurized thermal shock, the staff does not make any determinations or conclusions regarding this information in any potential future licensing applications or license periods.

The NRC staff's assessment of Watts Bar, Units 1 and 2, PTS is documented below.

Watts Bar, Units 1 and 2

During its review, the NRC staff assessed the material property values (e.g., initial RT_{NDT}, weight % Cu and weight % nickel (Ni)) for the "reactor vessel beltline materials" contained in tables 3-1 and 3-2 of WCAP-18769-NP for Watts Bar, Units 1 and 2, respectively, to confirm (1) these values were consistent with the corresponding values in the CLB or (2) revisions to the CLB values are justified and appropriate. The licensee explained that the chemistry values (e.g., weight % Cu and weight % Ni) for Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat #895075) at Watts Bar, Unit 2, was updated so that it is based on the average of all available data for this weld wire heat. The NRC staff finds this revision to the chemistry values for the Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat #895075) at Watts Bar, Unit 2 to be acceptable and reasonable because all available data for this weld wire heat number was used in determining the best-estimate values for the material, which is consistent with guidance in RG 1.99, Revision 2.

Based on its review, the NRC staff confirmed that the material property values, except for that noted above for the chemistry values for Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat #895075) at Watts Bar, Unit 2, are consistent with the licensee's CLB and therefore appropriate for use in determining RT_{PTS} values for 32 EFPY as part of the licensee's 2,496 TPBAR Implementation Project. Additionally, based on its review, the NRC staff finds that appropriate margin values, consistent with 10 CFR 50.61, were applied for each Watts Bar, Units 1 and 2, "reactor vessel beltline materials" for the purposes of addressing pressurized thermal shock.

During its review, the NRC staff assessed the material property values [e.g., initial RT_{NDT} (reference temperature of the material)], weight % Cu and weight % Ni) for the "reactor vessel extended beltline materials" contained in tables 3-1 and 3-2 of WCAP-18769-NP for Watts Bar, Units 1 and 2, respectively, to (1) confirm these values were consistent with the CLB, (2) confirm revisions to the CLB values are justified and appropriate, or (3) determine if these values are justified and appropriate if the RPV materials were not previously addressed in the CLB. Based on its review, the staff determined that the material property values were based on information from plant-specific CMTRs, fabrication records, or determined in accordance with or more conservative than BTP 5-3 for the specific RPV material. BTP 5-3 provides guidance and a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with Appendix G to 10 CFR Part 50 and 10 CFR 50.61.

Based on its review, the NRC staff finds the material property values for the "reactor vessel extended beltline materials" are acceptable and appropriate for use in determining RT_{PTS} values for 32 EFPY as part of the licensee's 2,496 TPBAR Implementation Project. Additionally, based on this verification, the staff finds that appropriate margin values, consistent with 10 CFR 50.61, were applied for each Watts Bar, Units 1 and 2, "reactor vessel extended beltline materials" for the purposes of addressing pressurized thermal shock.

In addition, the NRC staff noted that the licensee assessed relevant surveillance data to determine its credibility per the criteria in RG 1.99, Revision 2, and potential consideration as to whether it is appropriate to use the surveillance data when calculating RT_{PTS} values for Watts Bar, Units 1 and 2. Specifically, the licensee indicated that RT_{PTS} values for the following RPV materials in tables 6-1 and 6-3 of WCAP-18769-NP for Watts Bar, Units 1 and 2, respectively, were determined based on credible surveillance data:

- Unit 1 - Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)
- Unit 2 - Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)

During its review, the NRC staff noted that section 4, “Surveillance Data,” section 5, “Chemistry Factor,” appendix B, “Credibility Evaluation of the Watts Bar Unit 1 Surveillance Program,” and appendix C, “Credibility Evaluation of the Watts Bar Unit 2 Surveillance Program,” of WCAP-18769-NP provide the licensee’s assessment of surveillance data. Based on its review, the NRC staff verified that the licensee’s use and assessment of relevant credible surveillance data for the evaluation of pressurized thermal shock and RT_{PTS} values is appropriate and consistent with 10 CFR 50.61 and RG 1.99, Revision 2.

The NRC staff noted that table 7-1 of WCAP-18769-NP, identifies the consideration of non-credible surveillance data for the Intermediate Shell Forging 05 at Watts Bar, Unit 1, which is the limiting material with respect to the pressurized thermal shock for Unit 1. The staff noted that the licensee provided its assessment of the non-credible surveillance data for this RPV material for completeness and was not intended for demonstration that pressurized thermal shock is addressed in accordance with 10 CFR 50.61 because the surveillance data was deemed non-credible. Additionally, since the use of the non-credible surveillance data provided lower estimates of RT_{PTS}, the assessment of this surveillance this data was not considered relevant to the NRC staff’s evaluation of pressurized thermal shock and compliance with 10 CFR 50.61.

Based on its review, as described above related to material property information and credible surveillance data, the NRC staff verified that the projected RT_{PTS} values were calculated in accordance with 10 CFR 50.61. As such, the staff finds that the associated RT_{PTS} values for the “reactor vessel beltline materials” and the “reactor vessel extended beltline materials” contained in tables 6-1 and 6-3 of WCAP-18769-NP, respectively, are appropriate and are less than the applicable screening criteria specified in 10 CFR 50.61 through 32 EFPY as part of the licensee’s 2,496 TPBAR Implementation Project; therefore, they are acceptable.

3.1.4.2.3 Adjusted Reference Temperature and P-T Limits

The LAR proposed a change to Watts Bar, Units 1, TS 5.9.6, “Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR).”

Section 8 of WCAP-18769-NP, Revision 1, provides the licensee’s evaluation of adjusted reference temperature (ART) of the RPV. The NRC staff noted that WCAP-18769-NP, Revision 1, contains information and assessments of reactor pressure materials that are considered “Reactor Vessel Non-Beltline Materials” (i.e., materials that are not exposed to a neutron fluence of 1×10^{17} n/cm^2 through 32 EFPY), and reactor vessel integrity evaluations through 48 EFPY for Watts Bar, Unit 1. The licensee did not request review of, nor did the NRC staff review, these portions of WCAP-18769-NP, Revision 1; thus, the NRC staff does not make any determinations or conclusions regarding this information for any potential future licensing applications or license periods. Additionally, unless specifically identified in the NRC staff’s SE below related to ART, the staff does not make any determinations or conclusions regarding this information in any potential future licensing applications or license periods.

The staff's assessment of Watts Bar, Units 1 and 2, ART and use of WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Reference 24), as the PTLR methodology for Watts Bar, Unit 1 is documented below.

Watts Bar, Units 1 and 2

As described in SE section 3.1.4.2.2, the staff assessed the material property values (e.g., initial RT_{NDT}, weight % Cu and weight % Ni) for the "reactor vessel beltline materials" and "reactor vessel extended beltline materials" contained in tables 3-1 and 3-2 for Watts Bar, Units 1 and 2, respectively. Based on its review, the staff confirmed that the material property values for the "reactor vessel beltline materials" and "reactor vessel extended beltline materials" are consistent with the licensee's CLB or revisions to the CLB values are justified and appropriate; therefore, are appropriate for use in determining ART values for 32 EFPY P-T limits. Additionally, based on its review, the staff finds that appropriate margin values, consistent with RG 1.99, Revision 2, were applied for each Watts Bar, Units 1 and 2, "reactor vessel beltline materials" for the purposes of addressing ART.

As described in SE section 3.1.4.2.2, the NRC staff noted that the licensee assessed relevant surveillance data to determine its credibility per the criteria in RG 1.99, Revision 2, and potential consideration as to whether it is appropriate to use the surveillance data when calculating ART values for Watts Bar, Units 1 and 2. Based on its review, the NRC staff verified that the licensee's use and assessment of relevant credible surveillance data for the evaluation of ART values is appropriate and consistent with RG 1.99, Revision 2.

The NRC staff noted that tables 8-7 and 8-8 of WCAP-18769-NP, identify the consideration of non-credible surveillance data for the Intermediate Shell Forging 05 at Watts Bar, Unit 1, which is the limiting material for Unit 1. The NRC staff noted that the licensee provided its assessment of the non-credible surveillance data for this RPV material for completeness and was not intended for use in the P-T limits because the surveillance data was deemed non-credible.

Based on its review, as described above related to material property information and credible surveillance data, the NRC staff verified that the projected ART values were calculated in accordance with RG 1.99, Revision 2. As such, the staff finds that the associated ART values for the "reactor vessel beltline materials" and the "reactor vessel extended beltline materials" contained in tables 8-7, 8-8, 8-10, and 8-11 of WCAP-18769-NP, are appropriate through 32 EFPY and use in the P-T limits; therefore, they are acceptable.

Watts Bar, Unit 1

Additionally, the licensee proposed to revise Watts Bar, Unit 1, TS 5.9.6.b to add WCAP-14040-A, Revision 4. The NRC staff's safety evaluation for WCAP-14040-A, Revision 3 (later approved and reissued as Revision 4) contains the following conditions:

1. Licensees who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to [Generic Letter] GL 96-03 ["Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits"] related to the RPV material surveillance program.

2. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of TR WCAP-14040 should be modified to reflect this NRC staff conclusion.

3. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify/eliminate the existing RPV flange minimum temperature requirements or an exemption request to modify/eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's P-T limit curves.

Condition 1 requests that the licensee provide information consistent with GL 96-03 to demonstrate that the plant maintains a reactor vessel material surveillance program in accordance with Appendix H to 10 CFR Part 50. Watts Bar, Unit 1, exceeds this neutron fluence threshold and therefore is subject to these requirements and must maintain reactor vessel surveillance programs in accordance with 10 CFR 50, Appendix H. By letter dated January 18, 2024 (Reference 25), the NRC approved the current Reactor Vessel Surveillance Capsule Withdrawal Schedule for Watts Bar. The most recent summary technical reports documenting the post-irradiation mechanical testing of the Charpy V-notch impact specimens and tensile specimens from Capsule Z for Watts Bar, Unit 1 (Reference 26), was performed in accordance with 10 CFR Part 50, Appendix H, and ASTM International Specification E185-82. Additionally, the NRC staff's assessment of the licensee's application of credible surveillance data from the RPV material surveillance program is documented above, and was determined to be appropriate and in accordance with RG 1.99, Revision 2 for Watts Bar, Unit 1. Therefore, the NRC staff finds that the licensee has met the Condition 1 of the SE for WCAP-14040-A, Revision 4, and provided the necessary information about its reactor vessel material surveillance program and use of surveillance data.

Condition 2 requests that Revision 4 of WCAP-14040-A reflect the NRC staff conclusion that no exemption is required for licensee use of provisions in ASME Code Cases N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," or N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," in conjunction with the basic methodology contained in WCAP-14040, Revision 3, since these ASME Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. The NRC staff noted that table A-1, "Status of ASME Nuclear Code Cases Associated with the P-T Limit Curve/COMS Methodology," of Appendix A, "Relevant ASME Nuclear Code Cases" was updated in Revision 4 of WCAP-14040-A to indicate the date, edition, and addenda of ASME Code, Section XI, when the referenced code cases were approved by the ASME. As indicated in Condition 2, the edition and addenda of ASME Code, Section XI, that are listed in table A-1 have been incorporated by reference in 10 CFR 50.55a. The NRC staff notes that Condition 2 of WCAP-14040-A, Revision 4, is not unique to Watts Bar, Unit 1, and was addressed with the publication of WCAP-14040-A, Revision 4. The PTLR methodology used by Watts Bar, Unit 1, to generate P-T limits curves based on WCAP-14040-A, Revision 4, is consistent with the methodology in ASME Code Section XI, Appendix G. Therefore, the NRC staff finds that the updated information has adequately addressed the Condition 2 in the SE to WCAP-14040-A, Revision 4.

Condition 3 requests that the reactor vessel flange minimum temperature requirements be incorporated into a facility's P-T curves until Appendix G to 10 CFR 50 is revised to modify the existing reactor vessel flange minimum temperature requirement or an exemption request to modify these requirements is approved by the NRC for a specific facility. At the time of publication for WCAP-14040-A, Revision 4, there was an ongoing rulemaking activity associated with amending Appendix G to 10 CFR Part 50 to remove the requirements related to the metal temperature of the closure head flange and reactor vessel flange regions, which resulted in Condition 3 of WCAP-14040-A, Revision 4. However, since that time, as published in the *Federal Register* (85 FR 852), dated January 8, 2020, this rulemaking activity to amend Appendix G to 10 CFR Part 50 was discontinued and Petition for Rulemaking (PRM) 50-69 was denied. Therefore, the NRC staff finds that Condition 3 in the SE to WCAP-14040-A, Revision 4, has been adequately addressed because there have been no changes to the requirements of Appendix G to 10 CFR Part 50 regarding the reactor vessel flange minimum temperature requirements, and the licensee is not requesting a plant-specific exemption to the requirements.

Based on its review, the NRC staff finds the analytical methods in WCAP-14040-A, Revision 4 to determine the RCS pressure and temperature limits and COMS setpoints to be acceptable for use for Watts Bar, Unit 1, because the staff previously approved its in use in safety evaluation dated February 27, 2004 (Reference 27), and the licensee as has adequately addressed the conditions in the SE for the use of WCAP-14040-A, Revision 4, as described above. Therefore, the NRC staff concludes that by meeting all three conditions, amending the PTLR methodology to reference WCAP-14040-A, Revision 4, for Watts Bar, Unit 1, is acceptable.

3.1.4.3 Evaluation of Capsule Withdrawal Schedule Change

As a result of this LAR, the licensee stated that a revision to the capsule withdrawal schedule at Watts Bar, Units 1 and 2, is required due to the updated plant-specific neutron fluence that was addressed as part of the LAR. Thus, the licensee requested revision of the reactor vessel material surveillance capsule withdrawal schedule for Watts Bar, Units 1 and 2, in accordance with 10 CFR Part 50, Appendix H, Section III.B.3.

The NRC staff's evaluation of TVA's proposed revision to the capsule withdrawal schedule is documented in a letter dated January 18, 2024 (Reference 25).

3.1.4.4 Finding Regarding TPBAR Interface Issue 4

Based on the above evaluation in SE section 3.1.4, the NRC staff finds that the licensee has satisfactorily addressed TPBAR Interface Issue 4.

3.1.5 Evaluation of TPBAR Interface Issue 5 – Control Room Habitability Systems

In section 2.6.1 of NUREG-1672, the NRC staff concluded that a plant-specific assessment for the dose criteria must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

The proposed change for increasing the number of TPBARs that can be irradiated in the TPCs of Watts Bar, Units 1 and 2, from 1,792 TPBARs to 2,496 TPBARs is similar to changes already approved by the NRC through Watts Bar, Unit 1, Amendment No. 107 (Reference 5) and Watts Bar, Unit 2, Amendment No. 27 (Reference 6). Consistent with these previously approved amendments, this LAR is justified by the licensee, in part, based on analysis, testing, and evaluation of the TPBARs as reported previously by the DOE. The DOE has previously submitted a classified/proprietary version of the TPC Topical report NDP-98-153 and an unclassified/non-proprietary version, NDP-98-181, Revision 1, for NRC review.

Design-basis accident analyses that assume fuel damage in the accident progression, the large break loss-of-coolant accident (LBLOCA) and fuel handling accident (FHA) for Watts Bar, Units 1 and 2, require the determination of the core fission product inventory in order to determine the radiological consequences to potentially affected workers and members of the public.

The CLB fission product inventory was determined using ORIGEN 2.1 software. For this LAR, the licensee calculated the fission product inventory for TPCs with 2,496 TPBARs using ORIGEN-ARP/ORIGEN-S in SCALE 6.0 software, hereafter referred to as ORIGEN/ARP. Use of this update is acceptable because ORIGEN-ARP has been approved by NRC guidance. Specifically, RG 1.183, section 3.1 states, in part, that, “the core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP.” Therefore, the NRC staff finds the licensee’s proposed revision to the UFSAR to specify the use of the ORIGEN-ARP/ORIGEN-S in SCALE 6.0 software acceptable.

The source terms associated with the LBLOCA and the FHA are based on the core inventory because these accidents assume fuel damage.

The core inventory for Watts Bar, Units 1 and 2, has been revised as reflected in table 4.4-2 of the LAR. As stated in section 4.4 of Enclosure 1 to the LAR, the tritium inventory is determined using a design inventory of 1.2 grams of tritium per TPBAR, resulting in a total of 2.90×10^7 curies (Ci) of tritium in the core. The activities of the remaining radionuclides in the core inventory were determined based on a 96-feed equilibrium cycle, which consists of 108 once-burned assemblies and 85 twice-burned assemblies. Thus, the average assembly inventories, determined using ORIGEN-ARP, were multiplied by the number of fuel assemblies for that set (i.e., 108 once-burned assemblies and 85 twice-burned assemblies) and the results were summed to determine the total activity for each radionuclide.

The source terms associated with the main steam line break (MSLB), steam generator tube rupture (SGTR), loss of offsite power (LOOP), and waste gas decay tank (WGDT) rupture are based on the primary and secondary coolant radionuclide concentrations because these accidents are not assumed to result in fuel damage.

The radionuclide concentrations in the reactor coolant, the secondary steam and the secondary water were calculated using the same methodology as in the CLB, except with 2,496 TPBARs and an increase in integral fuel burnable absorber (IFBA) releases to 80 Ci/year. The CLB primary and secondary coolant concentrations are based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1-1984, “Radioactive Source Term for Normal Operation of Light Water Reactors.” Regarding the determination of the concentration of tritium in the primary, the licensee provided the following in section 4.4 of Enclosure 1 to the LAR:

> The total annual tritium expected from the TPC is 13,430 Ci/year and is based on 2,496 TPBARs with a permeation rate of 5 Ci/TPBAR/year (i.e., 12,480 Ci/year), an IFBA release rate of 80 Ci/year, and a non-TPC source of 870 Ci/year. This results in an average tritium concentration of 15.5 µCi/[gram] gm [(13,430 Ci/year*1.0 µCi/gm)/870 Ci/year] rounded up for conservatism. The concentration with two TPBAR failures was determined by adding the tritium inventory of two TPBARs to the average amount of tritium in the RCS and

dividing by the RCS mass. The average amount of tritium was determined by multiplying the average tritium concentration determined above by the RCS mass. Each TPBAR is assumed to have a maximum tritium inventory of 11,600 Ci at the end of a cycle. This resulted in an expected tritium concentration in the primary coolant of approximately 124 μCi/gm for two TPBAR failures.

Based on the NRC's review of the LAR and documents verified during the audit, the staff finds that the source terms used for the accident analyses are acceptable because the licensee used a code that is endorsed by RG 1.183, and it used methods previously approved by the NRC.

For Watts Bar, Units 1 and 2, the acceptance criteria for radiological accident consequence analyses are divided into doses impacting the public at the exclusion area boundary and low population zone, as defined in 10 CFR 100.11 and 10 CFR 50.67 (for the FHA), and those impacting the operators in the control room, as defined in GDC 19 and 10 CFR 50.67 (for the FHA). These acceptance criteria are in terms of whole-body dose, beta dose, and thyroid dose for all accidents except the FHA. The FHA is in terms of TEDE, per 10 CFR 50.67. The tritium dose does not contribute significantly to whole body or thyroid doses because the decay emission energy of radiation from tritium is insufficient to penetrate the skin and contribute to whole-body dose and the thyroid dose is calculated solely from the amount of radioiodine that can be taken up by a person who is exposed to radioactive material released during an accident.

The licensee determined the radiological doses for design basis accidents and provided the results in a series of tables in Enclosure 1 to the LAR: table 4.4-5 (LBLOCA), table 4.4-6 (FHA), table 4.4-7 (MSLB), table 4.4-9 (SGTR for Unit 1), table 4.4-10 (SGTR for Unit 2), table 4.4-11 (LOOP), and table 4.4-12 (WGDT rupture). As described in the NRC's audit plan (Reference 2), the staff requested the licensee's documentation that described the methodology, including parameters, calculations, and assumptions that demonstrate how the radiological doses were determined. The audit report (Reference 12) summarized the staff's observations regarding accidents that affect control room operator dose. The NRC staff verified that the licensee's analyses for the accidents evaluated are consistent with approaches previously approved by the NRC, as described in the licensee's UFSAR.

Based on the NRC staff's review of the information in the LAR and verification of the analyses during the audit, the staff finds that the licensee's estimates of the dose consequences of the design basis accidents are acceptable because they will continue to comply with the requirements of 10 CFR 100.11, 10 CFR Part 50 Appendix A, GDC 19, 10 CFR 50.67, and the accident-specific dose guidelines in RG 1.195 and RG 1.183, Revision 0.

In the LAR, the licensee stated that the Watts Bar licensing basis does not include an analysis for the radiological consequences of the failure of a small line carrying primary coolant outside containment.

The NRC staff considered the impact of the increase in TPBAR loading for the radiological consequences of the failure of a small line carrying primary coolant outside containment. The use of larger amounts of TPBARs could potentially increase the amount of fission products that could result in the reactor coolant. However, the limit on the RCS specific activity, TS 3.4.16, is not changing for either unit at Watts Bar. Therefore, the NRC staff finds that the current licensing basis for Watts Bar, Units 1 and 2, continues to be applicable to the larger number of TPBARs.

The NRC staff considered the impact of the increase in number of TPBARs to the source term for a rod ejection accident. The licensee stated that no specific analysis of a rod ejection accident was performed because it is bounded by the LBLOCA dose consequences, and the LBLOCA dose consequences are low enough to satisfy the acceptance criterion for a rod ejection accident. The NRC staff finds that since a rod ejection accident is a highly localized phenomena and the maximum TPBAR loading in any single fuel assembly is not changing, the overall number of TPBARs in the core is not expected to change the dose consequences of a rod ejection accident significantly, and not enough to cause them to be more severe than the dose consequences from an LBLOCA.

Based on the review described above, the NRC staff finds that the licensee has satisfactorily addressed TPBAR Interface Issue 5.

3.1.6 Evaluation of TPBAR Interface Issue 6 – Specific Assessment of Hydrogen Source and Timing or Recombiner Operation

In NUREG-1672, section 2.6.2, the NRC staff concluded that a plant-specific assessment is required to quantify the sources and to determine the time at which initiation of recombiner operation should commence to limit the hydrogen concentration to acceptable levels by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

In section 4.0, "Technical Evaluation," of the LAR, the licensee stated that there are no substantive changes to Interface Issue 6, as described in its 2017 application to revise the Watts Bar, Units 1 and 2, TS 4.2.1, "Fuel Assemblies" (Reference 11), and it remains applicable to this LAR. Therefore, Interface Issue 6 is not discussed.

The NRC staff's evaluation of TPBAR Interface Issue 6 is documented in section 3.2.1, "Specific Assessment of Hydrogen Source and Timing of Recombiner Operation," and section 3.2.19.2, "Assessment of Combustible Gas Control with a Tritium Production Core," of Reference 6. Because there have been no substantive changes to this interface issue, the NRC staff finds that TPBAR Interface Issue 6 continues to be satisfactorily addressed by the licensee.

3.1.7 Evaluation of TPBAR Interface Issue 7 – Light Load Handling System

In NUREG-1672, section 2.9, the NRC staff determined that a plant-specific analysis of light load handling system must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

The licensee indicated that the information related to TPBAR Interface Issue 7 provided in Reference 11 is still applicable, with an exception. In section 4.5 of the LAR, the licensee states that the spent fuel pit crane has been replaced and has a reduced capacity of 2,500 pounds (lbs) as opposed to the original crane's capacity of 4,000 lbs.

As previously reported in its December 2017 LAR (Reference 11), the licensee stated that the spent fuel pit crane will be used to handle TPBAR assemblies and consolidation canisters within the pool. The weight of a fuel assembly containing 24 TPBARs (including the hold-down assembly) is less than a fuel assembly with a rod cluster control assembly and the weight of a loaded canister submerged in water is less than 700 lbs, which is less than the 2,500 lb capacity of the replacement crane.

The NRC staff finds that the light load handling system information provided in Reference 11 is still applicable, and the staff's safety conclusion documented in Reference 6, section 3.2.19.3, "Light-Load Handling System," is still applicable. Therefore, the NRC staff finds the additional light load handling activities associated with the proposed increase in the maximum number of TPBARs from 1,792 to 2,496 for Watts Bar, Units 1 and 2, to be acceptable. The staff finds that TPBAR Interface Issue 7 has been satisfactorily addressed by the licensee.

3.1.8 Evaluation of TPBAR Interface Issue 8 – Station Service Water System

In NUREG-1672, section 2.9.1, the NRC staff determined that a quantitative analysis for the station service water system needs to be addressed by licensees participating in DOE's program for the CLWR production of tritium.

The licensee indicated that the information related to TPBAR Interface Issue 8, "Station Service Water System," provided in Reference 11 is still applicable, with exceptions. In section 4.6 of the LAR, the licensee indicated that it has updated the quantitative analysis of decay heat resulting from the proposed increase in TPBARs from 1,792 to 2,496. The licensee's updated analysis demonstrates that the heat loads that discharge into the component cooling system are bounded by what was previously analyzed for the previous increase to 1,792 TPBARs.

The NRC staff evaluated the results of the decay heat quantitative analysis and confirmed that the heat loads remain bounded by the analysis of record; therefore, the staff finds that the information provided in Reference 11 related to TPBAR Interface Issue 8 is still applicable, and the staff's safety conclusion documented in Reference 6, section 3.2.19.4, "Station Service Water System," is still applicable. Therefore, the NRC staff finds the TPBAR Interface Issue 8 has been satisfactorily addressed by the licensee.

3.1.9 Evaluation of TPBAR Interface Issue 9 – Ultimate Heat Sink

In NUREG-1672, section 2.9.1, the NRC staff determined that the effect on the ultimate heat sink (UHS) must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

The licensee indicated that the information related to TPBAR Interface Issue 9, "Ultimate Heat Sink," provided in Reference 11 is still applicable, with exceptions. In section 4.7 of the LAR, the licensee states that it has updated the quantitative analysis of decay heat resulted from the increase in TPBARs from 1,792 to 2,496. The licensee stated that its updated analysis has shown that the heat loads that discharge into the UHS are bounded by what was analyzed for the previous increase to 1,792 TPBARs.

The NRC staff evaluated the results of the decay heat quantitative analysis and confirmed that the heat loads remain bounded by the analysis of record; therefore, the staff finds that the information provided in Reference 11 related to TPBAR Interface Issue 9 is still applicable, and the staff's safety conclusion documented in Reference 6, section 3.2.19.5, "Ultimate Heat Sink," is still applicable. Therefore, the NRC staff finds the TPBAR Interface Issue 9 has been satisfactorily addressed by the licensee.

3.1.10 Evaluation of TPBAR Interface Issue 10 – New and Spent Fuel Storage

In NUREG-1672, section 2.9.2, the NRC staff concluded the activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

In section 4.0, “Technical Evaluation,” of the LAR, the licensee stated that there are no substantive changes to Interface Issue 10, as described in its 2017 application (Reference 11), and it remains applicable to this LAR. Therefore, TPBAR Interface Issue 10 is not discussed.

The NRC staff’s evaluation of TPBAR Interface Issue 10 is documented in section 3.2.1, “Specific Assessment of Hydrogen Source and Timing of Recombiner Operation,” and section 3.2.19.1, “Handling of TPBARs,” of Reference 6. Because there have been no substantive changes to this interface issue, the NRC staff finds that TPBAR Interface Issue 10 continues to be satisfactorily addressed by the licensee.

3.1.11 Evaluation of TPBAR Interface Issue 11 – Spent Fuel Pool Cooling and Cleanup System

In NUREG-1672, section 2.9.3, the NRC staff determined that quantitative analysis to determine the absolute spent fuel pool temperatures must be performed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

The licensee indicated that the information related to TPBAR Interface Issue 11, “Spent Fuel Pool Cooling and Cleanup System,” provided in Reference 11 is still applicable, with exceptions. In section 4.8 of the LAR, the licensee indicated that the previously established site-specific methodology for analysis is maintained but is proposed to be updated to reflect the TPBAR increase.

The NRC staff reviewed the results of the updated thermal analysis and noted that the SFP CCS is limited to 28.1 million British thermal units per hour (MBTU/hr) at design basis maximum UHS temperature and design basis heat exchanger fouling conditions. The NRC staff noted that the current Watts Bar licensing basis for the SFP CCS provides for cycle-specific analysis to establish the minimum decay time prior to discharge of fuel assemblies to the SFP based on plant and environmental conditions at the time of the outage. The licensee’s updated analysis establishes 47.4 MBTU/hr as the bounding heat load and considers the alternating discharges from the two Watts Bar units and the combined effects of anticipated UHS temperatures and recent heat exchanger performance testing. This method provides reasonable assurance that the maximum SFP water temperature would not exceed established limits.

The NRC staff finds that the effect of the Watts Bar, Units 1 and 2, TPC operation on SFP heat load is acceptable because administrative controls are in place to ensure that SFP total heat load and temperature remain within the current licensing basis values. In combination with the existing provision of an SFP cooling system having reliability consistent with the importance of decay heat removal at high heat loads, and the provision of makeup system capable of maintaining SFP water level under accident conditions, fuel storage design capabilities would remain consistent with the requirements of GDC 61, of Appendix A to 10 CFR Part 50. Therefore, the NRC staff finds the TPBAR Interface Issue 11 has been satisfactorily addressed by the licensee.

3.1.12 Evaluation of TPBAR Interface Issue 12 – Component Cooling Water System

In NUREG-1672, section 2.9.4, the NRC staff determined that the effect on the component cooling water system’s heat transfer and flow requirements from the increase in spent fuel pool heat load during cooldown operations due to the additional heat load generated by the TPC must be analyzed on a plant-specific basis by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

The licensee indicated that the information related to TPBAR Interface Issue 12, “Component Cooling Water System,” provided in Reference 11 is still applicable, with exceptions. In section 4.9 of the LAR, the licensee indicates that it has updated the quantitative analysis of decay heat resulting from the proposed increase in TPBARs from 1,792 to 2,496. The licensee’s updated analysis demonstrates that the heat loads that discharge into the CCW are bounded by what was analyzed for the previous increase to 1,792 TPBARs.

The NRC staff evaluated the results of the updated analysis and confirmed that the heat loads remain bounded by the previous analysis; therefore, the NRC staff finds that the information provided in Reference 11 related to TPBAR Interface Issue 12 is still applicable, and the staff’s safety conclusion documented in Reference 6, section 3.2.19.8, “Component Cooling Water System,” is still applicable. Therefore, the NRC staff finds that TPBAR Interface Issue 12 has been satisfactorily addressed by the licensee.

3.1.13 Evaluation of TPBAR Interface Issue 13 – Demineralized Water Makeup System

In NUREG-1672, section 2.9.5, the NRC staff determined that a detailed analysis for the effect on the demineralized water makeup system due the presence of TPBARs must be analyzed on a plant-specific basis by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

In section 4.10 of the LAR, the licensee indicates that the impact of increasing TPBARs in the core increases tritium levels in the RCS due to additional tritium permeation through the TPBARs. The licensee calculated that a tritium producing core with 2,496 TPBARs is expected to increase the average calculated RCS tritium concentration from 11.4 μCi/gm to 15.5 μCi/gm, assuming no extra dilution of the RCS. LAR section 4.11, “TPBAR Interface Issue 14: Liquid Waste Management Systems,” addresses how the waste management system is capable of handling the increase in tritium concentration without crediting additional dilutions of the RCS by the demineralized water makeup system (DWMS). The applicant also indicated that if additional dilution activities are necessary, the makeup water treatment plant has sufficient capacity to supply the filtered and demineralized water required for both operating units (including a TPC with up to 2,496 TPBARs) to the demineralized water, storage, and distribution system.

The NRC staff evaluated the information provided in the LAR and observed that operation with a TPC may increase tritium levels in the RCS due to normal reactor tritium production plus tritium permeation from the TPBARs. The LAR indicates that the liquid waste management system is capable of handling the increase in Tritium level in the RCS water. The staff’s evaluation of the liquid waste management system is addressed in section 3.1.14 of this SE.

The licensee also indicated that additional dilution operations may be performed in order to maintain the RCS tritium levels. Any increase in dilution activities would place increased demands on the DWMS for the required makeup. The licensee performed an analysis to determine the adequacy of the existing DWMS to meet increased demands which may result from TPC operation. The licensee determined that the plant’s current storage and purified water production capabilities are adequate to handle the potential increase in RCS tritium levels. The licensee’s evaluation indicates that the demineralized water makeup system capacity is adequate for plant operation with the TPC.

Since the necessary dilution for potential TPC operational impacts is within the normal capacity of the system and the purpose of the dilution is to maintain monitored releases within applicable limits and ALARA per 10 CFR Part 20, the NRC staff finds that sufficient storage and water makeup capacity is available to adequately meet any additional feed and bleed demands from Watts Bar, Units 1 and 2, TPC operation. Therefore, the NRC staff finds the TPBAR Interface Issue 13 has been satisfactorily addressed by the licensee.

3.1.14 Evaluation of TPBAR Interface Issue 14 – Liquid Waste Management Systems

In NUREG-1672, section 2.11.2, the NRC staff determined that enhanced plant-specific tritium monitoring, and surveillance programs and procedures for operator actions on abnormal tritium release events, and a plant-specific analysis to demonstrate that the plant continuously meets release concentration and dose limits need to be evaluated by licensees participating in DOE's program for the CLWR production of tritium.

To address TPBAR Interface Issue 14, the licensee supplemented its previous analyses for the Watts Bar, Units 1 and 2, LARs to include a TPC for both Unit 1 and Unit 2 with a maximum of 2,496 TPBARs. The same permeation rate of 5 Ci/TPBAR/year was considered. The licensee addressed the two plant-specific analyses related to TPBAR Interface Issue 14 of NUREG-1672: (1) plant-specific tritium monitoring and surveillance programs and procedures for operator actions during an abnormal tritium release event, and (2) plant-specific analyses to demonstrate that the plant continuously meets release concentration limits.

The licensee stated that the description of the plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event in a prior LAR (Reference 11) and evaluated by NRC staff in a prior SE (Reference 6) remain applicable to the LAR, with the exception of updated tritium permeation values provided in table 4.11-1. Table 4.11-1 shows that the values are still bounded by the 5 Ci/TPBAR/year permeation rate.

3.1.14.1 Evaluation of Source Terms

In the LAR, the licensee describes two radioactive source terms used in evaluating the radiological impact on normal operations and AOOs of TPC operations with 2,496 TPBARs.

The first of these is the conservative or design-basis source term. The design-basis source term assumes the maximum allowable release of radioactive material from the core to the reactor coolant due to reactor operations with fuel defects. This source term is used to evaluate the adequacy of plant design features such as shielding, ventilation and radwaste system processing capacities. The licensee assumed that the contribution to the design-basis source term of tritium permeating from the TPBARs will be 5 Ci/year/TPBAR. Figures 4.11-3A and 4.11-3B in the LAR show the yearly estimated tritium permeation rates to be less than 3.5 Ci/year/TPBAR in both Watts Bar, Units 1 and 2, TPCs, making the 5 Ci/year/TPBAR design-basis source term bounding.

The second source term is used to evaluate the plant effluents against the design criteria in 10 CFR 50, Appendix I, and to demonstrate compliance with the liquid and gaseous effluent concentration limits and dose limits to members of the public in 10 CFR Part 20. The realistic source term was based on guidance in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors," Revision 1, with an additional allowance for the tritium produced during TPC operation (Reference 28). The realistic source term does not assume maximum TS fuel leakage. However, the licensee elected to have the realistic source term conservatively assume a tritium contribution from TPBAR operations of 5 Ci/year/TPBAR and 2,496 TPBARs loaded in the core. The NRC staff notes that the assumption of 5 Ci/TPBAR/year is a conservative assumption for tritium permeation because historical TPC operation at Watts Bar, Units 1 and 2, has shown that the annualized per TPBAR permeation have consistently remained less than 3.5 Ci/year. The realistic annual tritium value from NUREG-0017 was determined to be 1,392 curies with an additional 12,480 curies (i.e. 2,496 TPBARs at 5 curies/TPBAR/year) from TPC operation; this analysis yields a total average annual value of 13,872 curies of tritium.

Consistent with previous Watts Bar, Units 1 and 2, licensing actions, this realistic source term differs from that which was assumed in the DOE TR because the new Watts Bar, Units 1 and 2, realistic source term does not consider the failure of two TPBARs to be a creditable event during the plant's lifetime. Therefore, the failure of two TPBARs event does not fall within the scope of the definition of an AOO.

3.1.14.2 Evaluation of Impacts on Public Dose and Effluents

To ensure public doses and radioactive effluent discharges will be within the limits of 10 CFR Part 20, the licensee applied the approach described in 10 CFR 20.1302(b)(2). The licensee calculated the sum-of-ratios of each isotope concentration (C) to its corresponding Effluent Concentration Limit (ECL)—as listed in 10 CFR 20, Appendix B, Table 2, Columns 1 and 2 for gaseous and liquid effluents, respectively—(i.e. C/ECL). A C/ECL sum of less than 1.0 indicates that the annual average effluent release is less than 50 mrem (or 50 percent of the public dose limit of 10 CFR 20.1301). As described in 10 CFR 20.1302(b)(2), the additional 50 percent of the dose limit is reserved for the external exposure that is limited to 50 mrem per year by 10 CFR 20.1302(b)(2)(ii).

The licensee controls effluents in accordance with Watts Bar TS 5.7.2.7, "Radioactive Effluent Controls Program," which imposes the requirement that the licensee maintain an Offsite Dose Calculation Manual. Under this program, the release of radioactive liquids from the liquid waste system is made after laboratory analysis of the material to be discharged. The licensee has procedural and engineering controls in place to ensure that inadvertent discharges do not occur and that discharges are performed when sufficient dilution flow is available. Watts Bar, Units 1 and 2, have sufficient storage to enable holdup, dilution, and adequate timing of radioactive releases. Tables 4.11-4, 4.11-5, 4.11-6, and 4-11-7 in the LAR provide the results of licensee's analyses and confirm that liquid and gaseous effluents will be within applicable limits.

The licensee evaluated the total site dose from operation of TPCs at Watts Bar, Units 1 and 2, to verify compliance with 10 CFR 20.1301(e), which enforces the U.S. Environmental Protection Agency's 40 CFR Part 190 limits. Table 4.11-11 in the LAR provides the results of this comparison and shows that calculated offsite doses originating from a site with two TPCs are within the limits of 40 CFR Part 190.

To verify that public radiation doses resulting from radioactive effluent releases will be maintained ALARA, the licensee compared the annual projected impact of TPC operation with 2,496 TPBARs to the numerical guides found in 10 CFR Part 50, Appendix I. Table 4.11-10 provides the results of this comparison and shows that the increase in tritium reactor coolant activity and resultant effluent releases will yield public doses that are within NRC design objectives in 10 CFR Part 50, Appendix I, and thus are considered ALARA.

The NRC staff finds that the licensee has demonstrated that the Watts Bar, Units 1 and 2, TPCs can be operated with 2,496 TPBARs while maintaining the gaseous and liquid effluents and the resultant public doses within the limits of 10 CFR Part 20.

3.1.14.3 Evaluation of Impacts on Occupation Dose

To estimate the impact on dose rates in containment from the operation of a TPC containing 2,496 TPBARs, the licensee applied non-TPC operating experience from Watts Bar, Unit 1, Cycle 3 and Sequoyah Nuclear Plant, Unit 1, Cycle 10. These two cycles consisted of long-term runs that enabled conditions for establishment of a strong correlation between RCS tritium concentration and the airborne tritium in containment. The average RCS tritium concentration value for these cycles was 1 μCi/gram. Consistent with the Watts Bar, Unit 1, tritium production rate during Cycles 11 and 12 with 544 TPBARs, the licensee estimated that the average RCS tritium concentration will be approximately 15.5 μCi/gram at a permeation rate of 5 Ci/TPBAR/year for 2,496 TPBARs. This average value results in an estimated containment tritium Derived Air Concentration-fraction of 1.24 which equates to a containment dose rate of about 3.1 mrem/hour.

The licensee estimated the impact of TPC operation on collective dose by adjusting the historic site average collective, committed effective dose equivalent (CEDE) by a ratio of the expected RCS tritium concentration of 15.5 μCi/gram to the average RCS tritium concentration of 1 μCi/gram from extended cycle operation with a non-TPC. The licensee conservatively attributed all historical CEDE to tritium exposure, even though, realistically, tritium is just one of many nuclides that contributed to historic CEDE. The licensee determined that TPC operation with 2,496 TPBARs will result in an additional 31 person-rem of CEDE. The additional CEDE combined with an additional estimated 1.7 person-rem/year for handling activities, based on Watts Bar, Units 1 and 2, operating experience, yields a total increase in collective CEDE of 32.7 person-rem/year for operation of a TPC with 2,496 TPBARs.

While collective radiation exposure is not a radiation safety standard, it is a useful indicator of the impact that TPC operation will have on the occupational aspects of the Watts Bar, Units 1 and 2, radiation protection program. Consequently, even with the addition of 32.7 person-rem/year of collective radiation exposure, Watts Bar, Units 1 and 2, will remain within the nominal range of contemporary industry performance as it relates to collective radiation exposure, as discussed in NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities" (https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0713/index.html).

To ensure that individual occupational doses are maintained ALARA, the licensee has implemented a tritium bioassay and control program through procedures RCI-137, "Radiation Protection Tritium Control Program," NPG-SPP-05.1, "Radiological Controls," and RCDP-7, "Bioassay and Internal Dose Program." The description of those procedures was provided in a previous LAR (Reference 11) and reviewed by the NRC staff in a prior SE (Reference 6). The licensee stated that the program described and analyzed in the prior LAR remains applicable to this current LAR. The NRC staff verified this by reviewing the procedures during a document audit. The NRC staff finds that the bioassay program is still acceptable because it satisfies the guidance of RG 8.32, "Criteria for Establishing a Tritium Bioassay Program."

The NRC staff finds that the licensee's estimates of the impact of TPC operation with 2,496 TPBARs on occupational dose are reasonable and within the nominal range of contemporary industry performance. Additionally, the NRC staff finds that through the licensee's radiation protection program, as described in the LAR and in previous LARs, sufficient radiological survey and monitoring capabilities will be available to provide reasonable assurance that occupational doses at Watts Bar, Units 1 and 2, with a TPC, can be maintained within the occupational dose limits of 10 CFR Part 20, and that the licensee has implemented procedures to achieve doses that are ALARA.

3.1.14.4 Staff Conclusion Regarding TPBAR Interface Issue 14

The NRC staff concludes that the licensee has adequately addressed the plant-specific factors associated with TPBAR Interface Issue 14 because (1) an acceptable survey program has been established and will continue to be informed by operating experience as evaluated in section 3.2.2 of this SE, and (2) review of the analysis in the prior SE in combination with the updated information in this LAR, the licensee has established adequate responses for operator actions during an abnormal tritium production event. The plant-specific analysis to demonstrate that the plant continuously meets release concentration limits is evaluated in section 3.1.14.2 of this SE.

3.1.15 Evaluation of TPBAR Interface Issue 15 – Process and Effluent Radiological Monitoring and Sampling System

In NUREG-1672, section 2.11.5, the NRC staff determined that a plant-specific assessment of the laboratory instrumentation and sampling frequencies and locations must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to irradiate TPBARs for the production of tritium.

In this LAR, the licensee stated that for TPBAR Interface Issue 15, the established site-specific methodology for analysis of the process and effluent radiological monitoring and sampling system is maintained but is updated to reflect the TPBAR increase to 2,496. The licensee stated that the prior LAR (Reference 11) regarding TPBAR Interface Issue 15 remains applicable to this current LAR, and that no additional sampling points were needed beyond those presently required by the Watts Bar, Units 1 and 2, TS.

In the licensee's previous evaluation, it verified that current techniques for tritium air sampling, liquid monitoring and liquid scintillation counting are appropriate for their effluent radiological monitoring and sampling system and system modifications are not warranted. The NRC staff's previous review found this evaluation acceptable because the licensee considered common release locations, the licensee has an active Ground Water Protection Initiative program per Nuclear Energy Institute 07-07, "Ground Water Protection Initiative," and the licensee will review and pursue system modifications as required based on operating experience (Reference 6).

The NRC staff finds that the licensee adequately addressed plant-specific factors associated with TPBAR Interface Issue 15 because of the adequacy of the licensee's past reviews of potential release points and because it has a program for continuously assessing the adequacy of its process and effluent monitoring and sampling program. Additionally, the NRC staff finds that the licensee still applies adequate methods of analyzing air and liquid samples containing tritium during normal operations, as well as during fuel handling evolutions.

3.1.16 Evaluation of TPBAR Interface Issue 16 – Use of LOCTA_JR Code for LOCA Analyses

In NUREG-1672, section 2.15.5, the NRC staff concluded that although the LOCTA_JR code was appropriate for use in the demonstration analyses and assessments, LOCTA_JR was not reviewed for licensing use and should be reviewed by the staff for licensing applications and for its interface with the specific plant licensing LOCA models before it is used in specific plant licensing applications.

The NRC staff previously reviewed and approved the small-break LOCA and large-break LOCA analyses for Watts Bar, Units 1 and 2, based on the 2016 Westinghouse FULL SPECTRUM™ LOCA (FSLOCA) Evaluation Model (Reference 29), which is supported by a TPBAR structural integrity analysis based on the FSLOCA Evaluation Model (Reference 30). Neither the LOCA analyses nor the TPBAR structural integrity analysis explicitly uses the LOCTA_JR code. Therefore, the NRC staff determines that TPBAR Interface Issue 16 is not applicable to these analyses.

3.1.17 Evaluation of TPBAR Interface Issue 17 – ATWS Analysis

In NUREG-1672, section 2.15.7, the NRC staff concluded that that licensees seeking to utilize a TPC must submit a plant-specific application containing a full anticipated transient without scram (ATWS) analysis, conducted in accordance with NRC regulations and approved standards.

An ATWS is an anticipated operational occurrence during which an automatic reactor scram is required but fails to occur because some common mode fault in the reactor protection system.

TVA previously submitted plant-specific ATWS analyses for Watts Bar, Units 1 and 2 (References 31 and 11, respectively), to address this interface issue. The Unit 1 analysis included a comparison of full power moderator temperature coefficients (MTCs) as a function of cycle length for TPCs and non-TPCs. The variability was shown to be small, with a trend towards more negative MTCs at the beginning of the cycle for TPCs. This is due to the fact that for the TPC cores being modeled, the negative reactivity from the TPBARs limits the amount of soluble boron required to maintain criticality. This lower soluble boron concentration results in more negative moderator feedback, which results in additional margin to the ATWS overpressurization event.

The NRC staff determined that the insertion of 2,496 TPBARs into the Watts Bar, Units 1 and 2, cores has sufficient margin to the ATWS overpressurization event based on the ATWS analyses previously submitted by the licensee for Watts Bar, Units 1 and 2, and the fact that the TPBARs do not appear to have an impact that would alter the core response to an ATWS event. The NRC staff finds that the licensee has satisfactorily addressed TPBAR Interface Issue 17.

3.1.18 Evaluation of Post-LOCA Subcriticality

The post-LOCA subcriticality analysis supports evaluations for each reload core to demonstrate that the core will remain subcritical during the reflood phase and during the cold-leg and hot-leg sump recirculation phases of emergency core cooling system operation. An additional analysis is performed for the long-term cooling phase of hot-leg recirculation.

In Reference 32, the licensee requested approval to use the FSLOCA evaluation model to evaluate the peak cladding temperatures for large-break and small-break LOCAs. This LAR requested approval to use separate simulations performed in accordance with the FSLOCA evaluation model as part of the new TPBAR stress analysis methodology developed by Pacific Northwest National Laboratory and Westinghouse to provide a recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures. The LOCA-specific

TPBAR stress analysis methodology relies on conditions resulting from LBLOCA simulations according to FSLOCA methodology. The staff approved all the aspects of the TPBAR stress analysis for structural integrity such as Watts Bar, Units 1 and 2, core model, cladding stress analysis acceptance criteria, conservatism built into the integrity analysis, post-LOCA criticality analysis, and uncertainty analysis.

The NRC staff determined that the application of the new TPBAR stress analysis methodology demonstrates that TPBAR integrity will be maintained following an LBLOCA. The presence of intact TPBARs is credited in the post-LOCA criticality evaluation as a negative reactivity contribution. The post-LOCA subcriticality margin is increased. The standard reload methodology for a core containing 2,496 TPBARs confirms that post-LOCA subcriticality is maintained.

3.2 Evaluation of Proposed Changes to Technical Specifications

3.2.1 Evaluation of Proposed Changes to TS 4.2.1, “Fuel Assemblies”

Section 50.36(c)(4) of 10 CFR requires that TS include Design Features, which are defined as those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in safety limits, limiting safety systems settings, limiting control settings, LCOs, or SRs. TS 4.2.1, “Fuel Assemblies,” contains the Design Features information for Fuel Assemblies. As described in section 2.2 of this SE, TVA proposed to revise Watts Bar, Units 1 and 2, TS 4.2.1 to increase the maximum number of TPBARs allowed in the reactor in an operating cycle from 1,792 to 2,496.

The NRC staff reviewed the proposed changes. The information submitted by the licensee, as evaluated by the NRC staff in section 3.1 of this SE, demonstrates that all applicable regulatory requirements are met for this change. The modified sentence in TS 4.2.1 for each unit authorizing the placement of a maximum of 2,496 TPBARs into the reactor in an operating cycle is necessary to ensure TS 4.2.1, as amended, continues to contain a description of the fuel assembly materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety which are not covered in safety limits, LCOs or SRs, as required by 10 CFR 50.36(c)(4). Therefore, the staff finds the proposed change to Watts Bar, Units 1 and 2, TS 4.2.1 acceptable based on this review.

3.2.2 Evaluation of Proposed Changes to TS 5.9.6, “Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)”

On June 17, 2021, the NRC approved a license amendment (Reference 33) that revised Watts Bar, Unit 2, TS 5.9.6.b to add WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” as a reference approved methodology for the Unit 2 PTLR. As described in section 2.2 of this SE, the LAR proposes to add WCAP-18124-NP-A, Revision 0, Supplement 1-NP-A, Revision 0, to the Unit 2 TS 5.9.6.b because it provides the justification necessary to allow the licensee to apply the RAPTOR-M3G method in the extended beltline regions of RPVs on a generic basis.

In addition, the LAR proposes to modify Watts Bar, Unit 1, TS 5.9.6 to completely align with proposed Unit 2, TS 5.9.6 (i.e., specify the same requirements and cite the same methodologies to be used for the PTLR). The LAR also proposes to delete existing references 1, 2, 3, 4, and 5.

The proposed changes delete the previously approved PTLR methodologies for Unit 1 and replaces them with the NRC-approved PTLR methodologies that were previously approved for Unit 2: (1) WCAP-14040-A, Rev. 4, and (2) WCAP-18124-NP-A, Rev. 0. In addition, similar to Unit 2, the LAR proposes to add WCAP-18124-NP-A. Rev. 0 Supplement 1-NP-A, Rev. 0 to TS 5.9.6.b for Unit 1. In section 3.1.4.1 of this SE, the NRC staff found the use of these NRC approved methodologies to be applicable to Watts Bar, Units 1 and 2, and that the licensee satisfactorily addressed the limitations and conditions established for these methodologies. Therefore, the NRC staff finds that they are acceptable for determining the pressure and temperature limits for Watts Bar, Units 1 and 2.

In a letter dated August 4, 2011, from the NRC to the Technical Specifications Task Force (Reference 34), the NRC informed the TSTF of the following:

> The NRC staff will no longer accept license amendment requests (LARs) to implement technical specification (TS) changes in accordance with Traveler TSTF-363. The standard TS (STS) in NUREGs-1430, -1431, -1432, -1433, and -1434 will be revised in Revision 4 to reflect the TS the way they were prior to approval of TSTF-363.
>
> The NRC staff will continue to accept LARs to implement the TS changes in accordance with Travelers TSTF-408 and TSTF-419 with modification. Currently the STS show the full topical report or methodology citation as located in the PTLR. In order for NRC staff to approve LARs for these two travelers, the full topical report or methodology citation will need to be included in the TS, not in the PTLR. The STS in NUREGs-1430, -1431, -1432, -1433, and -1434 will be revised in Revision 4 to reflect this change.

Both units have TSs that, consistent with the STS, require the licensee to list the analytical methods used to determine RCS P/T limits. Accordingly, the NRC staff compared the proposed changes with the guidance in the latest applicable STS, which for Watts Bar is NUREG-1431, “Standard Technical Specifications, Westinghouse Plants, Revision 5.0, Volume 1” (Reference 10), and the August 4, 2011, letter to the TSTF. Because Watts Bar, Unit 1, is adopting WCAP-18124-NP-A, Revision 0, as the neutron fluence calculational methodology for the evaluation of reactor vessel specimens to support the determination of the RCS P-T limits, and WCAP-14040-A, Rev. 4 as the methodology for calculating COMS setpoints and RCS heatup and cooldown limit curves, it is appropriate for these methodologies to be added to TS 5.9.6.a and 5.9.6b. Similarly, Watts Bar, Units 1 and 2, are adopting WCAP-18124-NP-A, Revision 0, Supplement 1-NP-A, Revision 0 to adjust the applicability of the neutron fluence calculational methodology for the region of the RPV near the active core. Since these methodologies will be used by Watts Bar, Units 1 and 2, as part of the neutron fluence calculational methodology for the evaluation of reactor vessel specimens to support the determination of the RCS P-T limits, it is appropriate that they be added to TS 5.9.6.b for each unit. For each of the methodologies discussed above, the licensee has proposed adding the complete methodology citation in TS 5.9.6. This is consistent with the guidance in the STS and in the August 4, 2011, letter. Therefore, the NRC staff finds that these changes are acceptable.

The NRC staff evaluated the proposed changes to the Watts Bar, Unit 1, TS 5.9.6.a. The added text clarifies that the PTLR establishes and documents the power operated relief valve lift settings required to support the COMS and the COMS arming temperature, in addition to the current requirements for RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates for LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” and LCO 3.4.12, “Cold Overpressure Mitigation System (COMS).” Therefore, the NRC staff finds that the change is an administrative change and is acceptable.

The NRC staff reviewed the proposed TS changes for technical clarity and consistency with the requirements for customary terminology and formatting. The staff finds that the proposed TS changes are consistent with Chapter 16.0 of the SRP and NUREG-1431. In addition, 10 CFR 50.36(c)(5) requires that TS include administrative controls. Per the regulation, “administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.” The staff’s review of TS 5.9.6, as amended for Watts Bar, Units 1 and 2, will continue to include provisions and reporting requirements necessary to assure operation of the facility in a safe manner. Based on this, the NRC staff finds that the amended TS 5.9.6 will continue to meet 10 CFR 50.36(c)(5) and is acceptable.

3.3 Technical Conclusion

The NRC staff evaluated the plant-specific TPBAR Interface Issues. As documented in SE sections 3.1.1 through 3.1.17, the staff found that the licensee adequately addressed the TPBAR Interface Issues. The staff also found that the standard reload methodology for the Watts Bar, Units 1 and 2, cores containing 2,496 TPBARs confirms that post-LOCA subcriticality will be maintained. Regarding gaseous and liquid effluents, the NRC staff found that the operation of Watts Bar, Units 1 and 2, with 2,496 TPBARs will not result in public doses that exceed the limits of 10 CFR Part 20. Additionally, the staff found that the licensee’s estimates of the impact of TPC operation with 2,496 TPBARs can be maintained within the occupational dose limits of 10 CFR Part 20. Lastly, the staff found that the proposed changes to the TSs, as revised, will continue to meet the regulations in 10 CFR 50.36(c)(4) and (c)(5), and the proposed revision to the Watts Bar Dual-Unit UFSAR is acceptable.

Therefore, the NRC staff concludes, with reasonable assurance, that the licensee has demonstrated that the Watts Bar, Units 1 and 2, cores can be operated with up to 2,496 TPBARs, within its CLB, and without exceeding the public and occupation dose limits in 10 CFR Part 20.

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 February 2, 2024. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact regarding this license amendment was published in the *Federal Register* on February 23, 2024 (89 FR 13757). Based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

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. Tennessee Valley Authority (TVA) Letter CNL-23-002, "Application to Revise Watts Bar Nuclear Plant Units 1 and 2 Technical Specifications to Change the Number of Tritium Producing Burnable Absorber Rods (WBN-TS-21-02) and Proposed Revision to Reactor Vessel Surveillance Capsule Removal Schedule for Units 1 and 2," dated March 20, 2023 (Agencywide Documents Access and Management System Accession No. ML23079A270).
2. U.S NRC e-mail from Kimberly Green to Russell Wells, Subject: Audit Plan Related to Review of the Watts Bar Nuclear Plant, Units 1 and 2, License Amendment Request to Increase the Number of Tritium Producing Burnable Absorber Rods (EPID L-2023-LLA-0039), dated May 25, 2023 (ML23150A247).
3. U.S. DOE, Topical Report NPD-98-191, Revision 1, "Tritium Production Core (TPC) Topical Report (Unclassified, Non-Proprietary Version)," dated February 8, 1999 (ML16077A093).
4. U.S. NRC, NUREG-1672, "Safety Evaluation Report related to the Department of Energy's topical report on the tritium production core," dated May 1999 (ML20209H927).
5. U.S NRC letter to Joseph W. Shea, TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Revised Technical Specification 4.2.1 'Fuel Assemblies' to Increase the Maximum Number of Tritium Producing Burnable Absorber Rods (CAC No. MF6050)," dated July 29, 2016 (ML16159A057).
6. U.S. NRC letter to Joseph W. Shea, TVA, "Watts Bar Nuclear Plant, Units 1 and 2 – Issuance of Amendment Regarding Revision to Watts Bar Nuclear Plant, Unit 2, Technical Specification 4.2.1, 'Fuel Assemblies,' and Watts Bar Nuclear Plant, Units 1 and 2, Technical Specifications Related to Fuel Storage (EPID L-2017-LLA-0427)," dated May 22, 2019, (ML18347B330).
7. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 4, Section 4.2, "Fuel System Design," Revision 3, dated March 2007 (ML070740002).
8. NUREG-0800, Chapter 13, "Conduct of Operations," Section 13.5.2.1, "Operating and Emergency Operating Procedures," Revision 2, dated March 2007 (ML070100635).
9. NUREG-0800, Chapter 16.0, "Technical Specifications," Revision 3, dated March 2010 (ML100351425).
10. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 5.0, Volume 1, Specifications, dated September 2021 (ML21259A155).
11. TVA letter CNL-17-144, "Application to Revise Watts Bar Unit 2 Technical Specification 4.2.1, 'Fuel Assemblies,' and Watts Bar Units 1 and 2 Technical Specifications Related to Fuel Storage (WBN-TS-17-028)," dated December 20, 2017 (ML17354B282).
12. U.S. NRC letter to James Barstow, TVA, "Watts Bar Nuclear Plant, Units 1 and 2 – Regulatory Audit Summary Related to Request to Increase the Number of Tritium Producing Burnable Absorber Rods," dated January 3, 2024 (ML23346A138).

13. NUREG-0800, Chapter 4, Section 4.4, “Thermal and Hydraulic Design,” Revision 2, dated March 2007 (ML070550060).
14. Westinghouse Electric Company, LLC (Westinghouse)WCAP-14565-P-A/WCAP-15306-NP-A, “VIPRE-01 Modeling Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” dated October 1999 (package ML993160101, document ML993160096).
15. Westinghouse Report, WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” dated July 2018 (ML18204A010).
16. Westinghouse Report, WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials,” dated May 31, 2022 (ML22153A139).
17. U.S. NRC, Regulatory Issue Summary 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” dated October 14, 2014 (ML14149A165).
18. TVA Letter WBL-21-056, “Watts Bar Nuclear Plant Unit 1 – Revised Pressure and Temperature Limits Report (PTLR),” dated December 9, 2021 (ML21343A151).
19. TVA Letter WBL-23-018, “Watts Bar Nuclear Plant Unit 2 – Revised Pressure and Temperature Limits Report (PTLR),” dated April 10, 2023 (ML23100A044).
20. Westinghouse Report, WCAP-15046, “Analysis of Capsule U from the Tennessee Valley Authority Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program,” dated June 1998 (ML073460336).
21. U.S. NRC, Branch Technical Position 5-3, “Fracture Toughness Requirements,” Revision 4, dated March 2019 (ML18338A516).
22. U.S. NRC letter to Oliver Kingsley, TVA, “Watts Bar Nuclear Plant – Generic Letter 92-01, Revision 1, ‘Reactor Vessel Structural Integrity’ and Low Upper Shelf Energy Evaluation,” dated May 11, 1994 (ML073230653).
23. TVA Letter WBL-20-004, “Watts Bar Nuclear Plant (WBN) Unit 2 – Analysis of Capsule U from Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program,” dated April 16, 2020, transmitting the Westinghouse Report, WCAP-18518-NP, Revision 0, “Analysis of Capsule U from the Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program,” dated March 2020 (ML20107F717).
24. Westinghouse Report, WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” dated May 2004 (ML050120209).
25. U.S. NRC letter to James Barstow, TVA, “Watts Bar Nuclear Plant, Units 1 and 2 – Revision to the Reactor Vessel Material Surveillance Capsule Withdrawal Schedule (EPID L-2023-LLA-0039),” dated January 18, 2024 (ML24008A246).
26. TVA letter, “Watts Bar Nuclear Plant (WBN) Unit 1 – Analysis of Capsule Z from Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program,” dated November 9, 2007 (package ML073200241).
27. U.S. NRC letter to Gordon Bischoff, Westinghouse, “Final Safety Evaluation for Topical Report WCAP-14040, Revision 3, ‘Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RC Heatup and Cooldown Limit Curves’ (TAC No. MB5754),” dated February 27, 2004 (ML040620297).
28. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors,” Revision 1, dated April 1985 (ML112720411).
29. Westinghouse Report, WCAP-16996-NP-A, Revision 1, “Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology),” dated November 2016 (ML16343A235).
30. U.S. NRC letter to James Barstow, TVA, “Watts Bar Nuclear Plant, Units 1 and 2 – Issuance of Amendment Nos. 143 and 50 Regarding Implementation of Full Spectrum™ Loss-of-

Coolant-Accident Analysis (LOCA) and New LOCA-Specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology," dated February 26, 2021 (ML21034A169).
31. TVA letter, "Watts Bar Nuclear Plant - Tritium Production Program - Anticipated Transients Without Scram (ATWS)," dated September 29, 2000 (ML003759282).
32. TVA letter CNL-21-010, "Correction of Application to Implement the FULL SPECTRUM™ LOCA (FSLOCA) Methodology for Loss-of-Coolant Accident (LOCA) Analysis and New LOCA-specific Tritium Producing Burnable Absorber Rod Stress Analysis Methodology (WBN-TS-19-04) (EPID L-2020-LLA-0005)," dated January 26, 2021 (ML21027A143).
33. U.S. NRC letter to James Barstow, TVA, "Watts Bar Nuclear Plant, Unit 2 – Issuance of Amendment No. 53 Regarding Neutron Fluence Calculation Methodology (EPID L-2020-LLA-0167), dated June 17, 2021 (ML21148A100).
34. U.S. NRC letter to the Technical Specifications Task Force, "Implementation of Travelers TSTF-363, Revision 0, 'Revise Topical Report References in ITS 5.6.5, COLR [Core Operating Limits Report],' TSTF-408, Revision 1, 'Relocation of LTOP [Low-Temperature Overpressure Protection] Enable Temperature and PORV [Power-Operated Relief Valve] Lift Setting to the PTRL [Pressure-Temperature Limits Report],' and TSTF-419, Revision 0, 'Revise PTRL Definition and References in ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR,'" dated August 4, 2011 (ML110660285).

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Date: April 15, 2024

SUBJECT: WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 165 AND 72 REGARDING INCREASE IN THE MAXIMUM NUMBER OF TRITIUM PRODUCING BURNABLE ABSORBER RODS AND SUPPORTING CHANGES, AND REVISION TO THE UPDATED FINAL SAFETY ANALYSIS REPORT (EPID L-2023-LLA-0039) DATED APRIL 15, 2024

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