TENNESSEE VALLEY AUTHORITY

1101 Market Street, Chattanooga, Tennessee 37402

CNL-23-002

March 20, 2023

10 CFR 50.90
10 CFR 50 Appendix H

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

Watts Bar Nuclear Plant, Units 1 and 2
Facility Operating Licenses Nos. NPF-90 and NPF-96
NRC Docket Nos. 50-390 and 50-391

Subject: 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

References:

1. NRC letter to 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)

2. NRC letter to 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)

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," Tennessee Valley Authority (TVA) is submitting a request for an amendment to Facility Operating License Nos. NPF-90 and NPF-96 for the Watts Bar Nuclear Plant (WBN), Units 1 and 2, respectively.

The proposed change revises WBN Units 1 and 2 Technical Specification (TS) 4.2.1, 'Fuel Assemblies,' to change the number of tritium producing burnable absorber rods (TPBAR) that can be irradiated from 1,792 TPBARs to 2,496 TPBARs for each WBN unit. The license amendment request (LAR) also proposes a supporting change to WBN Units 1 and 2 TS 5.9.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

This change is analogous to the changes approved by the Nuclear Regulatory Commission (NRC) for increasing the maximum number of TPBARs in WBN Unit 1 License Amendment 107

(Reference 1) and WBN Unit 2 License Amendment 27 (Reference 2). Therefore, only relevant changes from References 1 and 2 to support the proposed increase in TPBARs for WBN Units 1 and 2 are addressed in this LAR.

Enclosure 1 to this letter provides the Evaluation of the Proposed Change including the description and assessment of the proposed change, regulatory analyses, and environmental considerations. Attachment 1 to Enclosure 1 provides the existing WBN Unit 1 TS pages marked-up to show the proposed changes. Attachment 2 to Enclosure 1 provides the existing WBN Unit 2 TS pages marked-up to show the proposed changes. Attachment 3 to Enclosure 1 provides the retyped WBN Unit 1 TS pages incorporating the proposed changes. Attachment 4 to Enclosure 1 provides the retyped WBN Unit 2 TS pages incorporating the proposed changes. Attachment 5 to Enclosure 1 provides a marked-up version of the WBN Units 1 and 2 TS 3.4.3 Bases. Changes to the existing TS Bases are provided for information only and will be implemented under the TS Bases Control Program. In support of TPBAR Interface Issue 5, “Control Room Habitability Systems,” (Section 4.4 to Enclosure 1), Attachment 6 to Enclosure 1 contains a revision to the WBN dual-unit Updated Final Safety Analysis Report (UFSAR) that TVA has determined requires prior NRC approval. Specifically, the proposed UFSAR change modifies 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.

Additionally, pursuant to 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Paragraph III.B.3, TVA is requesting NRC approval of a revision to the reactor vessel surveillance capsule removal schedule for WBN Units 1 and 2, as noted in Section 4.3 to Enclosure 1, regarding TPBAR Interface Issue 4.

NRC Administrative Letter (AL) 97-04 clarified the submittal requirements of 10 CFR 50, Appendix H. As stated in AL 97-04, "In this instance, as long as the plant's withdrawal schedule change meets the applicable ASTM standard, the plant will not be exceeding the operating authority already granted in its license. Therefore, a license amendment would not be required, although prior NRC approval to verify conformance with the ASTM standard is required by Appendix H." American Society for Testing and Materials (ASTM) E-185-82 is the applicable standard for WBN as described in Section 5.2.4.3 of the WBN dual-unit UFSAR. Because the proposed change described in Section 4.3 to Enclosure 1 satisfies the requirements of ASTM E-185-82, TVA has determined that a license amendment is not required for the proposed change to the reactor vessel surveillance capsule removal schedule for WBN Units 1 and 2, which is consistent with the guidance of AL 97-04. In support of the proposed revision to the reactor vessel surveillance capsule removal schedule for WBN Units 1 and 2, Enclosure 2 to this letter provides a copy of WCAP-18769-NP, Revision 1, “Watts Bar Units 1 & 2 Reactor Vessel Integrity Evaluations for the 2,496 TPBAR Implementation Project.”

TVA requests approval of the proposed license amendment within one year from the date of this submittal with implementation within 60 days of issuance of the amendment. In accordance with WBN Units 1 and 2 TS 5.9.6.c, TVA will submit the revised WBN PTLR to the NRC following NRC approval of the proposed revision to the WBN Units 1 and 2 reactor vessel surveillance capsule withdrawal schedule.

The proposed change is required to support a planned increase of TPBAR inventory in the WBN Unit 1 Cycle 19 refueling outage (U1R19) in fall 2024 and WBN Unit 2 U2R6 in spring 2025 to support Department of Energy requests.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS changes qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). In accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosure to the Tennessee State Department of Environment and Conservation.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to slrymer@tva.gov.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 20th day of March 2023.

Respectfully,

Digitally signed by Edmondson, Carla
Date: 2023.03.20 07:10:04 -04'00'

Kimberly D. Hulvey
Director, Nuclear Regulatory Affairs

Enclosures:

1. Evaluation of Proposed Change
2. WCAP-18769-NP, Revision 1

cc (Enclosures):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Watts Bar Nuclear Plant
NRC Project Manager – Watts Bar Nuclear Plant
Director, Division of Radiological Health – Tennessee State Department of Environment and Conservation

Enclosure 1

Evaluation of Proposed Change

Subject: 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

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1.0 SUMMARY DESCRIPTION

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, “Application for amendment of license, construction permit, or early site permit,” Tennessee Valley Authority (TVA) is submitting a request for amendments to Facility Operating License Nos. NPF-90 and NPF-96 for the Watts Bar Nuclear Plant (WBN), Units 1 and 2, respectively. The proposed change revises WBN Units 1 and 2 Technical Specification (TS) 4.2.1, ‘Fuel Assemblies,’ to change the number of tritium producing burnable absorber rods (TPBARs) that can be irradiated. The proposed change will authorize the irradiation of up to a maximum of 2,496 TPBARs in WBN Units 1 and 2.

This change is analogous to the changes approved by the Nuclear Regulatory Commission (NRC) for increasing the maximum number of TPBARs in WBN Unit 1 License Amendment 107 (Reference 1 of Section 7 to this enclosure) and WBN Unit 2 License Amendment 27 (Reference 2 of Section 7 to this enclosure). Therefore, only relevant changes to support the proposed increase in TPBARs for WBN Units 1 and 2 from References 1 and 2 of Section 7 to this enclosure are addressed in this license amendment request (LAR). This LAR also proposes a supporting change to WBN Units 1 and 2 TS 5.9.6, “Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR).”

The proposed change also contains a revision to the WBN dual-unit Updated Final Safety Analysis Report (UFSAR) that TVA has determined requires prior NRC approval. Specifically, the proposed UFSAR change modifies 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 proposed change is required to support a planned increase of TPBAR inventory in the WBN Unit 1 Cycle 19 refueling outage (U1R19) in fall 2024 and WBN Unit 2 U2R6 in spring 2025 to support Department of Energy (DOE) requests.

Additionally, TVA is requesting NRC approval of a revision to the reactor vessel (RV) surveillance capsule removal schedule for WBN Units 1 and 2, as noted in Section 4.3 to Enclosure 1 regarding TPBAR Interface Issue 4, pursuant to 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Paragraph III.B.3. NRC Administrative Letter (AL) 97-04, “NRC Staff Approval for Changes to 10 CFR Part 50, Appendix H, Reactor Vessel Surveillance Specimen Withdrawal Schedules,” clarified the submittal requirements of 10 CFR 50, Appendix H. As stated in AL 97-04, "In this instance, as long as the plant's withdrawal schedule change meets the applicable ASTM standard, the plant will not be exceeding the operating authority already granted in its license. Therefore, a license amendment would not be required, although prior NRC approval to verify conformance with the ASTM standard is required by Appendix H." ASTM E-185-82 is the applicable standard for WBN as described in Section 5.2.4.3 of the WBN dual-unit UFSAR. Because the proposed change described in Section 4.3 to Enclosure 1 satisfies the requirements of ASTM E-185-82, TVA has determined that a license amendment is not required for the proposed change to the RV surveillance capsule removal schedule for WBN Units 1 and 2, which is consistent with the guidance of AL 97-04.

2.0 DETAILED DESCRIPTION

This LAR revises WBN Units 1 and 2 TS 4.2.1 as described below.

WBN Unit 1 TS 4.2.1

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 (UO2) 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, WBN is authorized to place a maximum of ~~1,792~~ 2,496 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

WBN Unit 2 TS 4.2.1

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 (UO2) 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, WBN is authorized to place a maximum of ~~1,792~~ 2,496 Tritium Producing Burnable Absorber Rods into the reactor in an operating cycle.

The LAR also revises WBN Unit 1 TS 5.9.6 to be consistent with WBN Unit 2 TS 5.9.6. WBN Units 1 and 2 TS 5.9.6 are also revised to add WCAP-18124-NP-A Rev. 0 Supplement 1-NP-A, Rev. 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials." Further discussion on these changes is provided below.

- In References 2.0-1 and 2.0-2, NRC approved a license amendment for WBN Unit 2 to revise WBN Unit 2 TS 5.9.6.b to add WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” as a neutron fluence calculational methodology for the evaluation of reactor vessel specimens to support the determination of reactor coolant system pressure and temperature limits. Similarly, WBN Unit 1 TS 5.9.6.b is being revised to also add WCAP-18124-NP-A, Revision 0. The justification for use of WCAP-18124-NP-A, Revision 0, including conformance to the NRC limitations and conditions, in References 2.0-1 and 2.0-2 is applicable to the proposed change to WBN Unit 1 TS 5.9.6.b. The applicability of Limitation #1 of WCAP-18124-NP-A, Revision 0 limiting the applicability of the methodology to the region of the reactor pressure vessel (RPV) near the active core is adjusted via implementation of WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A, Revision 0.

Limitation #2 of WCAP-18124-NP-A, Revision 0 will be applied if least squares adjustment is employed.

- WBN Units 1 and 2 TS 5.9.6.b are also revised to add WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A, Revision 0. WCAP-18124-NP-A, Revision 0, Supplement 1-NP-A, Revision 0, provided the justification necessary to narrow Limitation #1 and allow licensees to apply the RAPTOR-M3G method in the extended beltline regions of RPVs on a generic basis. The NRC has determined that the fluence methods and qualifications described in WCAP-18124-NP-A, Revision 0, Supplement 1-NP-A are acceptable for referencing in licensing applications as described in the NRC final safety evaluation (Reference 2.0-3). As noted in Reference 2.0-3, Supplement 1-NP-A provides additional methodology requirements and qualification data to justify the application of the RAPTOR-M3G and FERRET fluence methods to the extended beltline. Also in Reference 2.0-3, the NRC reviewed WCAP-18124-NP-A, Revision 0, Supplement 1-NP-A, Revision 0, and determined that appropriate modeling techniques and adequate qualification were provided to apply RAPTOR-M3G to determine neutron fluence in the reactor vessel extended beltline and that the modeling techniques adhere to the guidance in Regulatory Guide (RG) 1.190, as appropriate, and exceed it when necessary.
- WBN Unit 1 TS 5.9.6.b is also being revised to add WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” consistent with WBN Unit 2 TS 5.9.6.b.1 and revising the references to be consistent with WBN Unit 2 TS 5.9.6.b. The NRC has found the methodology contained in WCAP-14040-A acceptable for referencing in licensing applications subject to three conditions as described in the NRC final safety evaluation (Reference 2.0-4). Revision 3 of WCAP-14040-A was reissued in May of 2004 as Revision 4 (Reference 2.0-5) with the referenced NRC acceptance letter and safety evaluation incorporated. In Reference 2.0-6, the NRC evaluated the use of WCAP-14040-A, Revision 4 in the licensing of WBN Unit 2. The same methodology and application of WCAP-14040-A, Revision 4 described in Reference 2.0-6 also applies to WBN Unit 1. The use of WCAP-14040-A, Revision 4 was also described in TPBAR Interface Issue 4, “Reactor Vessel Integrity,” in Reference 2.0-7.

Attachment 1 to Enclosure 1 provides the existing WBN Unit 1 TS pages marked-up to show the proposed changes. Attachment 2 to Enclosure 1 provides the existing WBN Unit 2 TS pages marked-up to show the proposed changes. Attachment 3 to Enclosure 1 provides the retyped WBN Unit 1 TS pages incorporating the proposed changes. Attachment 4 to Enclosure 1 provides the retyped WBN Unit 2 TS pages incorporating the proposed changes. Attachment 5 to Enclosure 1 provides a marked-up version of the WBN Units 1 and 2 TS 3.4.3 Bases. Changes to the existing TS Bases are provided for information only and will be implemented under the Technical Specification Bases Control Program. In support of Interface Issue 5, “Control Room Habitability Systems,” (Section 4.4 to this enclosure) Attachment 5 to Enclosure 1 contains a revision to the WBN dual-unit UFSAR that TVA has determined requires prior NRC approval.

References

2.0-1 TVA letter to NRC, CNL-20-008, "Application to Modify Watts Bar Nuclear Plant (WBN) Unit 2 Technical Specification 5.9.6, 'Reactor Coolant System Pressure and Temperature Limits Report,' (WBN-TS-19-28)," dated July 27, 2020 (ML20209A071)

2.0-2 NRC letter to 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)

2.0-3 Westinghouse Electric Company letter to NRC, LTR-NRC-22-19, "Submittal of WCAP-18124-NP-A, Revision 0, Supplement 1-P/NP-A 'Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials' (Proprietary/Non-Proprietary)," dated May 26, 2022 (ML22153A136)

2.0-4 NRC letter to Westinghouse Electric Company, "Final Safety Evaluation for Topical Report WCAP-14040, Revision 3, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves' (TAC No. MB5754)," dated February 27, 2004 (ML040620297)

2.0-5 WCAP-14040, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," dated May 2004 (ML050120209)

2.0-6 NUREG-0847, Supplement 22, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2," dated February 2011 (ML110390197)

2.0-7 TVA letter to NRC, 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)

3.0 BACKGROUND

The DOE and TVA have agreed to cooperate in a program to produce tritium for the National Security Stockpile by irradiating TPBARs at WBN Units 1 and 2.

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 component consisting of zirconium alloy material in the TPBAR (called a "getter") captures the produced tritium. Most of the tritium is contained within the TPBAR. However, a small fraction of the tritium permeates through the TPBAR cladding into the reactor coolant system. After the TPBARs are removed from the core and shipped to a DOE extraction facility, the TPBARs are heated in a vacuum at high temperature to extract the tritium.

WBN Unit 1 License Amendment 107 (Reference 3.0-1) approved the irradiation of up to 1,792 TPBARs in WBN Unit 1. WBN Unit 2 License Amendment 27 (Reference 3.0-2) approved the irradiation of up to 1,792 TPBARs in WBN Unit 2. WBN Unit 1 License Amendment 143 and WBN Unit 2 License Amendment 50 (Reference 3.0-3) approved the use of the loss-of-coolant accident (LOCA) specific TPBAR stress analysis methodology to evaluate the integrity of the TPBARs for the conditions expected during a large break LOCA and provide a recovery of margin in the post-LOCA criticality evaluation in the presence of assumed TPBAR failures.

As described in this LAR, TVA is requesting approval to irradiate up to 2,496 TPBARs in WBN Units 1 and 2. The number of TPBARs to be irradiated in any given operating cycle are evaluated in the reload safety evaluation and documented in the core operating limits report (COLR). The number of TPBARs will not exceed 2,496.

<u>References</u>

3.0-1 NRC letter to 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)

3.0-2 NRC letter to 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)

3.0-3 NRC letter to 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 (EPID L-2020-LLA-0005),” dated February 26, 2021 (ML21034A166 and ML21034A169)

4.0 TECHNICAL EVALUATION

This proposed change is justified based on analysis and evaluation of the impact of the increased number of irradiated TPBARs. The affected 17 plant-specific interface items from NUREG-1672 (Reference 4.0-1) are addressed for WBN Units 1 and 2. The following is a listing of the NUREG-1672 interface items along with the section number where these items are addressed in this enclosure.

1. Handling of TPBARs (4.1)
3. Compliance with DNB Criterion (4.2)
4. Reactor Vessel Integrity Analysis (4.3)
5. Control Room Habitability Systems (4.4)
7. Light-Load Handling System (4.5)
8. Station Service Water System (4.6)
9. Ultimate Heat Sink (4.7)
11. Spent Fuel Pool Cooling and Cleanup System (4.8)
12. Component Cooling Water System (4.9)
13. Demineralized Water Makeup System (4.10)
14. Liquid Waste Management System (4.11)
15. Process and Effluent Radiological Monitoring and Sampling System (4.12)
16. Use of LOCTA_JR Code for LOCA Analyses (4.13)
17. ATWS Analysis (4.14)

Each of the above sections contains one or more quotes from NUREG-1672 followed by a discussion of the plant-specific evaluation of the interface item. There are no substantive changes to Interface Issues 2, “Procurement and Fabrication Issues,” 6, “Specific Assessment of Hydrogen Source and Timing or Recombiner Operation,” and 10, “New and Spent Fuel Storage,” as described in Reference 4.0-2 and they remain applicable to this LAR. Therefore, Interface Issues 2, 6, and 10 are not discussed in the following sections.

<u>Reference</u>

4.0-1 NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report on the Tritium Production Core," May 1999 (ML20209H927)

4.0-2 TVA letter to NRC, 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)

4.1 TPBAR Interface Issue 1: Handling of TPBARs

NUREG-1672, Section 1.3, "DOE did not address the activities required to remove the TPBARS from the fuel assemblies and prepare them for shipment because these activities are dependent on the fuel pool design. Therefore, the staff has identified this as an interface item that 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."

NUREG-1672, Section 2.9.2, "In addition, DOE did not address the activities required to remove the TPBARs from the fuel assemblies and prepare them for shipment because these activities are dependent on the fuel pool design. Therefore, the staff has identified this as an interface item that 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."

NUREG-1672, Section 3.7, "DOE has described the consequences of potential handling damage resulting from refueling operations and during onsite fuel assembly movement and handling with TPBARs installed. If an irradiated TPBAR is breached because of mishandling in the spent fuel pool, only a small fraction of the tritium inventory would be released. The tritium in the open pores of the pellet (tens of Ci) will be released when water comes in contact with the pellet. Further release may occur gradually due to the limited leaching of the pellets and would provide adequate time to isolate the damaged TPBAR cluster to prevent further release into the pool. DOE did not address post-irradiation movement of the TPBARs outside of fuel assemblies. Therefore, the staff has identified this as an interface item that 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 information regarding Interface Issue 1 provided in Reference 4.1-1 applies to this LAR except as provided below.

- TVA has determined that 2,496 TPBARs per unit can be harvested and shipped by utilizing two crews for those evolutions to improve these durations from previous estimates.

  The durations to consolidate two units of maximum load TPBARs (i.e., 4,992 TPBARs) into canisters stored in the spent fuel pool (SFP) and shipped off-site are part of ongoing initiatives to manage competition for resources (e.g., refuel floor space, spent fuel pool, auxiliary building crane) associated with other activities. The following initiatives are underway or planned to streamline and reduce durations of work on the refuel floor, current limitations in the SFP, and dry casks that contribute to refuel floor scheduling challenges.

  1. Increase the tritium shipping cask capacity to four TPBAR canisters.
  2. Dry cask change to permit the storage of spent fuel with higher decay heat.
  3. Modification to the tritium consolidation fixture to enhance operation.

4. Replaced the spent fuel pool (SFP) bridge crane to improve operation and reduce maintenance.
5. New Consolidation Building to reduce tritium production process bottlenecks.
6. Replace the vacuum drying skid to increase efficiency.
7. Working multiple/longer shifts to improve turnover and reduce durations of campaigns.
8. Improved high-definition cameras to facilitate more efficient TPBAR inspections.

- The weight of the completed TPBAR consolidation fixture (TCF), is 9,050 pounds (i.e., 3,800 Upper TCF plus 5,250 Lower TCF). The TCF has been successfully utilized to consolidate TPBARs since 2005.

Reference

4.1-1 TVA letter to NRC, 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)

4.2 TPBAR Interface Issue 3: Compliance with DNB Criterion

NUREG-1672, Section 2.4.4, "DOE's analyses regarding the incorporation of the TPBARs in the reference plant showed that the bypass flow will remain within its design limit of 8.4 percent, and that the DNB criterion will continue to be met with no feature of the TPBAR component affecting the coolability of the core. The staff agrees with this assessment. However, the continued compliance with the DNB criterion, given the operating conditions of a particular plant, must be evaluated. The staff has identified this as an interface item that 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 information regarding Interface Issue 3 provided in Reference 4.2-1 applies to this LAR except as provided below.

For the WBN 2,496 TPBAR core, the normal thermal-hydraulic departure from nucleate boiling (DNB) related reload analyses were performed using VIPRE-01 (Reference 4.2-2) and are described in Reference 4.2-1.

Therefore, the presence of TPBARs in the reload core design did not challenge the DNB criterion. An explicit check of the DNB criterion is included in the cycle-specific reload safety evaluation performed for each WBN reload core. Continued performance of this check validates the acceptability of each reload core for operation within the DNB design limits.

<u>References</u>

4.2-1 TVA letter to NRC, 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)

4.2-2 WCAP-15306-NP-A, "VIPRE-01 Modeling Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Westinghouse Electric Company, October 1999 (ML993160096)

4.3 TPBAR Interface Issue 4: Reactor Vessel Integrity

NUREG-1672, Section 2.5.3, "The TPC topical report identifies the applicable regulations and describes methods for demonstrating compliance with Appendices G and H to 10 CFR Part 50 and with 10 CFR 50.61. In the TPC topical report, DOE concludes, and the staff agrees, that the reference plant's pressure/temperature limits report (PTLR) and final safety analysis report (FSAR) would need to be updated to reflect the change to the PTS value and include the updated P-T curves for the applicable EFPYs. In addition, because the reactor vessel integrity analyses are dependent upon the plant-specific materials properties and neutron fluence, the staff concludes 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. The staff has identified this as an interface item that 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 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,” (Enclosure 2). This report presents the evaluation of the WBN Units 1 and 2 reactor pressure vessels with respect to RV integrity. WBN Units 1 and 2 have been approved for 40 years of operation, however; the evaluations in this report are applicable for 32 and 48 effective full-power years (EFPY). Forty years of operation (32 EFPY) is considered end of license and 60 years of operation (48 EFPY) is considered end-of-license-extension. Note that calculations for WBN Unit 2 were only performed up to 32 EFPY. A summary of results for the WBN Units 1 and 2 reactor vessel integrity evaluations is provided. Based on the results, the WBN Units 1 and 2 reactor pressure vessels continue to meet reactor pressure vessel integrity regulatory requirements through the end-of-license and end-of-license extension.

The summary of results provided also includes changes to the surveillance capsule removal schedule for eventual inclusion into the WBN Units 1 and 2 PTLR. The basis for these updates continues to satisfy the requirements of ASTM E-185-82. Specifically, the changes to the schedule are required due to the plant-specific neutron fluence that was addressed as part of this interface item. Changes to the lead factors, EFPY removal times, and expected capsule fluence are seen in the revised schedule to reflect the recently addressed plant-specific neutron fluence as part of this interface issue.

Reactor vessel material surveillance program requirements are described in 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Paragraph III.B.3 of 10 CFR 50, Appendix H states that a proposed material withdrawal schedule must be submitted with a technical justification per 10 CFR 50.4 and approved prior to implementation.

Accordingly, TVA is requesting NRC approval of the proposed revision to the RV surveillance capsule removal schedule for the WBN Units 1 and 2, which is provided in Table 4-1 of the WBN Unit 1 PTLR (Appendix A to TVA System Description Document SDD-N3-68-4001, “Reactor Coolant System”) and Table 4.0-1 of the WBN Unit 2 PTLR (Appendix B to SDD-N3-68-4001).

NRC AL 97-04 clarified the submittal requirements of 10 CFR 50, Appendix H, as follows.

> *"In this instance, as long as the plant's withdrawal schedule change meets the applicable ASTM standard, the plant will not be exceeding the operating authority already granted in its license. Therefore, a license amendment would not be required, although prior NRC approval to verify conformance with the ASTM standard is required by Appendix H."*

ASTM E-185-82 is the applicable standard for WBN in accordance with Section 5.2.4.3 of the WBN dual-unit UFSAR. Because the proposed change described in this enclosure satisfies the requirements of ASTM E-185-82, TVA has determined that a license amendment is not required, which is consistent with the guidance of AL 97-04. WCAP-18769-NP, Revision 1 (Enclosure 2) provides additional information in support of the proposed revision to the reactor vessel surveillance capsule removal schedule for WBN Units 1 and 2. Specifically, additional information regarding the refueling outage in which WBN Unit 2 Capsule W is expected to be withdrawn, and other considerations in planning the EFPY window for WBN Unit 2 Capsule X, is included in Enclosure 2, but is not in the proposed revision to the WBN Unit 2 surveillance capsule removal schedule. This information was originally included in Enclosure 2 as a planning tool to assist the utility with withdrawing the capsule at the expected time, but it has not been included in the proposed schedule to allow planning and approval of capsule withdrawal based on parameters that are directly applicable to ASTM E-185-82.

Tables 4.3-1 and 4.3-2 provide the current WBN Units 1 and 2 surveillance capsule removal schedules, respectively as shown in the WBN PTLR. The WBN Unit 2 surveillance capsule removal schedules were recently revised by References 4.3-1 and 4.3-2. Tables 4.3-3 and 4.3-4 provide the proposed changes to the WBN Units 1 and 2 surveillance capsule removal schedules, respectively. In accordance with WBN Units 1 and 2 TS 5.9.6.c, TVA will submit the revised WBN PTLR to the NRC following NRC approval of the proposed revision to the WBN Units 1 and 2 RV surveillance capsule withdrawal schedule.

<u>References:</u>

4.3-1 TVA letter to NRC, CNL-22-005, “Revision to Watts Bar Nuclear Plant, Unit 2, Reactor Vessel Surveillance Capsule Withdrawal Schedule,” dated January 24, 2022 (ML22024A450)

4.3-2 NRC letter to TVA, “Watts Bar Nuclear Plant, Unit 2 – Revision to the Reactor Vessel Material Surveillance Capsule Withdrawal Schedule (EPID L-2022-LLL-0000),” dated November 14, 2022 (ML22293A408)

Table 4.3-1
Current WBN Unit 1 Surveillance Capsule Removal Schedule

Table 4-1 Surveillance Capsule Removal Schedule				
Capsule	Capsule Location	Lead Factor[(a)]	Withdrawal EFPY[(b)]	Fluence (n/cm^2)[(a)]
U	56°	5.00	1.20	4.47×10^{18} (c)
W	124°	5.05	3.88	1.08×10^{19} (c)
X	236°	5.03	6.63	1.71×10^{19} (c)
Z	304°	5.06	9.37	2.40×10^{19} (c)
V	58.5°	4.31	(d)	6.02×10^{19}
Y	238.5°	4.31	(e)	Standby

(Notes):

(a) Updated from Capsule Z dosimetry analysis.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Plant specific evaluation.

(d) Capsule V will be removed during the last scheduled outage before estimated capsule exposure to a neutron fluence equal to two times the peak RPV neutron fluence at 60 years of operation (i.e. 54 EFPY).

(e) Capsule Y shall remain inserted in the reactor vessel on standby until needed to fulfill future 10 CFR 50, Appendix H or license renewal requirements.

Table 4.3-2
Current WBN Unit 2 Surveillance Capsule Removal Schedule

TABLE 4.0-1
Watts Bar Unit 2 Surveillance Capsule Removal Schedule [(a)]

Capsule	Orientation of Capsule	Lead Factor	Removal Time	Expected Capsule Fluence (n/cm^2,E > 1.0 MeV)
U	Dual 34°	4.70	2.0 EFPY (EOC 2)	0.604×10^{19}
W	Single 34°	4.66	7.0 EFPY	1.94×10^{19} [(b)]
X	Dual 34°	4.69	7.0 EFPY to 13.7 EFPY	1.94×10^{19} to 3.88×10^{19} [(c)]
Z	Single 34°	4.69	Note (d)	Note (d)
V	Dual 31.5°	4.04	Note (d)	Note (d)
Y	Dual 31.5°	4.04	Note (d)	Note (d)

Notes:

(a) This information is taken from the withdrawal schedule contained in WCAP-18518-NP (Ref. 12). EOC = End-of-Cycle

(b) Approximate Fluence at vessel inner wall at End-of-Life (32 EFPY). This capsule should be withdrawn at the outage nearest to but following 7.0 EFPY of operation.

(c) Capsule X should be removed between 11.7 EFPY and 13.7 EFPY if possible. Capsule X must be removed between 7.0 EFPY and 13.7 EFPY in order to satisfy the recommendations of the third capsule end-of-license per ASTM E185-82 (Ref. 7). This removal EFPY should be re-visited at a later date, such as after Capsule W is removed.

(d) Capsules Z, V, and Y should remain in the reactor. If additional metallurgical data is needed, withdrawal and testing of these capsules should be considered. In the event that Capsule W cannot be removed, then Capsule Z may serve as a backup and be removed instead during the same outage.

Table 4.3-3
Proposed Revision to the WBN Unit 1 Surveillance Capsule Removal Schedule

Table 4-1
Surveillance Capsule Removal Schedule

Capsule	Capsule Location	Capsule Lead Factor[(a)]	Removal Time (EFPY)[(b)]	Capsule Fluence (n/cm^2)[(a)]
U	56°	4.87	1.20	4.6×10^{18}
W	124°	4.78	3.88	1.08×10^{19}
X	236°	4.83	6.62	1.75×10^{19}
Z	304°	4.76	9.29	2.40×10^{19}
V	58.5°	4.11[(c)]	24.1[(d)]	5.44×10^{19}
Y	238.5°	4.06[(c)]	[(e)]	Standby

NOTES:

(a) Capsule lead factors and fluence values are taken from Section 2.2.1 of WCAP-18769-NP, Revision 1

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Capsule V lead factor is that projected at the end-of-cycle (EOC) 18, the anticipated withdrawal date. Capsule Y lead factor is calculated at 48 EFPY.

(d) Projected EFPY at the EOC 18, the anticipated withdrawal date of Capsule V. This removal ensures the capsule exposure remains before two times the peak reactor pressure vessel (RPV) neutron fluence (2.73×10^{19}) at 60 years of operation (48 EFPY).

(e) Capsule Y shall remain inserted in the reactor vessel on standby until needed to fulfill future 10 CFR 50, Appendix H or license renewal requirements.

Table 4.3-4
Proposed Revision to the WBN Unit 2 Surveillance Capsule Removal Schedule

TABLE 4.0-1
Watts Bar Unit 2 Surveillance Capsule Removal Schedule

Capsule	Capsule Location	Capsule Lead Factor[(a)]	Removal Time (EFPY)[(b)]	Capsule Fluence (n/cm^2, E > 1.0 MeV)[(a)]
U	Dual 34°	4.80	2.00 EFPY (EOC 2)	0.614×10^{19}
W	Single 34°	4.87	6.8 EFPY[(c)]	1.93×10^{19}
X	Dual 34°	~4.8	6.8 EFPY to 13.8 EFPY[(d)]	1.93×10^{19} to 3.86×10^{19} [(d)]
Z	Single 34°	4.55	Standby[(e)]	Standby[(e)]
V	Dual 31.5°	3.94	Standby[(e)]	Standby[(e)]
Y	Dual 31.5°	3.94	Standby[(e)]	Standby[(e)]

Notes:

(a) Capsule lead factors and fluence values are taken from Section 2.2.2 of WCAP-18769-NP, Revision 1

(b) Effective full power years (EFPY) from plant startup.

(c) Capsule W should be withdrawn at the outage nearest to but following 6.8 EFPY of operation.

(d) Capsule X should be removed between 10.4 EFPY and 13.8 EFPY if possible. Capsule X must be withdrawn between 6.8 EFPY and 13.8 EFPY in order satisfy the recommendations of the third capsule for EOL per ASTM E185-82. The removal EFPY of the third capsule should be revisited at a later date, such as after Capsule W is removed.

(e) Capsules Z, V, and Y should remain in the reactor. If additional metallurgical data is needed, withdrawal and testing of these capsules should be considered. Per ASTM E185-82 and NUREG 1801, Revision 2, it is recommended that the capsules be removed prior to reaching a fluence of two times the peak fluence at EOL. In the event that Capsule W cannot be withdrawn, Capsule Z may be removed instead, during the same outage, to satisfy the ASTM E185-82 requirements for the second withdrawn capsule

4.4 TPBAR Interface Issue 5: Control Room Habitability Systems

NUREG-1672, Section 2.6.1, "Therefore, the staff concludes that, except for the dose criteria issue, the TPC topical report adequately addresses this matter, but that a plant-specific assessment will be needed. The staff has identified this as an interface item that must be addressed by a licensee referencing the TPC topical report in its plant-specific application for authorization to produce tritium for DOE."

The TPBAR Interface Issue 5 information provided for the WBN Unit 1 and WBN Unit 2 Tritium Production Program increase to 2,496 TPBAR is based on the applicable WBN Unit 1 and WBN Unit 2 precedent documents (References 4.4-1 through 4.4-7). For the 2,496 TPBAR increase, the established site-specific design basis analyses for control room (CR) dose have been updated to reflect the TPBAR increase. As noted in Section 4.1-5 to Enclosure 1 of Reference 4.4-7, the current accident source term (i.e., fission product inventory) for the tritium production core (TPC) for WBN Units 1 and 2 was calculated using ORIGEN2.1. The core radionuclide activity inventory for the TPC with 2,496 TPBARs was calculated using ORIGEN-ARP/ORIGEN-S in SCALE 6.0 (Reference 4.4-10). Attachment 6 to this enclosure contains a revision to the WBN dual-unit UFSAR that TVA has determined requires prior NRC approval. Specifically, the proposed UFSAR change modifies the source term for design basis accident analyses to allow the core fission product inventory to be calculated using an ORIGEN-ARP/ORIGEN-S in SCALE 6.0. The precedent for using this updated computer code is described in Section 5.2 to this enclosure.

<u>Impacts on Station Accident Analysis</u>

The American Nuclear Society (ANS) classification of nuclear plant conditions divides plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows.

Condition I: Normal Operation and Operational Transients

Condition II: Faults of Moderate Frequency

Condition III: Infrequent Faults

Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur.

TPBARs were designed to withstand the rigors associated with Conditions I through IV events. Implementation of the full spectrum loss of coolant accident (FSLOCA™[1]) methodology and the new TPBARs stress analysis methodology discussed in the large

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break loss of coolant accident (LBLOCA) analyses confirm that TPBARs do not rupture during a LBLOCA. However, the LBLOCA dose analyses still model TPBAR failures as a bounding case.

In addition, TPBAR failure tritium coolant concentrations are conservatively included in the remainder of the design-basis accidents associated with TPBARs except the fuel handling accident (FHA). The source terms associated with the LBLOCA and FHA are based on the core inventory. 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 concentrations. The effect of increasing to 2,496 TPBARs on each of these source terms is discussed below.

The current licensing basis regulatory limits for WBN are established in terms of whole body dose, beta dose, and thyroid dose, except for the FHA, which is in terms of total effective dose equivalent (TEDE). Tritium does not affect the whole body or thyroid doses. The decay emission energy of tritium is insufficient to penetrate the skin and contribute to the whole-body dose, and the thyroid dose is explicitly limited to inhalation of radioiodine. To demonstrate the effect on radiological consequences of the increased tritium in the TPC, TVA included calculated TEDE for the FHA, and, for informational purposes, the remaining accidents.

<u>Change to Method of Evaluation Described in the Updated Final Safety Analysis Report</u>

The core radionuclide activity inventory for the tritium production core (TPC) with 2,496 TPBARs was calculated using ORIGEN-ARP/ORIGEN-S in SCALE 6.0. The current TPC inventory, which was originally developed to support WBN Unit 1 License Amendment No. 40 for a TPC with 2,304 TPBARs, was calculated using ORIGEN2.1. WBN dual-unit UFSAR Section 15.1.7.1, “Radioactivity in the Core,” identifies ORIGEN as being used to calculate the fission product inventory and provides references for the computer code. The change in computer codes, from ORIGEN2.1 to ORIGEN-ARP/ORIGEN-S, is a change to the calculational framework used to determine the accident source term.

<u>Core Inventory</u>

The 2,496 TPBAR equilibrium fuel cycle is based on 108-feed assemblies, an increase from the current 96-feed assembly core. The source term analysis included a tritium inventory of 1.2 grams of tritium/TPBAR/cycle, which results in a total of 2.90E+07 Ci of tritium in the core. The activity inventories for the other radionuclides were calculated using ORIGEN-ARP/ORIGEN-S in the SCALE 6.0 computer code. The activity inventories were based on 108 once burned fuel assemblies and 85 twice burned fuel assemblies. The average assembly inventories 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 added together to determine the total activity inventory for each radionuclide.

Table 4.4-1 provides information used in the determination of the TPC inventory. Table 4.4-2 provides the core inventory.

Table 4.4-1: Parameters Used to Determine the TPC Inventory	
Parameter	**TPC Value**
Power (MWt)	3480
Cycle Energy objective [Effective Full Power Days (EFPD)]	510/cycle
Enrichment (initial weight percent)	4.95

Table 4.4-2: Core Inventory for 2,496 TPC			
Nuclide	**Total Core Inventory (Ci)**	**Average Assembly Inventories (Ci)**	
		Once-burned fuel [End of Cycle (EOC)]	**Twice-burned fuel (EOC)**
Kr-83m	1.21E+07	6.64E+04	5.82E+04
Kr-85m	2.62E+07	1.46E+05	1.23E+05
Kr-85	9.58E+05	4.01E+03	6.18E+03
Kr-87	5.21E+07	2.92E+05	2.42E+05
Kr-88	6.99E+07	3.93E+05	3.23E+05
Kr-89	8.75E+07	4.94E+05	4.02E+05
Kr-90	9.16E+07	5.21E+05	4.17E+05
Xe-131m	1.20E+06	5.91E+03	6.63E+03
Xe-133m	5.48E+06	2.84E+04	2.85E+04
Xe-133	1.84E+08	9.60E+05	9.50E+05
Xe-135m	3.97E+07	2.02E+05	2.10E+05
Xe-135	5.95E+07	3.28E+05	2.83E+05
Xe-137	1.75E+08	9.13E+05	9.00E+05
Xe-138	1.67E+08	8.83E+05	8.48E+05
Xe-139	1.25E+08	6.67E+05	6.24E+05
Xe-140	8.60E+07	4.65E+05	4.22E+05
I-130	1.28E+06	4.30E+03	9.54E+03
I-131	9.35E+07	4.80E+05	4.91E+05
I-132	1.37E+08	7.04E+05	7.14E+05
I-133	1.94E+08	1.01E+06	1.00E+06
I-134	2.19E+08	1.15E+06	1.12E+06
I-135	1.85E+08	9.59E+05	9.55E+05
I-136m	4.14E+07	2.10E+05	2.20E+05
Br-83	1.20E+07	6.60E+04	5.76E+04
Br-84m	7.64E+05	3.77E+03	4.20E+03
Br-84	2.22E+07	1.23E+05	1.04E+05
Br-85	2.61E+07	1.45E+05	1.22E+05
Br-87	4.11E+07	2.31E+05	1.91E+05
Cs-134	1.14E+07	3.39E+04	9.08E+04
Cs-135	4.50E+01	1.79E-01	3.02E-01
Cs-136	4.56E+06	1.74E+04	3.15E+04
Cs-137	9.41E+06	3.69E+04	6.38E+04

Table 4.4-2: Core Inventory for 2,496 TPC			
Nuclide	Total Core Inventory (Ci)	Average Assembly Inventories (Ci)	
		Once-burned fuel [End of Cycle (EOC)]	Twice-burned fuel (EOC)
Cs-138	1.82E+08	9.58E+05	9.27E+05
Cs-139	1.70E+08	8.97E+05	8.65E+05
Cs-140	1.45E+08	7.72E+05	7.28E+05
Cs-141	1.13E+08	5.94E+05	5.74E+05
Rb-88	7.10E+07	3.98E+05	3.29E+05
Rb-89	9.33E+07	5.25E+05	4.32E+05
Rb-90m	2.80E+07	1.54E+05	1.34E+05
Rb-90	8.49E+07	4.81E+05	3.88E+05
Rb-91	1.14E+08	6.37E+05	5.32E+05
Se-84	2.16E+07	1.21E+05	1.01E+05
Sr-89	9.69E+07	5.45E+05	4.48E+05
Sr-90	7.40E+06	3.07E+04	4.81E+04
Sr-91	1.22E+08	6.77E+05	5.72E+05
Sr-92	1.29E+08	7.13E+05	6.16E+05
Sr-93	1.43E+08	7.78E+05	6.90E+05
Sr-94	1.40E+08	7.60E+05	6.79E+05
Y-90	7.63E+06	3.14E+04	4.98E+04
Y-91m	7.06E+07	3.93E+05	3.32E+05
Y-91	1.26E+08	6.98E+05	5.90E+05
Y-92	1.31E+08	7.21E+05	6.23E+05
Y-93	1.46E+08	7.96E+05	7.07E+05
Y-94	1.53E+08	8.26E+05	7.46E+05
Y-95	1.58E+08	8.47E+05	7.80E+05
Y-96	9.54E+07	5.15E+05	4.68E+05
Zr-95	1.64E+08	8.79E+05	8.10E+05
Zr-97	1.63E+08	8.61E+05	8.27E+05
Nb-95	1.65E+08	8.85E+05	8.16E+05
Nb-97m	1.55E+08	8.17E+05	7.85E+05
Nb-97	1.64E+08	8.66E+05	8.33E+05
Mo-99	1.77E+08	9.22E+05	9.11E+05
Tc-99m	1.57E+08	8.18E+05	8.13E+05
Tc-99	1.24E+03	4.99E+00	8.26E+00
Tc-101	1.61E+08	8.31E+05	8.44E+05
Ru-103	1.39E+08	6.76E+05	7.75E+05
Ru-105	9.20E+07	4.22E+05	5.47E+05
Ru-106	3.98E+07	1.51E+05	2.76E+05
Ru-107	4.77E+07	2.06E+05	3.00E+05
Rh-103m	1.39E+08	6.75E+05	7.74E+05

Table 4.4-2: Core Inventory for 2,496 TPC			
Nuclide	Total Core Inventory (Ci)	Average Assembly Inventories (Ci)	
		Once-burned fuel [End of Cycle (EOC)]	Twice-burned fuel (EOC)
Rh-105m	2.61E+07	1.20E+05	1.55E+05
Rh-105	8.38E+07	3.87E+05	4.94E+05
Rh-106	4.41E+07	1.69E+05	3.03E+05
Rh-107	4.85E+07	2.10E+05	3.04E+05
Sn-130	1.66E+07	8.73E+04	8.43E+04
Sb-127	8.03E+06	3.91E+04	4.48E+04
Sb-129	2.54E+07	1.25E+05	1.40E+05
Sb-130m	2.86E+07	1.49E+05	1.48E+05
Sb-130	2.59E+07	1.33E+05	1.36E+05
Sb-133	6.03E+07	3.21E+05	3.01E+05
Te-125m	1.40E+05	5.25E+02	9.83E+02
Te-127m	1.31E+06	6.27E+03	7.50E+03
Te-127	7.98E+06	3.87E+04	4.47E+04
Te-129m	4.51E+06	2.22E+04	2.49E+04
Te-129	2.37E+07	1.17E+05	1.31E+05
Te-131m	1.75E+07	8.70E+04	9.49E+04
Te-131	7.97E+07	4.11E+05	4.15E+05
Te-132	1.34E+08	6.89E+05	6.95E+05
Te-133m	9.22E+07	4.85E+05	4.69E+05
Te-133	1.04E+08	5.47E+05	5.34E+05
Te-134	1.77E+08	9.40E+05	8.85E+05
Ba-137m	8.95E+06	3.51E+04	6.07E+04
Ba-139	1.74E+08	9.14E+05	8.84E+05
Ba-140	1.68E+08	8.85E+05	8.52E+05
Ba-141	1.57E+08	8.28E+05	7.94E+05
Ba-142	1.49E+08	7.89E+05	7.46E+05
La-140	1.72E+08	8.99E+05	8.78E+05
La-141	1.58E+08	8.33E+05	8.00E+05
La-142	1.53E+08	8.11E+05	7.70E+05
La-143	1.49E+08	7.93E+05	7.40E+05
Ce-141	1.58E+08	8.35E+05	8.02E+05
Ce-143	1.50E+08	7.98E+05	7.46E+05
Ce-144	1.20E+08	5.95E+05	6.61E+05
Ce-145	1.01E+08	5.39E+05	5.09E+05
Pr-143	1.49E+08	7.98E+05	7.45E+05
Pr-144	1.21E+08	5.99E+05	6.66E+05
Pr-145	1.02E+08	5.39E+05	5.09E+05
Np-239	1.69E+09	8.19E+06	9.50E+06

Table 4.4-2: Core Inventory for 2,496 TPC			
Nuclide	Total Core Inventory (Ci)	Average Assembly Inventories (Ci)	
		Once-burned fuel [End of Cycle (EOC)]	Twice-burned fuel (EOC)
H-3	2.90E+07	-	-

Primary and Secondary Coolant Concentrations

The current licensing basis primary and secondary coolant concentrations are based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-18.1-1984. Table 4.4-3 provides the parameters used and Table 4.4-4 provides the resulting radionuclide concentrations. Table 4.4-3 is the same as Table 4.1-4 of Reference 4.4-7.

The concentration of tritium for a TPC was calculated using the same methodology as used for the existing TPC, except with 2,496 TPBARs instead of 1,792 TPBARs, and 80 curie (Ci)/year for integral fuel burnable absorber (IFBA) releases, up from 40 Ci/year. The non-TPC tritium source of 870 Ci/year is unchanged. The average tritium concentration without any TPBAR failures was determined by multiplying the average non-TPC tritium concentration by the ratio of the total annual tritium expected for a TPC by that expected for a non-TPC. ANSI/ANS-18.1-1984 states that the average tritium concentration should be assumed to be 1.0 micro(μ)Ci/gram(gm) in the primary coolant. 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/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.

Table 4.4-3: Parameters used for the Primary and Secondary Coolant Concentrations	
Parameter	**Value**
Thermal Power (MWt)	3480
Steam Flow Rate (lb/hr)	1.5E+07
Weight of water in RCS (lb)	4.71E+05
Weight of water in all steam generators (SGs) (lb)	3.80E+05
Reactor coolant letdown flow rate (purification) (lb/hr)	3.7E+04
Reactor coolant letdown flow rate (yearly average for boron control) (lb/hr)	845
SG Blowdown flow total (lb/hr)	3.00E+04
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	1.0
Flow through the purification system cation demineralizer (lb/hr)	3.7E+03
Ratio of condensate demineralizer flow rate to the total steam flow rate	0.0
Fraction of the noble gas activity in the letdown stream which is not returned to the RCS (not including the boron recovery system)	0.0

Table 4.4-4: Primary and Secondary Coolant Concentrations			
Nuclide Class	**Reactor Coolant WBN µCi/gm**	**Secondary Water WBN µCi/gm**	**Secondary Steam WBN µCi/gm**
Class 1			
Kr-85m	1.90E-01	0.00E+00	4.04E-08
Kr-85	2.59E-01	0.00E+00	5.36E-08
Kr-87	1.79E-01	0.00E+00	3.58E-08
Kr-88	3.33E-01	0.00E+00	7.03E-08
Xe-131m	6.89E-01	0.00E+00	1.42E-07
Xe-133m	7.87E-02	0.00E+00	1.69E-08
Xe-133	2.73E+00	0.00E+00	5.66E-07
Xe-135m	1.55E-01	0.00E+00	3.23E-08
Xe-135	1.00E+00	0.00E+00	2.13E-07
Xe-137	4.06E-02	0.00E+00	8.49E-09
Xe-138	1.43E-01	0.00E+00	2.99E-08
Class 2			
Br-84	1.90E-02	1.12E-07	1.12E-09
I-131	4.67E-02	4.65E-06	4.65E-08
I-132	2.44E-01	5.28E-06	5.28E-08
I-133	1.51E-01	1.11E-05	1.11E-07
I-134	4.01E-01	3.71E-06	3.71E-08
I-135	2.92E-01	1.31E-05	1.31E-07
Class 3			
Rb-88	2.26E-01	7.70E-07	3.78E-09
Cs-134	7.18E-03	7.51E-07	3.87E-09
Cs-136	8.88E-04	9.07E-08	4.53E-10
Cs-137	9.51E-03	1.00E-06	5.01E-09
Class 4			
N-16	4.00E+01	1.18E-06	1.18E-07
Class 5			
H-3	1.00E+00	1.00E-03	1.00E-03
TPC 2,496	1.55E+01	1.55E-02	1.55E-02
Two TPBAR failure	1.24E+02	1.24E-01	1.24E-01
Class 6			
Na-24	5.11E-02	3.34E-06	1.67E-08
Cr-51	3.18E-03	3.37E-07	1.63E-09
Mn-54	1.64E-03	1.69E-07	8.60E-10
Fe-55	1.23E-03	1.28E-07	6.52E-10
Fe-59	3.07E-04	3.12E-08	1.59E-10
Co-58	4.71E-03	4.94E-07	2.45E-09
Co-60	5.42E-04	5.73E-08	2.87E-10

Table 4.4-4: Primary and Secondary Coolant Concentrations			
Nuclide Class	**Reactor Coolant WBN µCi/gm**	**Secondary Water WBN µCi/gm**	**Secondary Steam WBN µCi/gm**
Zn-65	5.22E-04	5.47E-08	2.61E-10
Sr-89	1.43E-04	1.48E-08	7.54E-11
Sr-90	1.23E-05	1.28E-09	6.52E-12
Sr-91	1.06E-03	5.87E-08	2.93E-10
Y-90	1.23E-05	1.28E-09	6.52E-12
Y-91m	5.43E-04	4.92E-09	2.46E-11
Y-91	5.32E-06	5.46E-10	2.86E-12
Y-93	4.63E-03	2.54E-07	1.29E-09
Zr-95	3.99E-04	4.16E-08	2.06E-10
Nb-95	2.87E-04	2.86E-08	1.48E-10
Mo-99	6.68E-03	6.23E-07	2.99E-09
Tc-99m	5.29E-03	2.15E-07	1.12E-09
Ru-103	7.68E-03	8.05E-07	4.16E-09
Ru-106	9.20E-02	9.64E-06	4.69E-08
Rh-103m	7.68E-03	8.05E-07	4.16E-09
Rh-106	9.20E-02	9.64E-06	4.69E-08
Ag-110m	1.33E-03	1.38E-07	7.03E-10
Te-129m	1.95E-04	2.03E-08	1.01E-10
Te-129	2.82E-02	3.46E-07	1.73E-09
Te-131m	1.59E-03	1.29E-07	6.43E-10
Te-131	9.14E-03	4.29E-08	2.22E-10
Te-132	1.77E-03	1.66E-07	8.28E-10
Ba-137m	9.51E-03	1.00E-06	5.01E-09
Ba-140	1.34E-02	1.34E-06	6.71E-09
La-140	2.63E-02	2.26E-06	1.12E-08
Ce-141	1.54E-04	1.58E-08	8.05E-11
Ce-143	2.97E-03	2.40E-07	1.22E-09
Ce-144	4.09E-03	4.17E-07	2.14E-09
Pr-143	2.97E-03	2.40E-07	1.22E-09
Pr-144	4.09E-03	4.17E-07	2.14E-09
W-187	2.67E-03	2.03E-07	1.03E-09
Np-239	2.30E-03	2.08E-07	1.04E-09

Large Break Loss of Cooling Accident

The WBN LBLOCA analyses of record were revised to account for the increase in TPBARs by utilizing the core inventory calculated for the 2,496 TPBAR TPC as described above. The parameters remain the same as submitted by TVA in Reference 4.4-11 and approved by the NRC in Reference 4.4-12. Resulting dose consequences are provided in Table 4.4-5.

Two cases are considered. One case analyzes a single failure such that one whole train of the emergency gas treatment system (EGTS) fails from the beginning of the accident. The second case analyzes a single failure in the controls of the EGTS such that one set of EGTS dampers is assumed to be in the full exhaust position [pressure control operator (PCO) failure case].

The WBN LBLOCA radiological dose consequences for the CR and offsite [two-hour exclusion area boundary (EAB), and 30-day low population zone (LPZ) analysis] results shown in Table 4.4-5 are below the 10 CFR Part 100, 10 CFR Part 50 Appendix A General Design Criterion (GDC) 19 regulatory criteria.

Table 4.4-5: WBN Dose Consequences from a LBLOCA					
Unit 1/2 Single Train EGTS Case					
Dose (rem)	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	8.75E-01	5	2.05E+00	1.87E+00	25
Beta	7.54E+00	30	1.16E+00	2.27E+00	300
Inhalation (ICRP-30)	3.73E+00	30	4.01E+01	1.42E+01	300
TEDE	2.33E+00	5	3.66E+00	2.33E+00	25
Unit 1 PCO Control Failure Case					
Dose (rem)	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	1.08E+00	5	2.47E+00	2.35E+00	25
Beta	9.44E+00	30	1.45E+00	2.68E+00	300
Inhalation (ICRP-30)	3.19E+00	30	3.13E+01	1.23E+01	300
TEDE	2.65E+00	5	3.48E+00	2.57E+00	25
Unit 2 PCO Control Failure Case					
Dose (rem)	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	1.05E+00	5	2.40E+00	2.28E+00	25
Beta	9.17E+00	30	1.40E+00	2.63E+00	300
Inhalation (ICRP-30)	3.17E+00	30	3.13E+01	1.22E+01	300
TEDE	2.60E+00	5	3.42E+00	2.51E+00	25

Fuel Handling Accident

The WBN FHA analysis of record was revised to account for the new assembly inventory due to increasing the maximum number of TPBARs from 1,792 to 2,496 in the TPC.

Three FHA cases are evaluated. The first case considered is an FHA in the SFP area located in the auxiliary building. This case was evaluated using the alternate source term (AST) assumptions from RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000. In this case, no credit is taken in the analysis for the auxiliary building gas treatment system (ABGTS).

The second case is an open containment case for an FHA inside containment where there is open communication between the containment and the auxiliary building. This evaluation also uses the AST assumptions from RG 1.183 with no credit for any filtration systems.

The third case is a TPBAR only accident in the SFP, which has been previously evaluated, and is not impacted by the increase to 2,496 TPBARs TPC because each fuel assembly has the same number (24) of TPBARs and only tritium release is considered. Because tritium is low energy beta decay only, the spent fuel pit monitors, and the CR intake monitors do not respond to the tritium; therefore, the auxiliary building exhaust will not be isolated, resulting in the releases being discharged out the Auxiliary Building vent.

The WBN FHA radiological dose consequences analysis results shown in Table 4.4-6 are below the 10 CFR 50.67 and RG 1.183 limits.

Table 4.4-6: WBN Dose Consequences from an FHA					
TPBAR Only FHA	**Dose (rem)**				
Parameter	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
TEDE	1.16E+00	5	2.88E-01	8.06E-02	6.25
Auxiliary Building FHA (RG 1.183)	**Dose (rem)**				
Parameter	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
TEDE	2.26E+00	5	2.65E+00	7.42E-01	6.25
Containment FHA (RG 1.183)	**Dose (rem)**				
Parameter	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
TEDE	2.24E+00	5	2.65E+00	7.42E-01	6.25

Main Steam Line Break

The WBN MSLB analysis of record was revised to account for increasing the maximum number of TPBARs from 1,792 to 2,496 in the TPC and includes the WBN Units 1 and 2 replacement steam generators (RSGs). The parameters remain the same as described in References 4.4-7 and 4.4-13. The WBN calculated radiological consequences for the MSLB with a 2,496 TPBAR core, as shown in Table 4.4-7, are below the 10 CFR Part 100, 10 CFR Part 50 Appendix A GDC 19 dose limits, and NUREG-0800 guidance.

Table 4.4-7: WBN Dose Consequences from a MSLB Accident

Parameter	Dose (rem)				
Pre-Accident Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	5.78E-03	5	2.57E-02	1.07E-02	25
Beta	5.59E-02	30	8.35E-03	4.09E-03	300
Thyroid (ICRP-30)	1.27E+01	30	2.56E+00	1.25E+00	300
TEDE	4.44E-01	5	1.80E-01	8.44E-02	25
Accident Initiated Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	1.18E-02	5	1.10E-01	1.36E-01	2.5
Beta	9.66E-02	30	2.66E-02	3.25E-02	30
Thyroid (ICRP-30)	1.73E+01	30	3.32E+00	4.98E+00	30
TEDE	6.33E-01	5	3.62E-01	5.13E-01	2.5

Steam Generator Tube Rupture

The WBN SGTR analyses of record was revised to account for increasing the maximum number of TPBARs from 1,792 to 2,496 in the TPC. The parameters remain the same as described in References 4.4-13 and 4.4-14 except for the mass releases. These were updated in accordance with 10 CFR 50.59 as part of implementing PAD5. The WBN Unit 2 RSGs are similar to the Unit 1 RSGs; however, the mass releases are different and therefore the SGTR dose results are slightly different. The revised mass releases are provided in Table 4.4-8.

Table 4.4-8 SGTR Mass Releases

Parameter	U1	U2
Primary Coolant Mass Release (lbm)		
Total	163,200	130,900
Flashed	9146.8	9588.5
Secondary side release from ruptured SG (lbm)		
0-2 hr	109,300	109,800
2-8 hr	34,800	32,900
Secondary side release from intact SG (lbm)		
0-2 hr	544,100	575,800
2-8 hr	924,300	968,700

The WBN calculated radiological consequences for the SGTR with a 2,496 TPBAR core, as shown in Table 4.4-9 and Table 4.4-10, are below the 10 CFR Part 100, 10 CFR Part 50 Appendix A GDC 19 dose limits, and NUREG-0800 guidance.

Table 4.4-9: WBN Unit 1 Dose Consequences from an SGTR Accident					
Parameter	**Dose (rem)**				
Pre-Accident Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	8.77E-02	5	3.69E-01	1.08E-01	25
Beta	9.66E-01	30	2.09E-01	6.39E-02	300
Thyroid (ICRP-30)	2.29E+01	30	1.32E+01	3.77E+00	300
TEDE	1.29E+00	5	1.25E+00	3.60E-01	25
Accident Initiated Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	8.32E-02	5	5.51E-01	1.61E-01	2.5
Beta	9.53E-01	30	2.46E-01	7.52E-02	30
Thyroid (ICRP-30)	3.93E+00	30	6.95E+00	2.04E+00	30
TEDE	6.66E-01	5	1.18E+00	3.43E-01	2.5

Table 4.4-10: WBN Unit 2 Dose Consequences from an SGTR Accident					
Parameter	**Dose (rem)**				
Pre-Accident Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	5.19E-02	5	3.58E-01	1.04E-01	25
Beta	5.65E-01	30	1.93E-01	5.78E-02	300
Thyroid (ICRP-30)	1.52E+01	30	1.36E+01	3.88E+00	300
TEDE	8.14E-01	5	1.27E+00	3.63E-01	25
Accident Initiated Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	4.85E-02	5	5.36E-01	1.55E-01	2.5
Beta	5.50E-01	30	2.28E-01	6.84E-02	30
Thyroid (ICRP-30)	2.59E+00	30	6.99E+00	2.03E+00	30
TEDE	3.97E-01	5	1.17E+00	3.37E-01	2.5

Loss of Offsite Power (LOOP)

The WBN loss of offsite power (LOOP) analysis of record was revised to account for increasing the maximum number of TPBARs from 1,792 to 2,496 in the TPC. The parameters remain the same as discussed in Reference 4.4-7, whose analysis bounds both units.

There is no Standard Review Plan (SRP) or RG for this accident. This is a simple best estimate analysis. The TS limiting case is calculated utilizing a factor of 13,880 as a multiplier to the realistic case, except for tritium. This is the scaling factor determined to scale the realistic secondary coolant inventory to the TS 3.7.14 limit of 0.10 μCi/gm of Iodine-131 dose equivalent.

The calculated radiological consequences for the LOOP with an increase from a 1,792 TPBAR core to a 2,496 TPBAR core shown in Table 4.4-11 are below the 10 CFR Part 100 and 10 CFR Part 50 Appendix A GDC 19 dose limits.

Table 4.4-11: WBN Dose Consequences from a LOOP					
Parameter	**Dose (rem)**				
Realistic Case	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	9.00E-09	5	2.70E-08	1.54E-08	2.5
Beta	3.58E-04	30	2.14E-05	1.22E-05	30
Thyroid (ICRP-30)	2.58E-06	30	3.42E-06	1.96E-06	30
TEDE	5.84E-03	5	3.50E-04	2.00E-04	2.5
TS Limiting Case	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	1.25E-04	5	3.74E-04	2.14E-04	2.5
Beta	1.73E-03	30	2.12E-04	1.21E-04	30
Thyroid (ICRP-30)	3.58E-02	30	4.74E-02	2.71E-02	30
TEDE	7.26E-03	5	3.27E-03	1.87E-03	2.5

Waste Gas Decay Tank Rupture

The WGDT dose consequence analysis of record was previously revised to account for a TPC by utilizing a tritium source term based on 2,500 TPBARs with a permeation rate of 10 Ci/TPBAR/year and two TPBAR failures as discussed in Reference 4.4-7. The WGDT is a common system for both WBN units; therefore, the analysis in Reference 4.4-7 applies to both units. This existing analysis bounds updating the TPC to 2,496 TPBARs with a 5 Ci/TPBAR/year permeation rate and two TPBAR failures.

The calculated radiological consequences for the WGDT rupture shown in Table 4.4-12, remain substantially below 10 CFR Part 100, 10 CFR Part 50 Appendix A GDC 19 dose limits, and NUREG-0800 guidance.

Table 4.4-12: WBN Dose Consequences from a WGDT Rupture

Parameter	**Dose (rem)**				
RG 1.24 Analysis	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	9.44E-01	5	5.96E-01	1.67E-01	2.5
Beta	8.17E+00	30	1.62E+00	4.52E-01	30
Thyroid (ICRP-30)	1.08E-02	30	1.29E-02	3.60E-03	30
TEDE	1.25E+00	5	3.52E-01	9.84E-02	2.5
Realistic Analysis	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Gamma	3.76E-02	5	2.64E-02	7.38E-03	2.5
Beta	4.48E-01	30	8.86E-02	2.48E-02	30
Thyroid (ICRP-30)	9.78E-03	30	1.18E-02	3.29E-03	30
TEDE	2.61E-01	5	5.57E-02	1.56E-02	2.5

Failure of Small Lines Carrying Primary Coolant Outside Containment

The WBN 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 stated, in NUREG-0847 (Reference 4.4-8) and subsequent Supplement 25 (Reference 4.4-9), that the WBN UFSAR did not contain this analysis and that the NRC performed their own confirmatory analysis and found this to be acceptable.

Rod Ejection Accident

As discussed in the NUREG-0847, Supplement 25, the source term for a rod ejection accident is considerably less than for a LBLOCA. Because the dose consequence results for the WBN LBLOCA are less than the SRP acceptance criteria for a rod ejection accident (25 percent of the values in 10 CFR 100), the rod ejection accident is not explicitly analyzed.

References

4.4-1 NRC letter to TVA, “Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884),” dated September 23, 2002 (ML022540925)

4.4-2 TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015,” dated August 20, 2001 (ML012390106 and ML012390115)

4.4-3 TVA letter to NRC, “Watts Bar Nuclear Plant - Responses to Request for Additional Information (RAI) Regarding Tritium Production - Interface Item Numbers 1, 6, 7, 10, 11, and 12 (TAC No. MB1884),” dated October 29, 2001 (ML020320146)

4.4-4 TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision to the Spent Fuel Pool Cooling Analysis Methodology,” dated April 20, 2001 (ML011170181)

4.4-5 TVA letter to NRC, “Watts Bar Nuclear Plant – Responses to RAI Regarding Spent Fuel Pool Cooling Analysis Methodology (TAC No. MB1884),” dated November 14, 2001 (ML020150176)

4.4-6 NRC letter to TVA, “Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Spent Fuel Pool Cooling Analysis Methodology Change (TAC Nos. MB1807 and MB1884),” dated February 21, 2002 (ML020580612)

4.4-7 TVA letter to NRC, 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)

4.4-8 NUREG-0847, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2,” dated June 1982 (ML072060490)

4.4-9 NUREG-0847, Supplement 25, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2,” dated December 2011 (ML12011A024)

4.4-10 SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, ORNL/TM-2005/39, Version 6, Vols. I–III, January 2009. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-750.

4.4-11 TVA letter to NRC, CNL-19-077, “Application to Modify Watts Bar Nuclear Plant (WBN) Unit 1 and Unit 2 Technical Specifications 3.6.15, ‘Shield Building,’ (WBN-TS-19-10),” dated December 6, 2019 (ML19340B773)

4.4-12 NRC letter to TVA, “Watts Bar Nuclear Plant, Units 1 and 2 - Issuance of Amendment Nos. 139 and 45 Regarding Revisions to Technical Specification 3.6.1.5, ‘Shield Building’ (EPID L-2019-LLA-0272),” dated December 8, 2020 (ML20245E413)

4.4-13 TVA letter to NRC, CNL-15-001, "Application to Revise Technical Specification 4.2.1, ‘Fuel Assemblies,’ (WBN-TS-15-03),” dated March 31, 2015 (ML15098A446)

4.4-14 TVA letter to NRC, CNL-21-019, “Watts Bar Nuclear Plant Unit 2 – License Amendment Request to Revise Updated Final Safety Analysis Report Section 15.5.5 – Steam Generator Tube Rupture Dose Analysis (WBN-TS-20-04),” dated March 2, 2021 (ML21061A346)

4.4-15 NRC letter to 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 (EPID L-2020-LLA-0005),” dated February 26, 2021 (ML21034A166 and ML21034A169)

4.5 TPBAR Interface Issue 7: Light Load Handling System

NUREG-1672, Section 2.9.1, "'DOE evaluated the affect (sic) of TPBARs on the light load handling system for the reference plant against the guidance of SRP Section 9.1.4. DOE states, and the staff agrees, that the incorporation of the TPBARs has no effect on this system. However, DOE concludes, and the staff agrees, that because of the increase in weight of TPBARs compared to burnable poison rod assemblies, this effect should be evaluated on a plant-specific basis. The staff has identified this as an interface item that 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 information regarding Interface Issue 7 provided in Reference 4.5-1 is applicable to this LAR except as provided below.

The spent fuel pit crane has been replaced and has a reduced capacity of 2,500 pounds as opposed to the original crane’s capacity of 4,000 pounds. This is acceptable because the capacity is above the maximum load to be handled by the crane, and the information regarding Interface Issue 7 provided in Reference 4.5-1 remains valid with the decreased capacity.

Reference:

4.5-1 TVA letter to NRC, 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)

4.6 TPBAR Interface Issue 8: Station Service Water System

NUREG-1672, Section 2.9.1, "The staff has reviewed the information presented by DOE and concludes that the effect on the SSWS is not safety significant, because the additional heat load introduced by TPBARs is very low and is indirectly transferred to the SSWS. The staff also agrees that, during the generic review of the TPC topical report, a quantitative analysis of the effect of the TPBARs on the SSWS was not appropriate. However, DOE concludes, and the staff agrees, that a quantitative analysis for the SSWS needs to be addressed by licensees participating in DOE program for the CLWR production of tritium. The staff has identified this as an interface item that 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 information regarding Interface Issue 8 provided in Reference 4.6-1 is applicable to this LAR except as provided below.

The TPBAR Interface Issue 8 information provided for the WBN Tritium Production Program LAR is based on the applicable WBN precedent documents (References 4.6-1, 4.6-2, 4.6-3, 4.6-4, 4.6-5, 4.6-6, 4.6-7, and 4.6-8).

The design basis function of the station service water system, which is the essential raw cooling water (ERCW) system for WBN, includes providing a cooling loop for heat removal from the component cooling system (CCS). The ERCW supplies water from the ultimate heat sink (UHS) (Tennessee River) to cool primarily safety related components. The CCS is the primary means for cooling the plant and removing residual decay heat during late stages of plant cooldown and during outages. The CCS intermediate cooling loop provides a heat sink to the SFP cooling and cleanup system (SFPCCS) and residual heat removal (RHR) System.

Tritium Impact on Decay Heat to ERCW

Utilizing the methodology established on previous TPBAR projects at WBN, TVA has updated the quantitative analysis of expected impact of decay heat on the SFPCCS. The updates to these SFPCCS analyses have shown that the heat loads from these systems to CCS are bounded by what was analyzed for the previous increase to 1,792 TPBARs. The existing RHR System analysis was found to be bounding of the 2,496 TPBAR change such that there was no impact, beyond what was previously analyzed, on the CCS or the ERCW System. Interface Issue 12 (Section 4.9) provides a more detailed discussion of these changes.

As the CCS heat loads associated with the increase to 2,496 TPBARs are bounded by what was previously analyzed, there is no impact to the ERCW System with the increase to 2,496 TPBARs.

ERCW Summary

The ERCW system has adequate capacity and cooling margin to perform its safety and non-safety functions with the changes in decay heat loads imposed by the increase to 2,496 TPBARs. TVA procedures and cycle-specific analysis demonstrate that SFP heat loads are maintained within allowable limits [see Interface Issue 11 (Section 4.8) for further discussion]. Tritium production activities do not have an adverse impact on the ERCW heat removal capabilities.

References:

4.6-1 TVA letter to NRC, 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)

4.6-2 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.6-3 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015," dated August 20, 2001 (ML012390106 and ML012390115)

4.6-4 TVA letter to NRC, "Watts Bar Nuclear Plant - Responses to Request for Additional Information (RAI) Regarding Tritium Production - Interface Item Numbers 1, 6, 7, 10, 11, and 12 (TAC No. MB1884)," dated October 29, 2001 (ML020320146)

4.6-5 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision to the Spent Fuel Pool Cooling Analysis Methodology," dated April 20, 2001 (ML011170181)

4.6-6 TVA letter to NRC, "Watts Bar Nuclear Plant – Responses to RAI Regarding Spent Fuel Pool Cooling Analysis Methodology (TAC No. MB1884)," dated November 14, 2001 (ML020150176)

4.6-7 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Spent Fuel Pool Cooling Analysis Methodology Change (TAC Nos. MB1807 and MB1884)," dated February 21, 2002 (ML020580612)

4.6-8 NRC letter to 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)

4.7 TPBAR Interface Issue 9: Ultimate Heat Sink

NUREG-1672, Section 2.9.1, "DOE evaluated the effect of TPBARs on the ultimate heat sink (UHS) for the reference plant against the guidance of SRP Section 9.2.5. The acceptance criteria specified in the SRP are based on meeting the relevant requirements of GDCs 2, 5, 44, 45, and 46 of Appendix A of 10 CFR Part 50. DOE states that the heat removal capability of the UHS may be affected by the TPC from the increase in the SFP heat load during cooldown operations and the subsequent effect on the component cooling water system and the station service water system. DOE concludes that the effect on the ultimate heat sink should be analyzed on a plant-specific basis. The staff agrees with this evaluation because the design of the ultimate heat sink is very plant-specific. The staff has identified this as an interface item that 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 TPBAR Interface Issue 9 information provided for the WBN Unit 1 and 2 Tritium Production Program Increase to 2,496 TPBAR LAR is based on the applicable WBN Unit 1 and 2 precedent documents (References 4.7-1 through 4.7-7). For the 2,496 TPBAR increase, the established site-specific methodology for analysis of the SFP heat load, which discharges heat to the UHS (via CCS and ERCW) is maintained but is updated to reflect the TPBAR increase.

The information regarding Interface Issue 9 in Reference 4.7-7 remains applicable to this license amendment request except as provided below.

<u>Tritium Impact on Decay Heat to ERCW</u>

Utilizing the methodology established on previous TPBAR projects at WBN, TVA has updated the quantitative analysis of expected impact of decay heat on the SFPCCS. The updates to these SFPCCS analyses have shown that the heat loads from these systems to CCS are bounded by what was analyzed for the previous increase to 1,792 TPBARs. The existing RHR System analysis was found to be bounding of the 2,496 TPBAR change such that there was no impact, beyond what was previously analyzed, on the CCS or the ERCW System. Interface Issue 12 (Section 4.9) provides a more detailed discussion of these changes.

As the CCS heat loads associated with the increase to 2,496 TPBARs are bounded by what was previously analyzed, there is no impact to the ERCW System with the increase to 2,496 TPBARs.

<u>Impact on Heat Rejection to the UHS</u>

As the CCS heat loads are bounded by what was previously analyzed, there is no impact to the UHS with the increase to 2,496 TPBARs.

<u>UHS Summary</u>

The UHS has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by the increase to 2,496 TPBARs. Site procedures and cycle-specific analysis demonstrate that SFP heat loads are maintained within allowable limits [see Interface Issue 11 (Section 4.8) for further discussion]. Tritium production activities do not have an adverse impact on the UHS heat removal capabilities.

References

4.7-1 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.7-2 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015," dated August 20, 2001 (ML012390106 and ML012390115)

4.7-3 TVA letter to NRC, "Watts Bar Nuclear Plant - Responses to Request for Additional Information (RAI) Regarding Tritium Production - Interface Item Numbers 1, 6, 7, 10, 11, and 12 (TAC No. MB1884)," dated October 29, 2001 (ML020320146)

4.7-4 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision to the Spent Fuel Pool Cooling Analysis Methodology," dated April 20, 2001 (ML011170181)

4.7-5 TVA letter to NRC, "Watts Bar Nuclear Plant -Responses to RAI Regarding Spent Fuel Pool Cooling Analysis Methodology (TAC No. MB1884)," dated November 14, 2001 (ML020150176)

4.7-6 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Spent Fuel Pool Cooling Analysis Methodology Change (TAC Nos. MB1807 and MB1884)," dated February 21, 2002 (ML020580612)

4.7-7 TVA letter to NRC, 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)

4.8 TPBAR Interface Issue 11: Spent Fuel Pool Cooling and Cleanup System

NUREG-1672, Section 2.9.3, "The staff has reviewed the information presented by DOE and concludes that the calculations performed by DOE may not represent the actual increase in pool temperature from incorporation of the TPBARs. However, on the basis of information submitted by DOE in its letter dated January 13, 1999, the decay heat generated by the TPBARs is very low; each TPBAR generates less than 3 watts of heat at 150 hours after reactor shutdown. The maximum temperature increase of a TPBAR due to internal heat generation is less than 3 °F. The reference plant could insert up to 3344 TPBARs in each reload. The total heat load increase due to TPBARs is about 0.003 percent compared with a 3565 MWt core rating of the reference plant. In considering its very low rate of heat generation, the staff concludes that the heat load increase from the incorporation of TPBARs in the spent fuel pool has an insignificant impact on the spent fuel pool heat load and the added heat load will be within the cooling capability of the SFPCS. However, further analysis with reliable data is required to determine the actual impact of the TPBARs. A quantitative analysis to determine the absolute spent fuel pool temperatures must be performed by licensees seeking to utilize a TPC because the capacity of the spent fuel pool and its associated cooling system design are very plant specific. The staff has identified this as an interface item that 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 TPBAR Interface Issue 11 information provided for the WBN Unit 1 and 2 Tritium Production Program Increase to 2,496 TPBAR LAR is based on the applicable WBN Unit 1 and 2 precedent documents (References 4.8-1 through 4.8-7). For the 2,496 TPBAR increase, the established site-specific methodology for analysis is maintained but is updated to reflect the TPBAR increase.

The information regarding this interface issue in Reference 4.8-7 remains applicable to this license amendment request except as provided below.

The design basis heat load for the spent fuel pool is limited to 28.1 million British thermal units per hour (MBTU/hr). This limit is based on the design basis maximum UHS temperature and design basis heat exchanger fouling conditions. Under this loading condition, dual-train operation (two pumps and two heat exchangers) maintain the pool water temperature at or below 124.7°F. Similarly, single train operation (one pump and one heat exchanger) maintains the pool water temperature at or below 151.2°F.

Cycle specific calculations are performed prior to the start of a refueling outage to determine the exact heat removal capability of the SFP system using recent heat exchanger performance testing and anticipated UHS temperatures; otherwise, the 28.1 MBTU/hr may not be exceeded. The cycle specific calculations take into consideration the exact heat removal capability of the SFP system using recent heat exchanger performance testing and anticipated UHS temperatures. Under these more favorable conditions, up to 47.4 MBTU/hr can be accommodated. Operating procedures provide the controls to ensure these limitations are met.

TVA calculations support limits on SFP temperatures and decay heat loadings including the following.

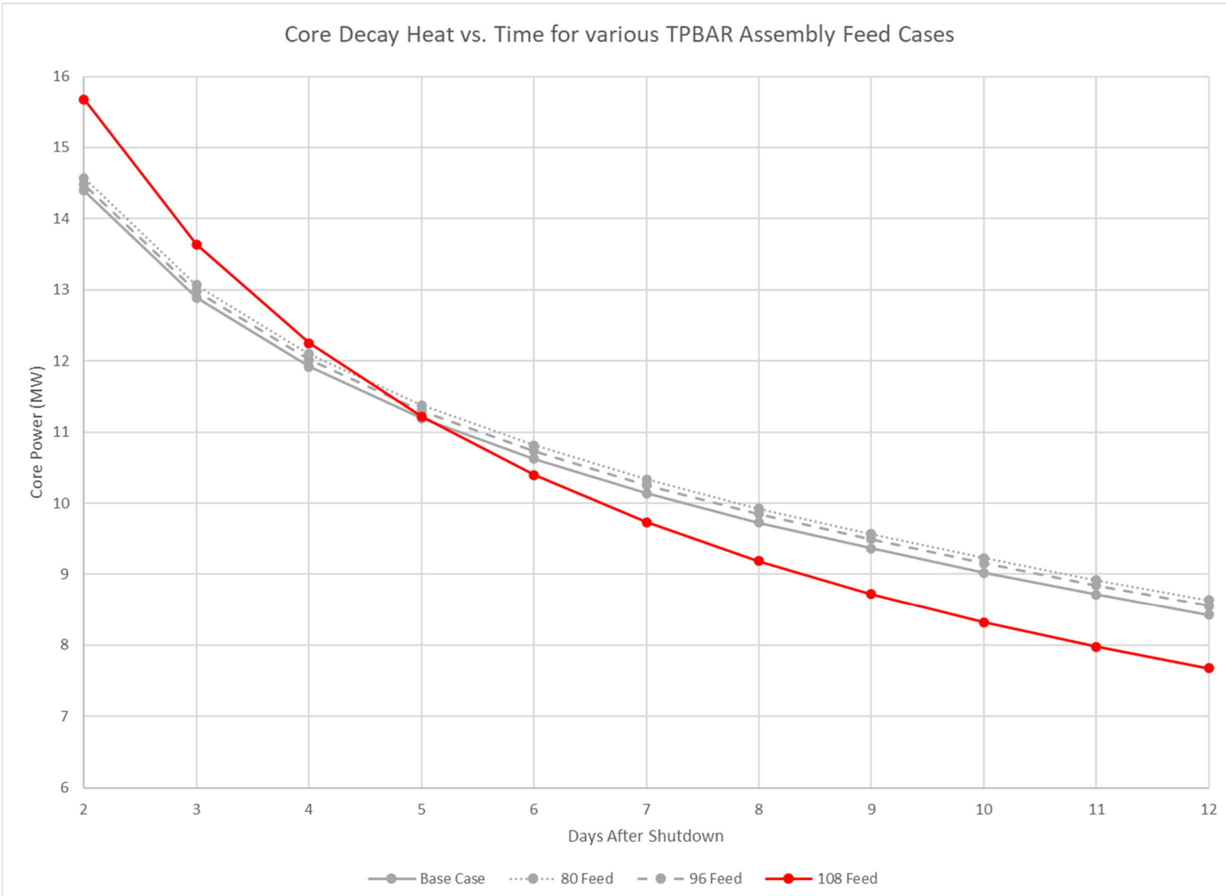
- Determination of core offload heat load limits and times for normal, emergency, and residual heat removal cooldown conditions under design basis conditions.
- Using decay heat load, determination of bulk water temperature, local water temperatures, fuel clad temperatures, departure from nucleate boiling, and time-to-boil.
- Determination of the impact of tritium production on SFP decay heat loads and the resultant impact on core offloading activities.
- Determination of alternative analysis to predict SFP transient thermal response for off-design values of SFP system heat exchanger fouling and component cooling water system (CCWS) temperatures less than design maximum of 95°F.

Design basis calculations provide input to successor calculations which evaluate the impacts on connected systems and components. These calculations conservatively assume a completely full SFP. TVA completes regular dry cask campaigns to minimize the spent fuel in the pool, resulting in heat loads which are less than what are conservatively assumed in these calculations.

Evaluation of Impact of the Increase to 2,496 TPBARs

The 2,496 TPBAR equilibrium fuel cycle considers a 108-assembly feed case and includes historic information for a base case, 80-assembly feed cases, and 96-assembly feed cases. Figure 4.8-1 provides a comparison of the four cases; the red line is the 2,496 TPBAR case and the gray lines are the historic cases.

Figure 4.8-1



As shown in Figure 4.8-1, the 2,496 TPBAR core decay heat exceeds the previously analyzed cases for the first 4.5 days (96-feed case then becomes bounding). By day 6, the existing analysis cases are bounding of the 2,496 TPBAR case (108-feed case). The difference in these decay heats is due, in part, to a code change that is described below.

The core decay heat generation values for the historic cases (base, 80-feed, and 96-feed) were developed using the computer code DHEAT. DHEAT is utilized to predict post shutdown core decay heat and decay heat from older stored fuel. DHEAT is based on methodology contained in ANSI/ANS-5.1-1994, RG 3.54, and NUREG/CR-2397. Except for the 108-feed core, the data utilized in the analysis were based on results from DHEAT-generated data sets for a base, 80-feed, and 96-feed cores.

For the 108-feed core, the total decay heat from irradiated fuel and activated fuel assembly structural components were calculated using ORIGEN from the SCALE 6.1 code package. ORIGEN calculations from the SCALE Code Package form the technical basis for significant portions of RG 3.54 and have been validated against integral decay heat measurements in NUREG/CR-5625 and NUREG/CR-6999.

The earliest time in which a full core offload (all 193 assemblies) can be initiated is approximately 100 hours after shutdown. Using a slightly earlier time of 96 hours (four days) is conservative for the evaluation of the impact of increasing to 2,496 TPBARs because of the trend shown in the figure above. Additionally, the 96-hour time period is conservative as it does not account for the time it takes to transfer the full core to the SFP. At 96 hours after shutdown, the base case has a heat load of 40.6 MBTU/hr [11.9 megawatts (MW)\ while the 108-feed case has a heat load of 42.0 MBTU/hr (12.3 MW), resulting in a heat load increase of less than 3.5%. Both values are below the 47.4 MBTU/hr value that has been previously analyzed for cycle specific offload conditions. Therefore, the increase in SFP heat load at 96 hours is within the acceptable SFP system operating range so long as a cycle specific analysis is complete. At approximately 144 hours (6 days), the impact of the 2,496 TPBAR increase is bounded by the base case and no adverse impacts are anticipated for a full core offload. Regardless of the specific timing of the offload, TVA procedures are utilized to manage the decay heat in the SFP to stay within acceptable limits (further discussion of TVA procedures for this process is provided below).

The design analyses consider the worst-case fuel pool loading scenario with the pool completely full following a full core offload. This provides additional conservatism to these analyses as TVA frequently conducts dry cask fuel storage campaigns to maximize SFP slot availability and minimize the base heat load in the SFP due to previously discharged fuel batches.

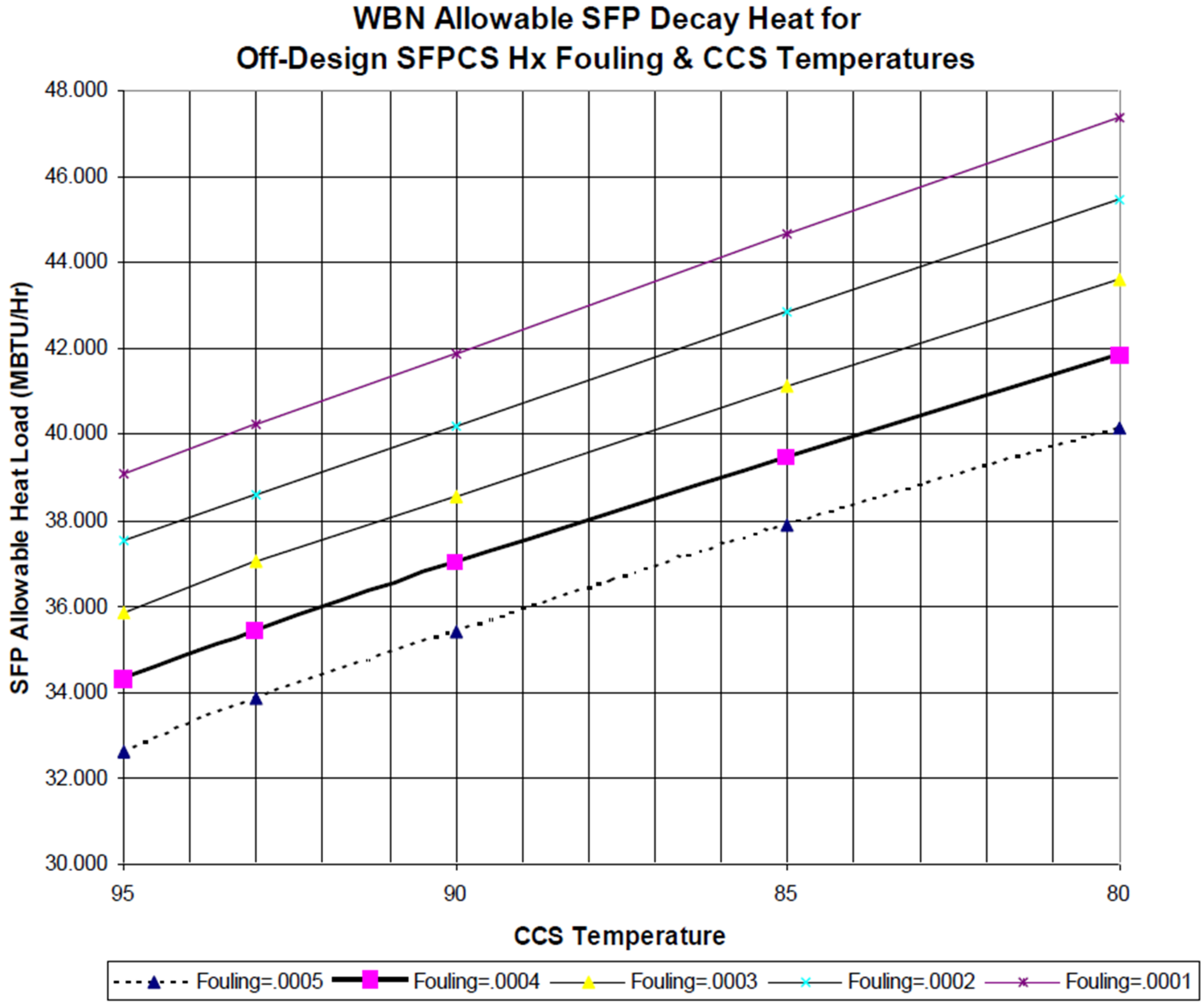
Cycle Specific Offload Analysis Methodology

TVA procedures provide guidance for the implementation of SFP heat loads higher than the design basis value of 28.1 MBTU/hr including:

- Current SFP heat exchanger fouling,
- The heat load at the start of core offload [based on cycle specific burnup, fuel design data, and the current spent fuel pool inventory (e.g., consideration for completion of any dry cask campaigns)], and
- Maximum expected river water temperatures during the refueling outage.

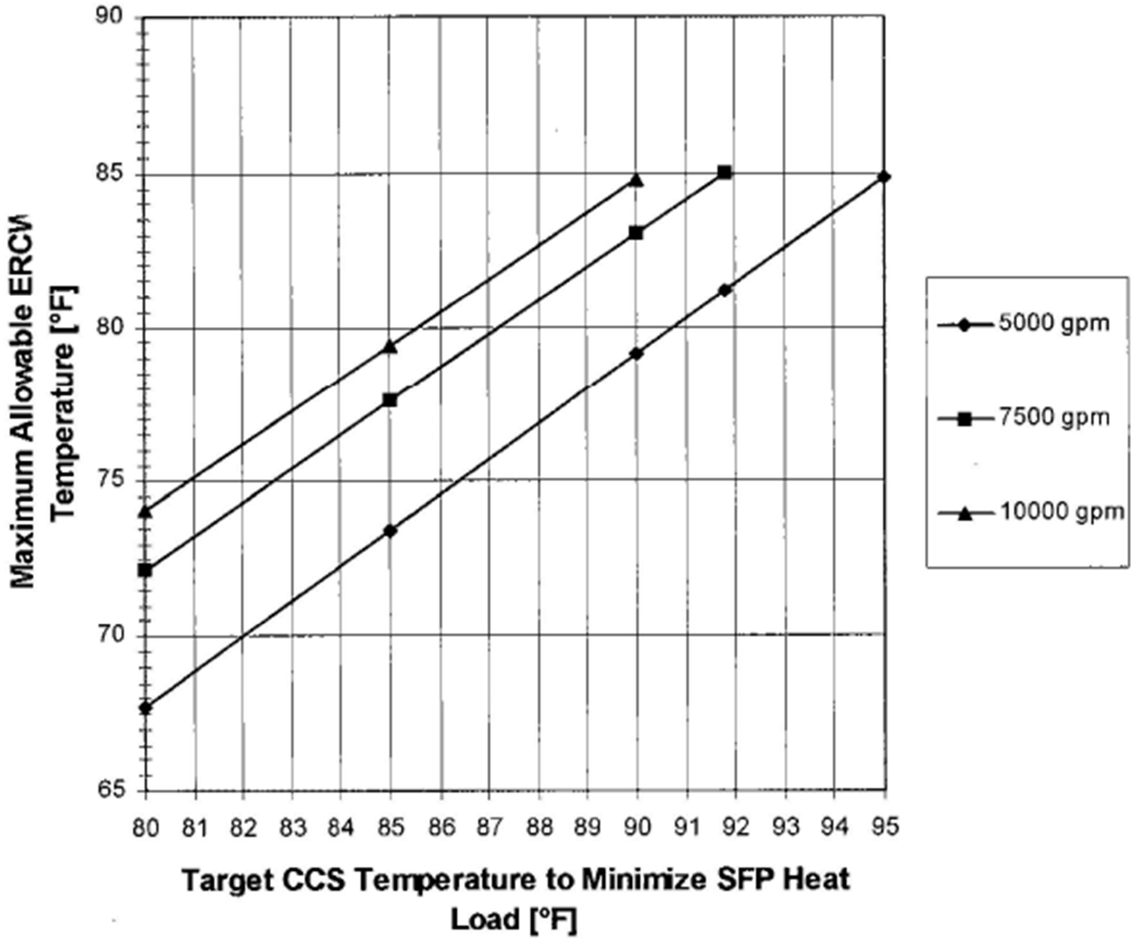
The SFP heat exchanger fouling and heat load at the start of core offload are used to determine the maximum component cooling system temperature that support removal of the heat load using Figure 4.8-2.

Figure 4.8-2


WBN Allowable SFP Decay Heat for
Off-Design SFPCS Hx Fouling & CCS Temperatures
SFP Allowable Heat Load (MBTU/Hr)
48.000
46.000
44.000
42.000
40.000
38.000
36.000
34.000
32.000
30.000
95
90
85
80
CCS Temperature
Fouling=.0005
Fouling=.0004
Fouling=.0003
Fouling=.0002
Fouling=.0001

The required CCS Temperature is then used in conjunction with the maximum expected river water temperature to determine the required ERCW flowrate using Figure 4.8-3.

Figure 4.8-3



The resultant ERCW flowrate is then used to ensure that adequate cooling is provided to the SFP system. If the ERCW required flowrate exceeds certain pre-determined values, further evaluation is required to determine if additional changes or limitations are needed to assure that SFP decay heat loading do not exceed the capacity of the SFPCCS, CCWS, and ERCW systems.

There is no impact to current station procedures and technical instructions due to the increase to 2,496 TPBARs. The current procedural requirements remain valid and applicable.

<u>Conclusions</u>

TVA design basis calculations, procedures, and other station documentation provide specific heat load requirements for offloading into the SFP. These documents provide the station with a rigorous, controlled mechanism for which the SFP heat load is managed to ensure all station commitments are met. Furthermore, the calculations which document the station design basis, conservatively assume a completely full SFP. TVA completes regular dry cask campaigns to minimize the spent fuel in the pool, resulting in heat loads which are less than what are conservatively assumed in these calculations.

The increase to 2,496 TPBARs results in modest changes to SFP heat loads and the resulting offload times. For the most limiting scenario of a full-core offload, the previously analyzed heat load is bounding of the 2,496 TPBARs case after approximately 4.5 days. For normal offloads under better than design performance conditions, procedural controls demonstrate that the offload schedule/timing preclude exceedance of analyzed station limits.

References

4.8-1 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.8-2 TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015,” dated August 20, 2001 (ML012390106 and ML012390115)

4.8-3 TVA letter to NRC, “Watts Bar Nuclear Plant - Responses to Request for Additional Information (RAI) Regarding Tritium Production - Interface Item Numbers 1, 6, 7, 10, 11, and 12 (TAC No. MB1884),” dated October 29, 2001 (ML020320146)

4.8-4 TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision to the Spent Fuel Pool Cooling Analysis Methodology,” dated April 20, 2001 (ML011170181)

4.8-5 TVA letter to NRC, “Watts Bar Nuclear Plant – Responses to RAI Regarding Spent Fuel Pool Cooling Analysis Methodology (TAC No. MB1884),” dated November 14, 2001 (ML020150176)

4.8-6 NRC letter to TVA, “Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Spent Fuel Pool Cooling Analysis Methodology Change (TAC Nos. MB1807 and MB1884),” dated February 21, 2002 (ML020580612)

4.8-7 TVA letter to NRC, 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)

4.9 TPBAR Interface Issue 12: Component Cooling Water System

NUREG- 1672, Section 2.9.4, "Because more fuel and TPBAR assemblies are removed from the core to the spent fuel pool during refueling, the maximum pool temperature will increase. Although the effect of the TPBARs on the CCWS is insignificant because the heat load generated by the TPBARs only amounts to about 3 watts per rod 150 hours after reactor shutdown, a substantial increase in heat load occurs as a result of a full core off-load. The additional heat load generated by the TPC to the spent fuel pool heat exchangers could increase the demand for CCWS flow. DOE stated that the system heat transfer and flow requirements may be affected by the TPBARs from the increase in spent fuel pool heat load during cooldown operations, and the effect on this system will need to be analyzed on a plant-specific basis. In response to the staff's RAI, DOE also stated that the increased spent fuel pool heat load does not come from the presence of TPBARs but from the increased number of fuel assemblies being replaced. The staff has identified this as an interface item that 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 TPBAR Interface Issue 12 information provided for the WBN Unit 1 and 2 Tritium Production Program Increase to 2,496 TPBAR LAR is based on the applicable WBN Unit 1 and 2 precedent documents (References 4.9-1 through 4.9-7). For the 2,496 TPBAR increase, the established site-specific methodology for analysis of the spent fuel pool heat load, which rejects heat to the CCS is maintained but is updated to reflect the TPBAR increase.

The information regarding Interface Issue 12 in Reference 4.9-7 remains applicable to this license amendment request except as provided below.

<u>Tritium Impact on RHR System Heat Loads</u>

The RHR and CCS systems can support a station cooldown after 7 hours. At 7 hours, the core decay heat is expected to be 27.3 MW (93.1 MBTU/hr) with the increase to 2,496 TPBARs. The RHR system heat load for cooldown at 7 hours corresponds to a core decay heat value of 28.6 MW (97.5 MBTU/hr). The expected core decay heat load (93.1 MBTU/hr) for 2,496 TPBARs is bounded by the current analysis of record (97.5 MBTU/hr), no adverse impacts to RHR cooldown and CCS are anticipated during cooldown.

<u>Tritium Impact on SFP System Heat Loads</u>

Utilizing the methodology established on previous TPBAR projects at WBN, TVA has updated the quantitative analysis of expected spent fuel decay heat for both TPCs and non-TPCs. Interface Issue 11 (Section 4.8) provides a detailed summary of the expected changes in SFP System heat loads. The following provides a detailed breakdown of how those changes in heat loads impact the CCS. This analysis is based on conservative, full pool SFP conditions for dual unit operation. TVA completes regular dry cask campaigns to minimize the spent fuel in the pool, resulting in heat loads which are less than what are conservatively assumed in this analysis.

Heat Generation Methodology

The SFP decay heat generation values for the base (non-tritium core), 80-feed, and 96-feed cores utilize the same methodology as applied in previous TPBAR projects at WBN.

For the 108-feed core, the total decay heat from irradiated fuel and activated fuel assembly structural components were calculated using ORIGEN from the SCALE 6.1 code package by Westinghouse. ORIGEN calculations from the SCALE Code Package form the technical basis for significant portions of RG 3.54 and have been validated against integral decay heat measurements in NUREG/CR-5625 and NUREG/CR-6999.

Projected Full Core Offload (193 Assemblies) Decay Heat Impact in the SFP

In accordance with WBN Units 1 and 2 TS 3.9.10, “Decay Time,” the earliest time in which core offload can be initiated is at 100 hours after core shutdown (i.e., after the reactor is subcritical). From plant experience, the latest time in which core offload is likely to begin is approximately 10 days after core shutdown. Station experience shows that offloads have typically started at day 6. The period from 100 hours, day 4, to day 10, represents the period in which a full core offload is most likely to begin. Because for any given outage, start of offload is predicated on outage management efficiencies, not design parameters, the estimated impact was taken as the average between the day 4 and day 10 effects. By utilizing DHEAT generated data (day 4r and day 10) for the 80-feed and 96-feed cores and comparing this data to the equivalent data of the base core, the results in Table 4.9-1 were determined, after averaging the day 4 and day 10 results. Similarly, ORIGEN generated data (day 4 and day 10) was utilized for the 108-feed core. The values presented in Table 4.9-1 are for a full core offload of 193 fuel assemblies.

Table 4.9-1: Increased Heat Load over Base Case

Feed Case	Day 4 (MWt)	Day 10 (MWt)	Average (MWt)
TPBAR 80-Feed Case	0.1818	0.2054	0.1936
TPBAR 96-Feed Case	0.0994	0.1304	0.1149
TPBAR 108-Feed Case	0.3363	-0.7014	-0.1826

Projected SFP Residual Heat Impact (Previously Discharged Batches)

For every refueling outage, there is an increase in residual heat in the SFP resulting from the addition of spent fuel to the pool. From inspection of the generated data for the multiple feed cases, the 96-feed Case residual decay heat values were found to be the greatest when compared to the other cases. This result is based on the latest compiled data by Westinghouse for all feed cases.

For WBN, the maximum design SFP capacity is 1,386 cells. A full core offload requires enough SFP area to store 193 fuel assemblies; therefore, the maximum number of cells allowed for general fuel storage is 1,193 cells. The core decay heat data used in the alternate SFP decay heat analysis for dual unit operation is based on conservative, bounding decay heat values for typical tritium production 80-feed assembly core, a 96-feed assembly core, or a 108-feed assembly core.

For the analysis of core offload time for dual unit operations, the 12th cycle would fill the SFP (i.e., if starting with Unit 1, Cycle 1, the SFP would no longer have the capacity for another discharge if Unit 2, Cycle 6 was a full core offload). Therefore, this analysis is based on 11 cycles discharged to the SFP. This analysis conservatively assumed initial makeup to bring the pool to capacity (i.e., additional assemblies were added at the time of Unit 1, Cycle 1.) Note that this analysis is conservative as it assumes both Unit 1 and Unit 2 discharge a 108-feed assemble core, causing the SFP to fill up faster, allowing for fewer core offloads, thus causing the SFP in this analysis to be filled with younger fuel which increases the decay heat.

For dual unit operation, assuming a bounding core (i.e., limiting for 80-, 96-, and 108-feed cores) would result in up-to a two-cycle difference before reaching full pool conditions, the effect of tritium on the SFP was determined to be at most an increase in residual heat of 0.1013 MWt.

Net SFP Decay Heat Impact Related to Tritium Production Activities

The net SFP decay heat impact for dual unit operations related to tritium production activities was obtained by adding the tritium impacts on core decay heat (80-feed, 96-feed, and 108-feed TPBAR Cases) and the limiting value for the SFP residual decay heat (previously discharge batches in the SFP) as shown in Table 4.9-2:

Table 4.9-2: Net SFP Heat Load Impact

Feed Case	**A: Impact of 193 Freshly Discharged Assemblies (MWt)**	**B: Impact of 1,193 “Legacy” Discharged Assemblies (MWt)**	**A + B: Net Impact (MWt)**
TPBAR 80-Feed Case	0.1936	0.1013	0.2949
TPBAR 96-Feed Case	0.1149	0.1013	0.2162
TPBAR 108-Feed Case	-0.1826	0.1013	-0.0813

The overall residual heat impacts from the tritium production activities ranges from 0.3 MWt to no significant impact (relative to a non-TPC). A conservative maximum approximation of the heat load increase for 2,496 TPBARs operation is 0.3 MWt relative to a non-TPC. This impact is unchanged from what was previously analyzed for operation of up to 1,792 TPBARs where the net impact was calculated to be up to 0.35 MWt relative to a non-TPC.

Results / Conclusions

Based on the analysis, it was shown that tritium production activities would have a minor impact on SFP decay heat loads. However, the increase from 1,792 TPBARs to 2,496 TPBARs does not result in any change in the net SFP decay heat impact due to tritium production activities (e.g., the 0.35 MWt value analyzed in Reference 4.9-7 remains bounding).

SFP Cooling Heat Rejection on CCS

As part of previous evaluations to support the increase to 1,792 TPBARs, the increase in SFP cooling rejected to CCS was evaluated for an increase in heat rejection. As described above, the evaluation for 2,496 TPBARs has determined that this previously analyzed value remains bounding. Therefore, the analysis of the impacts to CCS from the 1,792 TPBARs remains bounding for the increase to 2,496 TPBARs.

CCS Summary

The CCS has adequate capacity and cooling margin to perform its safety and non-safety related functions with the additional heat loads imposed by the increase to 2,496 TPBARs. Tritium production activities do not have an adverse impact on the CCS heat removal capabilities.

References

4.9-1 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.9-2 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 - Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) -Technical Specification (TS) Change No. TVA-WBN-TS-00-015," dated August 20, 2001 (ML012390106 and ML012390115)

4.9-3 TVA letter to NRC, "Watts Bar Nuclear Plant - Responses to Request for Additional Information (RAI) Regarding Tritium Production - Interface Item Numbers 1, 6, 7, 10, 11, and 12 (TAC No. MB1884)," dated October 29, 2001 (ML020320146)

4.9-4 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) - Unit 1 -Revision to the Spent Fuel Pool Cooling Analysis Methodology," dated April 20, 2001 (ML011170181)

4.9-5 TVA letter to NRC, "Watts Bar Nuclear Plant -Responses to RAI Regarding Spent Fuel Pool Cooling Analysis Methodology (TAC No. MB1884)," dated November 14, 2001 (ML020150176)

4.9-6 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 -Issuance of Amendment Regarding Spent Fuel Pool Cooling Analysis Methodology Change (TAC Nos. MB1807 and MB1884)," dated February 21, 2002 (ML020580612)

4.9-7 TVA letter to NRC, 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)

4.10 TPBAR Interface Issue 13: Demineralized Water Makeup System

NUREG-1672, Section 2.9.5, "The staff has reviewed the information presented by DOE and concludes that the incorporation of TPBARs in the reference plant does not have any significant impact on the demineralized water makeup system because only a very small quantity of tritium is released from the TPBARs to the primary coolant system. Because the design of the demineralized water makeup system is plant-specific, DOE concludes, and the staff agrees, that a detailed analysis for this effect is required from licensees participating in DOE's program for the CLWR production of tritium. The staff has identified this as an interface item that 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 TPBAR Interface Issue 13 information provided for the WBN Units 1 and 2 Tritium Production Program increase to 2,496 is based on the applicable WBN Unit 1 and 2 precedent documents (References 4.10-1 through 4.10-5). For the 2,496 TPBAR increase, the evaluation is summarized below.

Technical Evaluation

The demineralized water makeup system at WBN is designed to supply high purity makeup water to the SGs, to the primary water system, to the demineralized water and cask decontamination system, to clean, flush, and provide makeup for miscellaneous services. The makeup water treatment plant (MWTP) is designed to supply the filtered and demineralized water required for both units. The demineralized water, storage, and distribution system (DMWSDS) receives demineralized water from the MWTP, stores it, and distributes high purity demineralized water.

The impact of increasing TPBARs in the core increases tritium levels in the RCS due to additional tritium permeation through the TPBARs. NUREG-1672 concludes that a licensee must analyze the plant-specific capability of the demineralized water makeup system (DWMS) because this system differs plant to plant. As noted in Interface Issue 5 (Section 4.4 to this enclosure), TVA has 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. There is no regulatory limit on RCS tritium level, but the MWTP has the capacity to ensure sufficient demineralized water makeup capacity to accommodate more frequent dilution activities if required. Other MWTP interfacing systems (e.g., raw service water system, potable water system, station drainage system, service air system) are not impacted by the increase in TPBARs in the core.

See Interface Item 14 (Section 4.11) for further evaluation of radioactive waste management and design dose rates relative to regulatory criteria. The Interface Item 14 evaluation concludes that WBN continues to be in compliance with as low as reasonably achievable (ALARA) dose objectives per 10 CFR Part 50 Appendix I and liquid and gaseous radwaste release concentrations will continue to meet the limits of 10 CFR Part 20.

Demineralized Water Makeup System Summary

Increased TPBARs in the core increase tritium levels in the RCS due to additional tritium permeation. Though there is no regulatory limit on RCS tritium level, RCS tritium concentration can be managed with additional dilution activities, if necessary. The MWTP 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 DMWSDS, which stores and distributes water to the primary water system. A dedicated water source supplies raw water to the MWTP, with backup supply sources being the RCW pumps and a high pressure fire protection (HPFP) pump.

See Interface Item 14 (Section 4.11) for further evaluation of radioactive waste management and design dose rates relative to regulatory criteria.

References

4.10-1 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.10-2 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015," dated August 20, 2001 (ML012390106 and ML012390115)

4.10-3 TVA letter to NRC, "Watts Bar Nuclear Plant - Responses to Request for Additional Information (RAI) Regarding Tritium Production - Interface Item Numbers 1, 6, 7, 10, 11, and 12 (TAC No. MB1884)," dated October 29, 2001 (ML020320146)

4.10-4 NUREG-0847, Supplement 23, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2," dated July 2011 (ML11206A499)

4.10-5 NUREG-0847, Supplement 26, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2," dated June 2013 (ML13205A136)

4.11 TPBAR Interface Issue 14: Liquid Waste Management Systems

NUREG-1672, Section 2.11.2, "On the basis of the preceding discussion, the staff concludes that in both cases (the design-basis TPBAR permeation of tritium and the failure of two TPBARs) there is a sufficient margin in the reference plant so that the applicable release concentration and dose limits as presented in the plant technical specifications and ODCM will still be met even with the TPC operation. However, enhanced plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event are required. Furthermore, when the TPC topical report is applied to a candidate plant, a plant-specific analysis will be needed to demonstrate that the plant continuously meets release concentration and dose limits. The staff concludes that the methodology described in Section 2.11.3 of the TPC topical report is acceptable for plant-specific analysis. The staff has identified this as an interface item that 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 TPBAR Interface Issue 14 information provided for the WBN Unit 1 and WBN Unit 2 Tritium Production Program LAR is based on the relevant WBN Units 1 and 2 precedent documents (References 4.11-1 and 4.11-2, respectively), which authorized the irradiation of up to 1,792 TPBARs in WBN Units 1 and 2. The TVA application for those amendments (References 4.11-3 and 4.11-4, respectively) provided radiological analyses based on 1,792 TPBARs and a functional requirement of 5 Ci/TPBAR/year tritium permeation.

This section builds on the previously approved WBN Unit 1 and WBN Unit 2 analysis 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 is considered. This section addresses the two items discussed in the interface issue.

Plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event.

Plant-specific analyses to demonstrate that the plant continuously meets release concentration and dose limits.

Plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event

The description of the plant-specific tritium monitoring and surveillance programs and procedures for operator actions on an abnormal tritium release event in Reference 4.11-4 remains applicable to this license amendment, except as noted below.

There can be significant uncertainties in both the total (calculated) cumulative tritium to-date and the projected cumulative tritium generated from non-TPBAR sources. This results in a significant uncertainty in the amount of tritium attributable to TPBARs. The estimated cumulative tritium permeation per TPBAR for WBN tritium permeation for Unit 1 Cycles 6 through 17 and Unit 2 Cycle 4 with a 90 percent uncertainty is shown in Table 4.11-1.

Table 4.11-1 Estimated TPBAR Permeation for WBN Unit 1 Cycles 6 through 17 and Unit 2 Cycle 4			
Cycle Number	Cycle Length (EFPD)	End of Cycle (Ci/TPBAR)	**Last 365 Days (Ci/TPBAR/year)**
WBN Unit 1			
Cycle 6	483.7	3.5±1.1	3.3±1.2
Cycle 7	489.5	3.5±1.1	3.2±1.3
Cycle 8	432.1	2.8±0.8	2.7±0.9
Cycle 9	516.6	3.8±0.8	3.4±0.8
Cycle 10	513.3	4.3±0.7	4.0±0.8
Cycle 11	458.7	3.5±0.5	3.4±0.5
Cycle 12	501.5	3.6±0.6	3.2±0.6
Cycle 13	487.4	3.9±0.3	3.5±0.3
Cycle 14	484.1	3.4±0.3	3.1±0.3
Cycle 15	489.3	3.8±0.3	3.6±0.3
Cycle 16	497.8	4.5±0.4	3.9±0.3
Cycle 17	504.2	4.2±0.3	3.9±0.2
WBN Unit 2			
Cycle 4	402.0	3.5±1.7	3.4±1.7

Tritium Bioassay Program

The description of the tritium bioassay program in Interface Issue 14 in Reference 4.11-4 remains applicable to this license amendment. As noted in Reference 4.11-4, TVA procedures RCI-137, "Radiation Protection Tritium Control Program," NPG-SPP-05.1, "Radiological Controls," and RCDP-7, "Bioassay and Internal Dose Program," provide a graded approach for bioassay based on risk, work, and airborne conditions. These procedures also contain controls to ensure conformance with RG 8.32, “Criteria for Establishing a Tritium Bioassay Program,” and 10 CFR 20.1702, “Use of Other Controls.”

Plant-specific analyses to demonstrate that the plant continuously meets release concentration and dose limits

Tritium Source Term Definition and Discussion

The description of the tritium source term definition and discussion in Reference 4.11-4 remains applicable to this license amendment, except as noted below.

The radwaste system design basis source term and the realistic source term both addressed a TPC by adding a tritium source term based on 2,496 TPBARs and a permeation rate of 5 Ci/TPBAR/year. This permeation rate bounds that observed for WBN Unit 1 and WBN Unit 2 and is consistent with that approved by the NRC for WBN Unit 1 (Reference 4.11-1) and WBN Unit 2 (Reference 4.11-2).

As with other tritium producing components (e.g., fuel rods, control rods, secondary neutron source rods) some of the free tritium inventory in the TPBARs will permeate the cladding material and be released to the primary coolant. The design goal for this permeation process is to keep the tritium permeation ALARA. TPBAR permeation is nonlinear with respect to the core’s effective full power days (see Figures 4.11-2A and 4.11-2B). A typical TPBAR’s tritium inventory begins at zero at the start of the irradiation cycle and ends with about 9,200 Ci per TPBAR of tritium at the end of the irradiation cycle. TPBAR tritium permeation increases with the maximum permeation rates towards the end of the cycle. Figure 4.11-1A demonstrates this process by using Unit 1 Cycle 17 and Figure 4.11-1B for Unit 2 Cycle 4.

Figure 4.11-1A: WBN UNIT 1 Cycle 17 Estimated Total Non-TPBAR and Total Tritium Production/Releases to the RCS

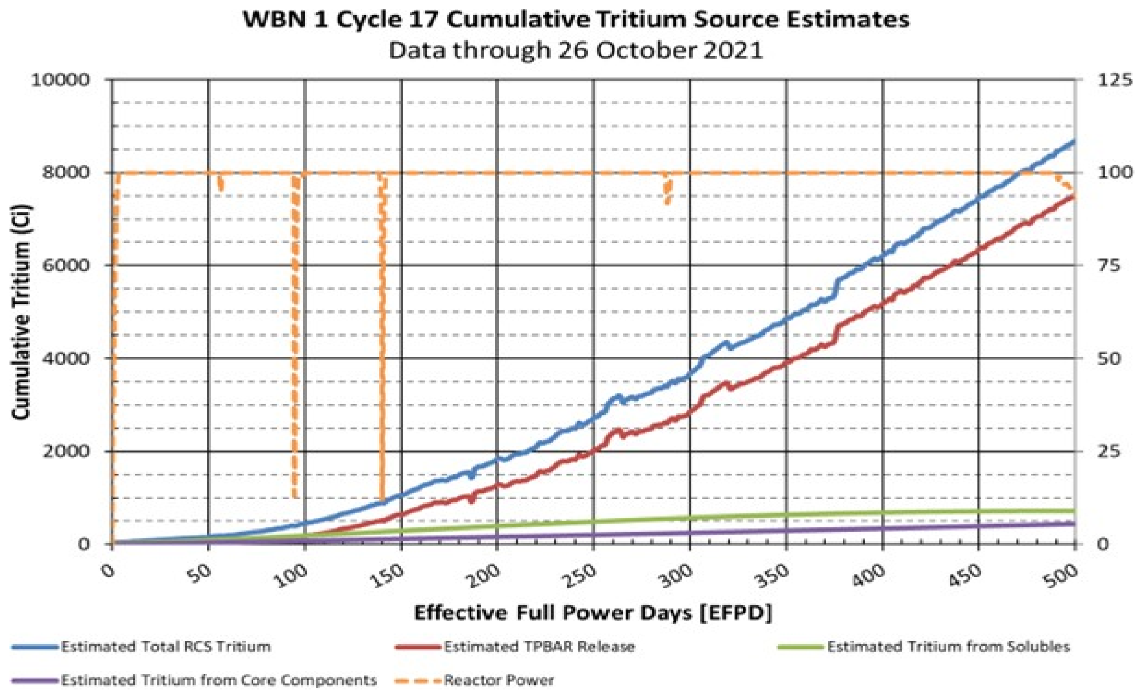
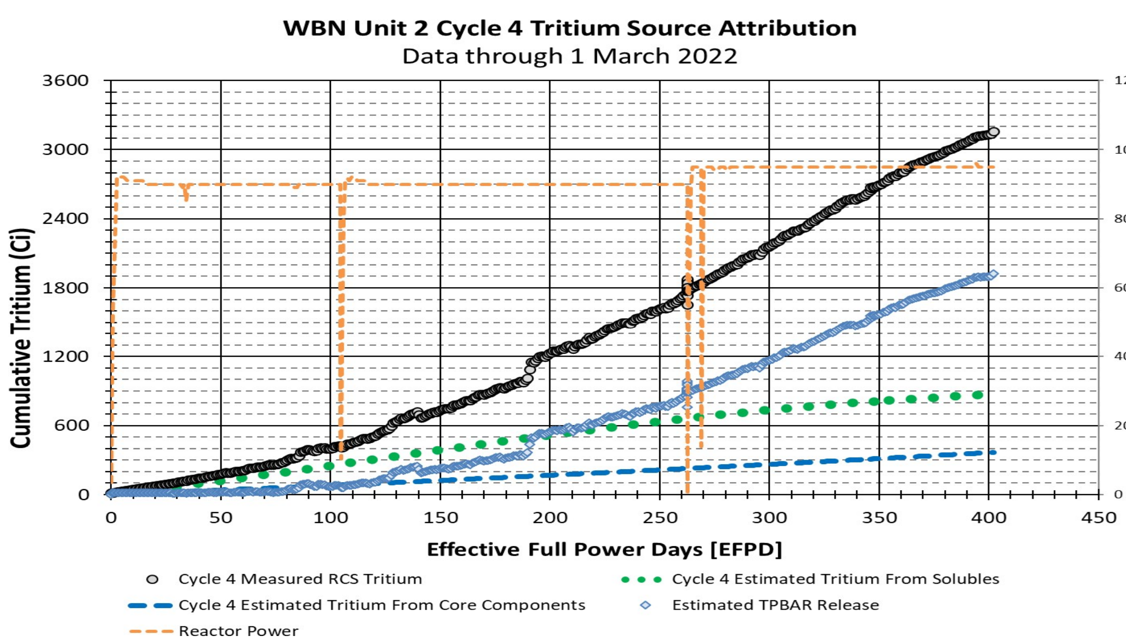


Figure 4.11-1B: WBN UNIT 2 Cycle 4 Estimated Total Non-TPBAR and Total Tritium Production/Releases to the RCS



The Mark 9.2 TPBAR design estimated cumulative tritium permeation per cycle time in EFPD per TPBAR for WBN Unit 1 Tritium Production Cycles 11 through 17 are shown in Figure 4.11-2A and for Unit 2 Cycle 4 in Figure 4.11-2B. The uncertainty bars represent the 90 percent confidence interval.

Figure 4.11-2A: Estimated TPBAR Permeation for WBN Unit 1 Cycles 11 through 17

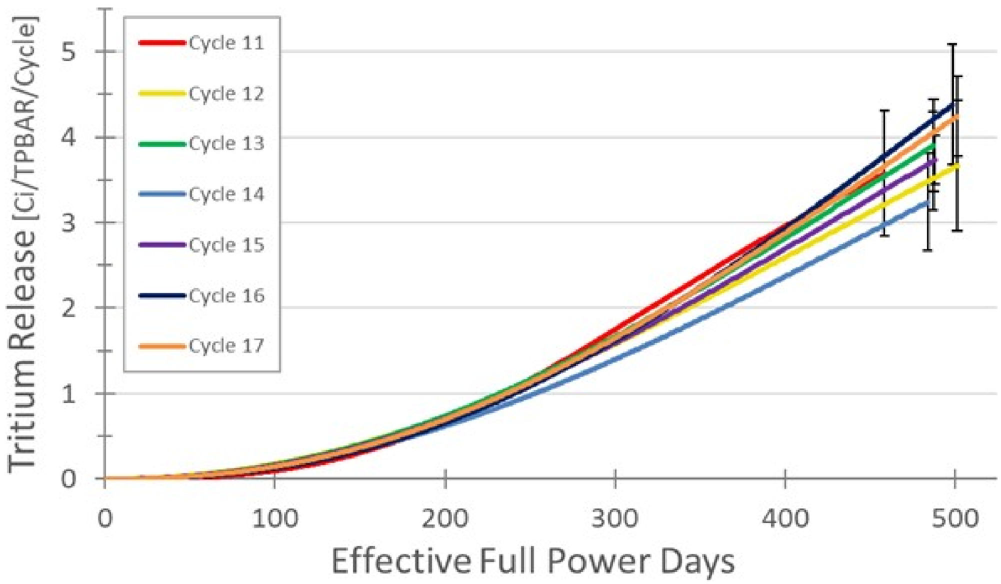
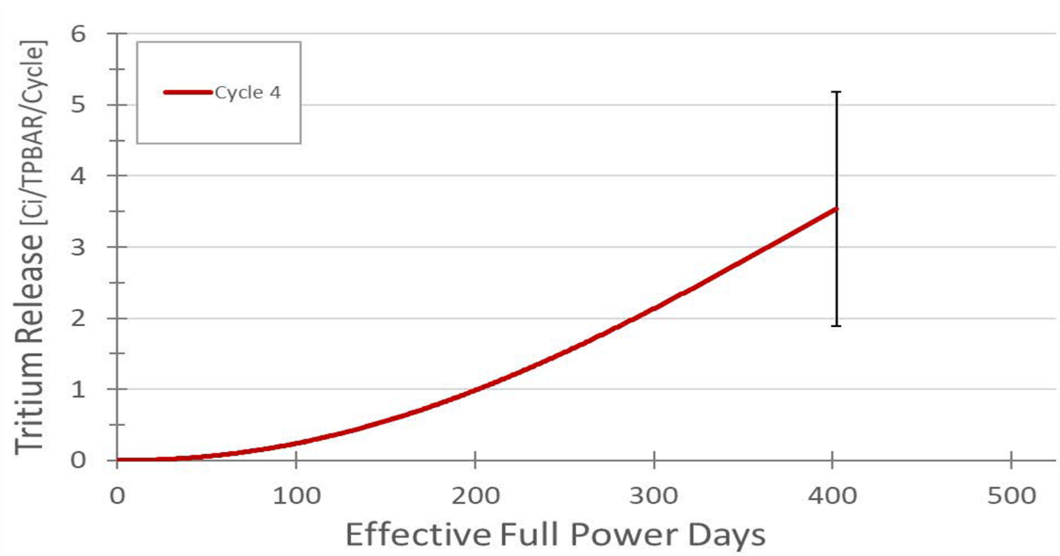


Figure 4.11-2B: Estimated TPBAR Permeation for WBN Unit 2 Cycle 4



When the TPBAR permeation estimates are presented in a calendar year format (see Figures 4.11-3A and 4.11-3B), corresponding to the NRC monitoring and reporting requirements, the annualized per TPBAR permeation have consistently remained less than 3.5 Ci/year. With the approximate 18-month fuel cycles, portions of multiple (i.e., two) fuel cycles will occur periodically in the same calendar years. The uncertainty bars represent the 90 percent confidence interval.

Figure 4.11-3A: Estimated Annual TPBAR Permeation for WBN Unit 1 Cycles 11-through 17

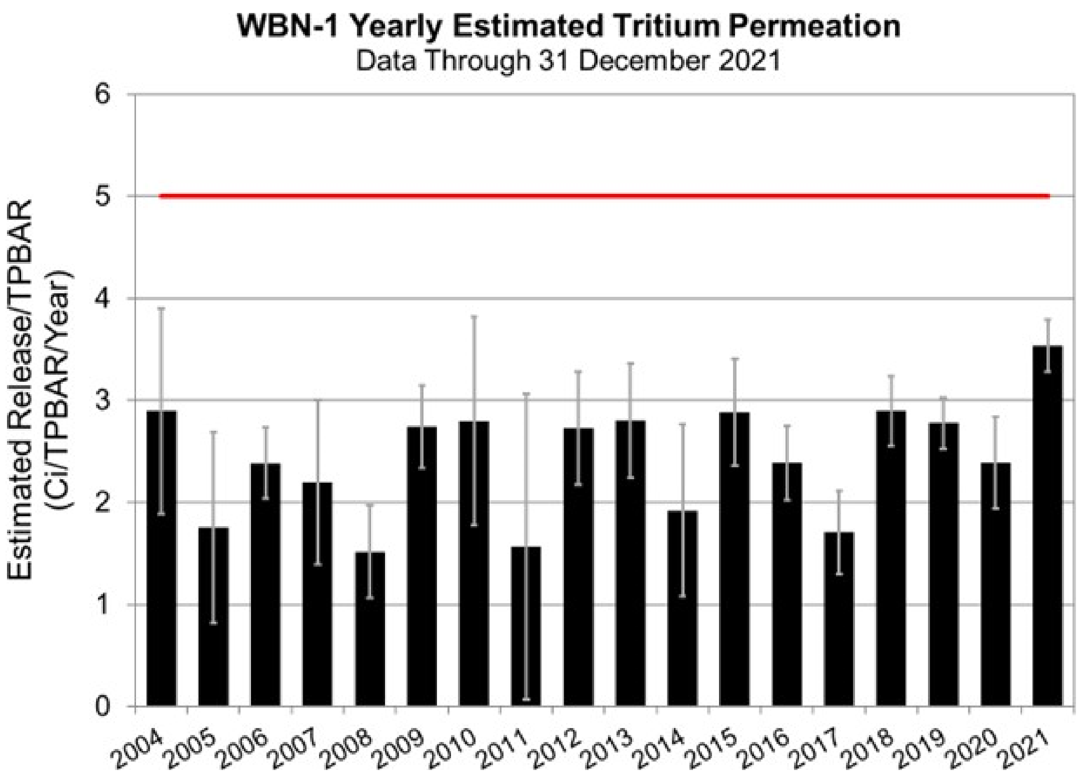
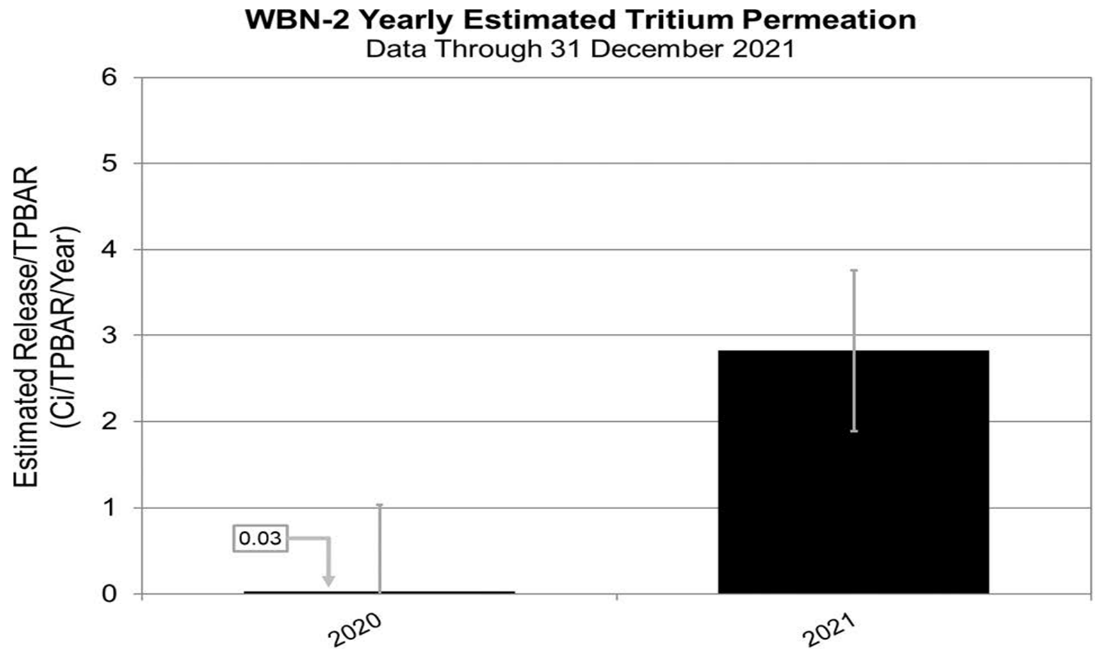


Figure 4.11-3B: Estimated Annual TPBAR Permeation for WBN Unit 2 Cycle 4



Radwaste System Design Basis Source Terms

For isotopes other than tritium, the annual release utilizing the design basis source term is determined by multiplying the expected annual release of each isotope based on the realistic source term by the ratio of the design basis RCS concentration (with 1% fuel defects) and the realistic primary coolant concentration. TVA has performed an analysis of the radioisotope core inventory for a TPC and is described in more detail in Interface Issue 5. A comparison of noble gas and iodine activities for a conventional core and a 2,496 TPC core is provided in Table 4.11-2. The Xe-135, I-131, I-132, and I-135 inventories are greater for the 2,496 TPC. These increases are offset by the decreases in the other isotopes and so any change would be insignificant. Therefore, the design basis RCS concentrations, other than tritium, currently used to determine the design basis releases remain applicable for a TPC. This is consistent with the approval of the TPC for WBN Unit 1 in Reference 4.11-1 and WBN Unit 2 in Reference 4.11-2.

Table 4.11-2: Radioisotope Non-TPC and TPC Comparison

Isotope	Total Core Inventory (Ci)	
	Conventional Core	TPC
Kr 85m	3.95E+07	2.62E+07
Kr 85	9.99E+05	9.58E+05
Kr 87	7.59E+07	5.21E+07
Kr 88	1.08E+08	6.99E+07
Xe 133	2.03E+08	1.84E+08
Xe 135m	5.46E+07	3.97E+07
Xe 135	5.55E+07	5.95E+07
Xe 138	1.79E+08	1.67E+08
I 131	8.80E+07	9.35E+07
I 132	1.34E+08	1.37E+08
I 133	1.97E+08	1.94E+08
I 134	2.31E+08	2.19E+08
I 135	1.79E+08	1.85E+08

Note 1: TPC is 108-Feed Equilibrium Core End-of-Cycle Operation at 3,480 MWt for 510 days.

The radwaste system design basis tritium source term was updated to account for the increase in tritium by adding the annual release from the TPBARs to that currently assumed for a non-TPC. The current non-TPC tritium source term is based on NUREG-0017 (Reference 4.11-6) and is calculated to be 1,392 Ci/year. The contribution from TPBARs is calculated to be 12,480 Ci/year (2,496 TPBARs at 5 Ci/TPBAR/ year). This results in an assumed total annual tritium release of 13,872 Ci/year. This was used to demonstrate the adequacy of the liquid and gaseous radwaste systems to meet the limits in 10 CFR Part 20.

Realistic Source Terms

The NRC's regulatory guidance on WBN's nominal tritium production is located in NUREG-0017. The calculated realistic average annual tritium per unit value from NUREG-0017 is 1,392 Ci. To account for a TPC, an additional 12,480 Ci/year (2,496 TPBARs at 5 Ci/year) was used. Therefore, a total average annual 13,872 Ci of tritium for a TPC was used to demonstrate continued compliance with the offsite ALARA dose objectives of 10 CFR Part 50 Appendix I.

Consistent with References 4.11-1 and 4.11-2, the realistic source term is different than what was assumed in the DOE topical report as the contribution of two failed TPBARs is not considered. No TPBAR failures are assumed because such failures are not expected or realistic.

Radwaste System Design Basis Operation

The liquid and gaseous radwaste system design basis demonstrates that there will be minimal impact to the treatment of fission and corrosion products with both units operating with 2,496 TPBARs.

Effluent releases to the environment are controlled to meet 10 CFR Part 20 release limits by WBN Unit 1 and WBN Unit 2 TS 5.7.2.7, Radioactive Effluent Controls Program. The Radioactive Effluent Release Report is submitted to NRC as required by WBN Unit 1 and WBN Unit 2 TS 5.9.3, "Radioactive Effluent Release Report." The report includes a summary of the quantities of radioactive liquid and gaseous effluents released from each unit and quantity of solid waste released from the site.

Release of the radioactive liquids from the liquid waste system is made only after laboratory analysis of the tank contents. If the activity is not below the offsite dose calculation manual (ODCM) limits, the liquid waste streams are returned to the waste disposal system for further processing by the mobile demineralizer. When the liquid waste meets ODCM limits, it is pumped to the discharge pipe through a normally locked closed manual valve and a remotely operated control valve, interlocked with a radiation monitor and a flow element in the cooling tower blowdown (CTB) line. This assures that sufficient CTB dilution flow is available for the discharge of radioactive liquids. A minimum of 30,000 gpm CTB dilution flow is required for discharge of radioactivity.

WBN has three large tanks in the liquid radwaste system, including the tritiated water storage tank (TWST), to support managing large volume/high tritium concentration RCS releases. The TWST has a capacity of 500,000 gallons, which is significantly more than the volume of the primary coolant. These tanks can be used for liquid effluent holdup, dilution, and timing of releases to ensure that the 10 CFR Part 20 effluent concentration limit values are met.

The current licensing basis analysis demonstrating the adequacy of the gaseous and liquid radwaste systems was updated to account for a 2,496 TPC in WBN Unit 1 and WBN Unit 2 by utilizing the radwaste system design basis source term described above.

Table 4.11-3 shows the result without radwaste system processing. Table 4.11-4 and Table 4.11-5 demonstrate that the liquid releases do not exceed the 10 CFR Part 20

Appendix B Table 2 limits. Table 4.1-5 shows the results when the calculated annual total quantity of radioactive material, except tritium and dissolved gases, is limited to five curies for each reactor at the WBN site.

TVA has calculated that the tritium release concentrations remain below 10 CFR 20 release concentration limits. The requirement for a minimum CTB dilution flow of 30,000 gpm for discharge of radioactivity into the CTB lines remains applicable for operation with 2,496 TPBARs.

Table 4.11-6 and Table 4.11-7 demonstrate that the gaseous design release concentrations are below the 10 CFR Part 20 Appendix B Table 2 limits. The designs of the gas and liquid radwaste systems meet the requirements of 10 CFR Part 20.

Table 4.11-3: Liquid Release, No Processing							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation Design Design C/ECL**	**Dual Unit Operation Design C/ECL**
Br-84	7.78E-04	2.49	1.93E-03	3.24E-11	4.00E-04	8.10E-08	1.62E-07
I-131	3.43E+00	53.88	1.85E+02	3.10E-06	1.00E-06	3.10E+00	6.19E+00
I-132	2.04E-01	3.98	8.12E-01	1.36E-08	1.00E-04	1.36E-04	2.72E-04
I-133	2.24E+00	27.21	6.10E+01	1.02E-06	7.00E-06	1.46E-01	2.92E-01
I-134	3.89E-02	1.63	6.34E-02	1.06E-09	4.00E-04	2.66E-06	5.31E-06
I-135	8.91E-01	7.94	7.08E+00	1.19E-07	3.00E-05	3.95E-03	7.91E-03
Rb-88	1.05E-02	17.96	1.89E-01	3.16E-09	4.00E-04	7.90E-06	1.58E-05
Cs-134	3.64E-01	41.78	1.52E+01	2.55E-07	9.00E-07	2.83E-01	5.66E-01
Cs-136	3.69E-02	169.68	6.26E+00	1.05E-07	6.00E-06	1.75E-02	3.50E-02
Cs-137	4.84E-01	157.73	7.63E+01	1.28E-06	1.00E-06	1.28E+00	2.56E+00
Na-24	4.22E-01	1.00	4.22E-01	7.07E-09	5.00E-05	1.41E-04	2.83E-04
Cr-51	2.52E-01	0.30	7.55E-02	1.27E-09	5.00E-04	2.53E-06	5.06E-06
Mn-54	1.38E-01	0.48	6.65E-02	1.11E-09	3.00E-05	3.71E-05	7.43E-05
Fe-55	1.10E-01	1.00	1.10E-01	1.84E-09	1.00E-04	1.84E-05	3.69E-05
Fe-59	2.64E-02	3.58	9.46E-02	1.59E-09	1.00E-05	1.59E-04	3.17E-04
Co-58	4.01E-01	5.53	2.22E+00	3.72E-08	2.00E-05	1.86E-03	3.72E-03
Co-60	6.26E-02	1.42	8.89E-02	1.49E-09	3.00E-06	4.97E-04	9.94E-04
Zn-65	4.24E-02	1.00	4.24E-02	7.11E-10	5.00E-06	1.42E-04	2.84E-04
Sr-89	1.14E-02	23.08	2.63E-01	4.41E-09	8.00E-06	5.51E-04	1.10E-03
Sr-90	1.03E-03	13.82	1.42E-02	2.39E-10	5.00E-07	4.77E-04	9.54E-04
Sr-91	4.82E-03	1.88	9.07E-03	1.52E-10	2.00E-05	7.60E-06	1.52E-05
Y-91m	2.81E-03	1.00	2.81E-03	4.71E-11	2.00E-03	2.35E-08	4.71E-08
Y-91	7.98E-04	1146.62	9.15E-01	1.53E-08	8.00E-06	1.92E-03	3.83E-03

Table 4.11-3: Liquid Release, No Processing							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation Design C/ECL**	**Dual Unit Operation Design C/ECL**
Y-93	2.22E-02	1.00	2.22E-02	3.72E-10	2.00E-05	1.86E-05	3.72E-05
Zr-95	3.31E-02	1.75	5.81E-02	9.73E-10	2.00E-05	4.87E-05	9.73E-05
Nb-95	2.56E-02	2.41	6.18E-02	1.04E-09	3.00E-05	3.45E-05	6.90E-05
Mo-99	2.56E-01	803.03	2.06E+02	3.45E-06	2.00E-05	1.72E-01	3.45E-01
Tc-99m	2.27E-01	1.00	2.27E-01	3.80E-09	1.00E-03	3.80E-06	7.61E-06
Ru-103	6.08E-01	1.00	6.08E-01	1.02E-08	3.00E-05	3.40E-04	6.79E-04
Ru-106	7.54E+00	1.00	7.54E+00	1.26E-07	3.00E-06	4.21E-02	8.42E-02
Te-129m	1.51E-02	1.00	1.51E-02	2.53E-10	7.00E-06	3.62E-05	7.23E-05
Te-129	1.35E-02	1.00	1.35E-02	2.26E-10	4.00E-04	5.66E-07	1.13E-06
Te-131m	3.05E-02	1.00	3.05E-02	5.11E-10	8.00E-06	6.39E-05	1.28E-04
Te-131	5.71E-03	1.00	5.71E-03	9.57E-11	8.00E-05	1.20E-06	1.39E-06
Te-132	7.48E-02	148.57	1.11E+01	1.86E-07	9.00E-06	2.07E-02	4.14E-02
Ba-140	9.07E-01	0.32	2.93E-01	4.91E-09	8.00E-06	6.14E-04	1.23E-03
La-140	1.26E+00	0.06	7.30E-02	1.22E-09	9.00E-06	1.36E-04	2.72E-04
Ce-141	1.20E-02	1.00	1.20E-02	2.01E-10	3.00E-05	6.70E-06	1.34E-05
Ce-143	6.17E-02	1.00	6.17E-02	1.03E-09	2.00E-05	5.17E-05	1.03E-04
Ce-144	3.33E-01	0.08	2.77E-02	4.64E-10	3.00E-06	1.55E-04	3.09E-04
Np-239	7.83E-02	1.00	7.83E-02	1.31E-09	2.00E-05	6.56E-05	1.31E-04
H-3	1252.80	1	1252.80	2.10E-05	1.00E-03	2.10E-02	4.20E-02
H-3 (TPC)	12484.8	1	12484.8	2.09E-04	1.00E-03	2.09E-01	4.18E-01
Total						**5.09E+00**	**1.02E+01**
Total (TPC)						**5.28E+00**	**1.06E+01**

Table 4.11-4: Liquid Release, Mobile Demineralizer Processing							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation Design C/ECL**	**Dual Unit Operation Design C/ECL**
Br-84	2.30E-04	2.49	5.72E-04	9.58E-12	4.00E-04	2.40E-08	4.79E-08
I-131	7.43E-02	53.88	4.00E+00	6.71E-08	1.00E-06	6.71E-02	1.34E-01
I-132	2.14E-02	3.98	8.52E-02	1.43E-09	1.00E-04	1.43E-05	2.86E-05
I-133	1.29E-01	27.21	3.51E+00	5.88E-08	7.00E-06	8.40E-03	1.68E-02
I-134	8.93E-03	1.63	1.46E-02	2.44E-10	4.00E-04	6.10E-07	1.22E-06
I-135	9.27E-02	7.94	7.36E-01	1.23E-08	3.00E-05	4.11E-04	8.23E-04
Rb-88	9.49E-03	17.96	1.70E-01	2.86E-09	4.00E-04	7.14E-06	1.43E-05
Cs-134	3.99E-02	41.78	1.67E+00	2.79E-08	9.00E-07	3.10E-02	6.21E-02
Cs-136	3.59E-03	169.68	6.09E-01	1.02E-08	6.00E-06	1.70E-03	3.40E-03
Cs-137	5.47E-02	157.73	8.63E+00	1.45E-07	1.00E-06	1.45E-01	2.89E-01
Na-24	3.48E-02	1.00	3.48E-02	5.83E-10	5.00E-05	1.17E-05	2.33E-05
Cr-51	1.12E-02	0.30	3.36E-03	5.62E-11	5.00E-04	1.12E-07	2.25E-07
Mn-54	7.45E-03	0.48	3.59E-03	6.01E-11	3.00E-05	2.00E-06	4.01E-06
Fe-55	1.10E-02	1.00	1.10E-02	1.84E-10	1.00E-04	1.84E-06	3.69E-06
Fe-59	3.24E-03	3.58	1.16E-02	1.95E-10	1.00E-05	1.95E-05	3.89E-05
Co-58	3.40E-02	5.53	1.88E-01	3.15E-09	2.00E-05	1.58E-04	3.15E-04
Co-60	1.83E-02	1.42	2.60E-02	4.36E-10	3.00E-06	1.45E-04	2.90E-04
Zn-65	8.77E-04	1.00	8.77E-04	1.47E-11	5.00E-06	2.94E-06	5.88E-06
Sr-89	3.46E-04	23.08	7.98E-03	1.34E-10	8.00E-06	1.67E-05	3.35E-05
Sr-90	3.68E-05	13.82	5.09E-04	8.52E-12	5.00E-07	1.70E-05	3.41E-05
Sr-91	5.00E-04	1.88	9.41E-04	1.58E-11	2.00E-05	7.88E-07	1.58E-06
Y-91m	2.94E-04	1.00	2.94E-04	4.93E-12	2.00E-03	2.46E-09	4.93E-09
Y-91	1.17E-04	1146.62	1.34E-01	2.25E-09	8.00E-06	2.81E-04	5.62E-04

Table 4.11-4: Liquid Release, Mobile Demineralizer Processing							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation Design C/ECL**	**Dual Unit Operation Design C/ECL**
Y-93	2.26E-03	1.00	2.26E-03	3.79E-11	2.00E-05	1.89E-06	3.79E-06
Zr-95	2.03E-03	1.75	3.56E-03	5.97E-11	2.00E-05	2.98E-06	5.97E-06
Nb-95	2.83E-03	2.41	6.83E-03	1.14E-10	3.00E-05	3.81E-06	7.63E-06
Mo-99	8.95E-03	803.03	7.19E+00	1.20E-07	2.00E-05	6.02E-03	1.20E-02
Tc-99m	6.80E-03	1.00	6.80E-03	1.14E-10	1.00E-03	1.14E-07	2.28E-07
Ru-103	1.32E-02	1.00	1.32E-02	2.21E-10	3.00E-05	7.37E-06	1.47E-05
Ru-106	1.66E-01	1.00	1.66E-01	2.78E-09	3.00E-06	9.27E-04	1.85E-03
Te-129m	3.22E-04	1.00	3.22E-04	5.40E-12	7.00E-06	7.71E-07	1.54E-06
Te-129	1.10E-03	1.00	1.10E-03	1.84E-11	4.00E-04	4.61E-08	9.22E-08
Te-131m	1.63E-03	1.00	1.63E-03	2.73E-11	8.00E-06	3.41E-06	6.83E-06
Te-131	3.71E-04	1.00	3.71E-04	6.22E-12	8.00E-05	7.77E-08	1.55E-07
Te-132	2.41E-03	148.57	3.58E-01	6.00E-09	9.00E-06	6.67E-04	1.33E-03
Ba-140	2.22E-02	0.32	7.18E-03	1.20E-10	8.00E-06	1.50E-05	3.01E-05
La-140	3.44E-02	0.06	1.99E-03	3.34E-11	9.00E-06	3.71E-06	7.42E-06
Ce-141	5.38E-04	1.00	5.38E-04	9.02E-12	3.00E-05	3.01E-07	6.01E-07
Ce-143	3.11E-03	1.00	3.11E-03	5.21E-11	2.00E-05	2.61E-06	5.21E-06
Ce-144	1.16E-02	0.08	9.64E-04	1.62E-11	3.00E-06	5.39E-06	1.08E-05
Np-239	2.91E-03	1.00	2.91E-03	4.88E-11	2.00E-05	2.44E-06	4.88E-06
H-3	1252.80	1	1252.80	2.10E-05	1.00E-03	2.10E-02	4.20E-02
H-3(TPC)	12484.8	1	12484.8	2.10E-05	1.00E-03	2.10E-01	4.18E-01
Total						**2.83E-01**	**5.65E-01**
Total (TPC)						**4.71E-01**	**9.42E-01**

Table 4.11-5: Direct SGBD Release/SGBD at Maximum Appendix I with 30,000 gpm CTB Dilution

Nuclide	LRW (Ci/year)	SGB Ci/year Scaled to 4.18 Ci	Des/Exp Ratio	Des (Ci/year)	Liquid (µCi/cc)	Liquid 10CFR20 ECL (µCi/cc)	Single Unit Operation C/ECL	Dual Unit Operation C/ECL
Br-84	2.30E-04	7.55E-03	2.49	8.12E-03	1.35E-10	4.00E-04	3.37E-07	6.75E-07
I-131	7.09E-02	3.13E-01	53.88	4.13E+00	6.87E-08	1.00E-06	6.87E-02	1.37E-01
I-132	2.12E-02	3.56E-01	3.98	4.40E-01	7.32E-09	1.00E-04	7.32E-05	1.46E-04
I-133	1.26E-01	7.48E-01	27.21	4.18E+00	6.94E-08	7.00E-06	9.91E-03	1.98E-02
I-134	8.90E-03	2.50E-01	1.63	2.65E-01	4.39E-09	4.00E-04	1.10E-05	2.20E-05
I-135	9.19E-02	8.83E-01	7.94	1.61E+00	2.68E-08	3.00E-05	8.93E-04	1.79E-03
Rb-88	9.49E-03	5.19E-02	17.96	2.22E-01	3.69E-09	4.00E-04	9.24E-06	1.85E-05
Cs-134	3.96E-02	5.06E-02	41.78	1.71E+00	2.83E-08	9.00E-07	3.15E-02	6.30E-02
Cs-136	3.55E-03	6.11E-03	169.68	6.08E-01	1.01E-08	6.00E-06	1.68E-03	3.37E-03
Cs-137	5.43E-02	6.74E-02	157.73	8.63E+00	1.43E-07	1.00E-06	1.43E-01	2.87E-01
Na-24	3.44E-02	2.25E-01	1.00	2.59E-01	4.31E-09	5.00E-05	8.62E-05	1.72E-04
Cr-51	1.09E-02	2.27E-02	0.30	2.60E-02	4.31E-10	5.00E-04	8.63E-07	1.73E-06
Mn-54	7.32E-03	1.14E-02	0.48	1.49E-02	2.48E-10	3.00E-05	8.27E-06	1.65E-05
Fe-55	1.09E-02	8.62E-03	1.00	1.95E-02	3.24E-10	1.00E-04	3.24E-06	6.49E-06
Fe-59	3.21E-03	2.10E-03	3.58	1.36E-02	2.26E-10	1.00E-05	2.26E-05	4.52E-05
Co-58	3.03E-02	3.33E-02	5.53	2.01E-01	3.34E-09	2.00E-05	1.67E-04	3.34E-04
Co-60	1.83E-02	3.86E-03	1.42	2.99E-02	4.96E-10	3.00E-06	1.65E-04	3.31E-04
Zn-65	8.36E-04	3.69E-03	1.00	4.53E-03	7.52E-11	5.00E-06	1.50E-05	3.01E-05
Sr-89	3.35E-04	9.97E-04	23.08	8.73E-03	1.45E-10	8.00E-06	1.81E-05	3.63E-05
Sr-90	3.58E-05	8.62E-05	13.82	5.81E-04	9.65E-12	5.00E-07	1.93E-05	3.86E-05
Sr-91	4.96E-04	3.95E-03	1.88	4.88E-03	8.11E-11	2.00E-05	4.06E-06	8.11E-06
Y-91m	2.91E-04	3.31E-04	1.00	6.22E-04	1.03E-11	2.00E-03	5.17E-09	1.03E-08
Y-91	1.17E-04	3.68E-05	1146.62	1.34E-01	2.23E-09	8.00E-06	2.79E-04	5.57E-04

Table 4.11-5: Direct SGBD Release/SGBD at Maximum Appendix I with 30,000 gpm CTB Dilution								
Nuclide	**LRW (Ci/year)**	**SGB Ci/year Scaled to 4.18 Ci**	**Des/Exp Ratio**	**Des (Ci/year)**	**Liquid (µCi/cc)**	**Liquid 10CFR20 ECL (µCi/cc)**	**Single Unit Operation C/ECL**	**Dual Unit Operation C/ECL**
Y-93	2.24E-03	1.71E-02	1.00	1.93E-02	3.21E-10	2.00E-05	1.61E-05	3.21E-05
Zr-95	2.00E-03	2.80E-03	1.75	6.31E-03	1.05E-10	2.00E-05	5.24E-06	1.05E-05
Nb-95	2.81E-03	1.93E-03	2.41	8.71E-03	1.45E-10	3.00E-05	4.82E-06	9.65E-06
Mo-99	8.70E-03	4.20E-02	803.03	7.03E+00	1.17E-07	2.00E-05	5.84E-03	1.17E-02
Tc-99m	6.58E-03	1.45E-02	1.00	2.11E-02	3.50E-10	1.00E-03	3.50E-07	7.00E-07
Ru-103	1.26E-02	5.42E-02	1.00	6.68E-02	1.11E-09	3.00E-05	3.70E-05	7.40E-05
Ru-106	1.58E-01	6.49E-01	1.00	8.07E-01	1.34E-08	3.00E-06	4.47E-03	8.94E-03
Te-129m	3.08E-04	1.37E-03	1.00	1.68E-03	2.79E-11	7.00E-06	3.98E-06	7.97E-06
Te-129	1.08E-03	2.33E-02	1.00	2.44E-02	4.05E-10	4.00E-04	1.01E-06	2.03E-06
Te-131m	1.60E-03	8.69E-03	1.00	1.03E-02	1.71E-10	8.00E-06	2.14E-05	4.27E-05
Te-131	3.66E-04	2.89E-03	1.00	3.26E-03	5.41E-11	8.00E-05	6.76E-07	1.35E-06
Te-132	2.34E-03	1.12E-02	148.57	3.59E-01	5.96E-09	9.00E-06	6.62E-04	1.32E-03
Ba-140	2.13E-02	9.03E-02	0.32	9.72E-02	1.61E-09	8.00E-06	2.02E-04	4.04E-04
La-140	3.32E-02	1.52E-01	0.06	1.54E-01	2.56E-09	9.00E-06	2.84E-04	5.68E-04
Ce-141	5.26E-04	1.06E-03	1.00	1.59E-03	2.64E-11	3.00E-05	8.78E-07	1.76E-06
Ce-143	3.05E-03	1.62E-02	1.00	1.93E-02	3.20E-10	2.00E-05	1.60E-05	3.20E-05
Ce-144	1.13E-02	2.81E-02	0.08	2.90E-02	4.82E-10	3.00E-06	1.61E-04	3.22E-04
Np-239	2.84E-03	1.40E-02	1.00	1.68E-02	2.80E-10	2.00E-05	1.40E-05	2.80E-05
H-3	1252.80		1	1252.80	2.08E-05	1.00E-03	2.08E-02	4.16E-02
H-3(TPC)	12484.8		1	12484.8	2.07E-04	1.00E-03	2.07E-01	4.15E-01
Total							**2.89E-01**	**5.79E-01**
Total (TPC)							**4.76E-01**	**9.52E-01**

Table 4.11-6: Gaseous Releases, Containment Purge Option							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation C/ECL**	**Dual Unit Operation C/ECL**
Kr-85m	2.92E+01	1.21E+01	3.54E+02	1.61E-10	1.00E-07	1.61E-03	3.21E-03
Kr-85	7.01E+02	3.40E+01	2.38E+04	1.08E-08	7.00E-07	1.54E-02	3.09E-02
Kr-87	1.84E+01	7.36E+00	1.35E+02	6.14E-11	2.00E-08	3.07E-03	6.14E-03
Kr-88	4.36E+01	1.22E+01	5.32E+02	2.41E-10	9.00E-09	2.68E-02	5.36E-02
Xe-131m	1.29E+03	2.92E+00	3.76E+03	1.71E-09	2.00E-06	8.54E-04	1.71E-03
Xe-133m	5.51E+01	4.29E+01	2.36E+03	1.07E-09	6.00E-07	1.79E-03	3.57E-03
Xe-133	3.56E+03	1.11E+02	3.94E+05	1.79E-07	5.00E-07	3.57E-01	7.14E-01
Xe-135m	9.62E+00	4.96E+00	4.78E+01	2.17E-11	4.00E-08	5.41E-04	1.08E-03
Xe-135	2.10E+02	6.89E+00	1.45E+03	6.56E-10	7.00E-08	9.37E-03	1.87E-02
Xe-138	8.59E+00	5.38E+00	4.63E+01	2.10E-11	2.00E-08	1.05E-03	2.10E-03
Br-84	5.60E-02	2.49E+00	1.39E-01	6.31E-14	8.00E-08	7.89E-07	1.58E-06
I-131	1.70E-01	5.39E+01	9.16E+00	4.15E-12	2.00E-10	2.08E-02	4.15E-02
I-132	7.40E-01	3.98E+00	2.95E+00	1.34E-12	2.00E-08	6.68E-05	1.34E-04
I-133	5.00E-01	2.72E+01	1.36E+01	6.17E-12	1.00E-09	6.17E-03	1.23E-02
I-134	1.19E+00	1.63E+00	1.94E+00	8.80E-13	6.00E-08	1.47E-05	2.93E-05
I-135	9.21E-01	7.94E+00	7.31E+00	3.32E-12	6.00E-09	5.53E-04	1.11E-03
Cs-134	2.27E-03	4.18E+01	9.48E-02	4.30E-14	2.00E-10	2.15E-04	4.30E-04
Cs-136	8.01E-05	1.70E+02	1.36E-02	6.16E-15	9.00E-10	6.85E-06	1.37E-05
Cs-137	3.48E-03	1.58E+02	5.49E-01	2.49E-13	2.00E-10	1.24E-03	2.49E-03
Cr-51	5.92E-04	3.00E-01	1.77E-04	8.04E-17	3.00E-08	2.68E-09	5.36E-09
Mn-54	4.31E-04	4.82E-01	2.08E-04	9.41E-17	1.00E-09	9.41E-08	1.88E-07
Fe-59	7.70E-05	3.58E+00	2.76E-04	1.25E-16	5.00E-10	2.50E-07	5.00E-07
Co-58	2.32E-02	5.53E+00	1.28E-01	5.82E-14	1.00E-09	5.82E-05	1.16E-04
Co-60	8.74E-03	1.42E+00	1.24E-02	5.63E-15	5.00E-11	1.13E-04	2.25E-04

Table 4.11-6: Gaseous Releases, Containment Purge Option							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation C/ECL**	**Dual Unit Operation C/ECL**
Sr-89	2.98E-03	2.31E+01	6.88E-02	3.12E-14	1.00E-09	3.12E-05	6.24E-05
Sr-90	1.14E-03	1.38E+01	1.58E-02	7.14E-15	6.00E-12	1.19E-03	2.38E-03
Zr-95	1.00E-03	1.75E+00	1.75E-03	7.96E-16	4.00E-10	1.99E-06	3.98E-06
Nb-95	2.45E-03	2.41E+00	5.91E-03	2.68E-15	2.00E-09	1.34E-06	2.68E-06
Ba-140	4.00E-04	3.23E-01	1.29E-04	5.86E-17	2.00E-09	2.93E-08	5.86E-08
H-3	1.39E+02	1.00E+00	1.39E+02	6.30E-11	1.00E-07	6.30E-04	1.26E-03
H-3 (TPC)	1.39E+03	1.00E+00	1.39E+03	6.29E-10	1.00E-07	6.29E-03	1.26E-02
C-14	1.12E+01	1.00E+00	1.12E+01	5.08E-12	3.00E-09	1.69E-03	3.39E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.54E-11	1.00E-08	1.54E-03	3.08E-03
Total						**4.52E-01**	**9.04E-01**
Total (TPC)						**4.58E-01**	**9.15E-01**

Table 4.11-7: Gaseous Releases, Continuous Filtered Containment Vent Option							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation C/ECL**	**Dual Unit Operation C/ECL**
Kr-85m	1.16E+01	1.21E+01	1.41E+02	6.38E-11	1.00E-07	6.38E-04	1.28E-03
Kr-85	6.79E+02	3.40E+01	2.31E+04	1.05E-08	7.00E-07	1.49E-02	2.99E-02
Kr-87	6.55E+00	7.36E+00	4.82E+01	2.19E-11	2.00E-08	1.09E-03	2.19E-03
Kr-88	1.55E+01	1.22E+01	1.89E+02	8.58E-11	9.00E-09	9.54E-03	1.91E-02
Xe-131m	1.19E+03	2.92E+00	3.47E+03	1.57E-09	2.00E-06	7.87E-04	1.57E-03
Xe-133m	5.05E+01	4.29E+01	2.17E+03	9.82E-10	6.00E-07	1.64E-03	3.27E-03
Xe-133	3.27E+03	1.11E+02	3.62E+05	1.64E-07	5.00E-07	3.28E-01	6.56E-01
Xe-135m	5.23E+00	4.96E+00	2.60E+01	1.18E-11	4.00E-08	2.94E-04	5.89E-04
Xe-135	1.11E+02	6.89E+00	7.64E+02	3.47E-10	7.00E-08	4.95E-03	9.90E-03
Xe-138	4.82E+00	5.38E+00	2.60E+01	1.18E-11	2.00E-08	5.88E-04	1.18E-03
Br-84	5.60E-02	2.49E+00	1.39E-01	6.31E-14	8.00E-08	7.89E-07	1.58E-06
I-131	1.70E-01	5.39E+01	9.16E+00	4.15E-12	2.00E-10	2.08E-02	4.15E-02
I-132	7.38E-01	3.98E+00	2.94E+00	1.33E-12	2.00E-08	6.66E-05	1.33E-04
I-133	4.99E-01	2.72E+01	1.36E+01	6.16E-12	1.00E-09	6.16E-03	1.23E-02
I-134	1.19E+00	1.63E+00	1.94E+00	8.80E-13	6.00E-08	1.47E-05	2.93E-05
I-135	9.19E-01	7.94E+00	7.30E+00	3.31E-12	6.00E-09	5.52E-04	1.10E-03
Cs-134	2.27E-03	4.18E+01	9.48E-02	4.30E-14	2.00E-10	2.15E-04	4.30E-04
Cs-136	8.01E-05	1.70E+02	1.36E-02	6.16E-15	9.00E-10	6.85E-06	1.37E-05
Cs-137	3.48E-03	1.58E+02	5.49E-01	2.49E-13	2.00E-10	1.24E-03	2.49E-03
Cr-51	5.92E-04	3.00E-01	1.77E-04	8.04E-17	3.00E-08	2.68E-09	5.36E-09
Mn-54	4.31E-04	4.82E-01	2.08E-04	9.41E-17	1.00E-09	9.41E-08	1.88E-07
Fe-59	7.70E-05	3.58E+00	2.76E-04	1.25E-16	5.00E-10	2.50E-07	5.00E-07
Co-58	2.32E-02	5.53E+00	1.28E-01	5.82E-14	1.00E-09	5.82E-05	1.16E-04

Table 4.11-7: Gaseous Releases, Continuous Filtered Containment Vent Option							
Nuclide	**Expected Release (Ci/year)**	**Des/Exp Ratio**	**Design (Ci/year)**	**Design (µCi/cc)**	**10CFR20 (ECL, µCi/cc)**	**Single Unit Operation C/ECL**	**Dual Unit Operation C/ECL**
Co-60	8.74E-03	1.42E+00	1.24E-02	5.63E-15	5.00E-11	1.13E-04	2.25E-04
Sr-89	2.98E-02	2.31E+01	6.88E-02	3.12E-14	1.00E-09	3.12E-05	6.24E-05
Sr-90	1.14E-03	1.38E+01	1.58E-02	7.14E-15	6.00E-12	1.19E-03	2.38E-03
Zr-95	1.00E-03	1.75E+00	1.75E-03	7.96E-16	4.00E-10	1.99E-06	3.98E-06
Nb-95	2.45E-03	2.41E+00	5.91E-03	2.68E-15	2.00E-09	1.34E-06	2.68E-06
Ba-140	4.00E-04	3.23E-01	1.29E-04	5.86E-17	2.00E-09	2.93E-08	5.86E-08
H-3	1.39E+02	1.00E+00	1.39E+02	6.30E-11	1.00E-07	6.30E-04	1.26E-03
H-3 (TPC)	1.39E+03	1.00E+00	1.39E+03	6.29E-10	1.00E-07	6.29E-03	1.26E-02
C-14	1.12E+01	1.00E+00	1.12E+01	5.08E-12	3.00E-09	1.69E-03	3.39E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.54E-11	1.00E-08	1.54E-03	3.08E-03
Total						**3.97E-01**	**7.94E-01**
Total (TPC)						**4.02E-01**	**8.05E-01**

Tritium Impacts on Public Dose During Normal Operation

Using the realistic TPC source terms for 2,496 TPBARs, the annual releases were reanalyzed. The other parameters remain the same except that the 2021 Land Use Survey data was used in calculating Non-TPC and TPC doses, which resulted in the maximally exposed individual organ changing and some doses decreasing when compared to References 4.11-4 and 4.11-11. The liquid annual releases are summarized in Table 4.11-8. The gaseous releases are summarized in Table 4.11-9.

These annual releases were then used to determine the offsite doses for releases of radionuclides in liquid and gaseous effluents from a single unit during normal operation and are summarized in Table 4.11-10. This table also lists the WBN regulatory established radioactive effluent guidelines and the estimated non-TPC values.

Table 4.11-8
Annual Discharge of the Liquid Waste Processing System (per Unit)

Isotope	LRW (no SGB) (Ci)	SGB with no CD process (Ci)	Total (Ci)
Br-84	2.30E-04	7.55E-03	7.77E-03
I-131	7.09E-02	3.13E-01	3.84E-01
I-132	2.12E-02	3.56E-01	3.77E-01
I-133	1.26E-01	7.48E-01	8.74E-01
I-134	8.90E-03	2.50E-01	2.59E-01
I-135	9.19E-02	8.83E-01	9.74E-01
Rb-88	9.49E-03	5.19E-02	6.14E-02
Cs-134	3.96E-02	5.06E-02	9.02E-02
Cs-136	3.55E-03	6.11E-03	9.66E-03
Cs-137	5.43E-02	6.74E-02	1.22E-01
Na-24	3.44E-02	2.25E-01	2.59E-01
Cr-51	1.09E-02	2.27E-02	3.36E-02
Mn-54	7.32E-03	1.14E-02	1.87E-02
Fe-55	1.09E-02	8.62E-03	1.95E-02
Fe-59	3.21E-03	2.10E-03	5.32E-03
Co-58	3.03E-02	3.33E-02	6.36E-02

Table 4.11-8 Annual Discharge of the Liquid Waste Processing System (per Unit)			
Isotope	**LRW (no SGB) (Ci)**	**SGB with no CD process (Ci)**	**Total (Ci)**
Co-60	1.83E-02	3.86E-03	2.22E-02
Zn-65	8.36E-04	3.69E-03	4.52E-03
Sr-89	3.35E-04	9.97E-04	1.33E-03
Sr-90	3.58E-05	8.62E-05	1.22E-04
Sr-91	4.96E-04	3.95E-03	4.45E-03
Y-91m	2.91E-04	3.31E-04	6.23E-04
Y-91	1.17E-04	3.68E-05	1.53E-04
Y-93	2.24E-03	1.71E-02	1.93E-02
Zr-95	2.00E-03	2.80E-03	4.80E-03
Nb-95	2.81E-03	1.93E-03	4.73E-03
Mo-99	8.70E-03	4.20E-02	5.07E-02
Tc-99m	6.58E-03	1.45E-02	2.11E-02
Ru-103	1.26E-02	5.42E-02	6.68E-02
Ru-106	1.58E-01	6.49E-01	8.08E-01
Te-129m	3.08E-04	1.37E-03	1.68E-03
Te-129	1.08E-03	2.33E-02	2.44E-02
Te-131m	1.60E-03	8.69E-03	1.03E-02
Te-131	3.66E-04	2.89E-03	3.26E-03
Te-132	2.34E-03	1.12E-02	1.35E-02
Ba-140	2.13E-02	9.03E-02	1.12E-01
La-140	3.32E-02	1.52E-01	1.85E-01
Ce-141	5.26E-04	1.06E-03	1.59E-03
Ce-143	3.05E-03	1.62E-02	1.92E-02

Table 4.11-8 Annual Discharge of the Liquid Waste Processing System (per Unit)			
Isotope	**LRW (no SGB) (Ci)**	**SGB with no CD process (Ci)**	**Total (Ci)**
Ce-144	1.13E-02	2.81E-02	3.94E-02
Np-239	2.84E-03	1.40E-02	1.69E-02
H-3	1252.8		1252.8
H-3 (TPC)	12484.8		12484.8
Total w/o H-3			**5.0**
Total w/ H-3			**1257.8**
Total w/(TPC)			**12489.8**

Table 4.11-9 Expected Annual Releases From the Gaseous Waste Process System with Continuous Filtered Containment Vent				
Nuclide	**Containment Building (Ci)**	**Aux. Building (Ci)**	**Turbine Building (Ci)**	**Total (Ci)**
Kr85m	5.16E+00	5.04E+00	1.37E+00	1.16E+01
Kr85	6.71E+02	6.86E+00	1.81E+00	6.79E+02
Kr87	5.99E-01	4.74E+00	1.21E+00	6.55E+00
Kr88	4.29E+00	8.82E+00	2.38E+00	1.55E+01
Xe131m	1.17E+03	1.83E+01	4.80E+00	1.19E+03
Xe133m	4.78E+01	2.09E+00	5.71E-01	5.05E+01
Xe133	3.17E+03	7.23E+01	1.91E+01	3.27E+03
Xe135m	2.74E-02	4.11E+00	1.09E+00	5.23E+00
Xe135	7.77E+01	2.65E+01	7.20E+00	1.11E+02
Xe137	4.38E-04	1.08E+00	2.87E-01	1.36E+00
Xe138	2.03E-02	3.79E+00	1.01E+00	4.82E+00
Ar41	3.40E+01	0.00E+00	0.00E+00	3.40E+01
Br84	1.01E-06	5.54E-02	5.69E-04	5.60E-02
I131	1.01E-02	1.36E-01	2.36E-02	1.70E-01
I132	1.81E-04	7.11E-01	2.68E-02	7.38E-01
I133	2.69E-03	4.40E-01	5.64E-02	4.99E-01
I134	5.45E-05	1.17E+00	1.88E-02	1.19E+00
I135	1.13E-03	8.51E-01	6.66E-02	9.19E-01
H3	1.39E+02	0.00E+00	0.00E+00	1.39E+02
H3 (TPC)	1.39E+03	0.00E+00	0.00E+00	1.39E+03
Cr51	9.21E-05	5.00E-04	0.00E+00	5.92E-04

Table 4.11-9
Expected Annual Releases From the Gaseous Waste Process System with Continuous Filtered Containment Vent

Nuclide	Containment Building (Ci)	Aux. Building (Ci)	Turbine Building (Ci)	Total (Ci)
Mn54	5.30E-05	3.78E-04	0.00E+00	4.31E-04
Co57	8.20E-06	0.00E+00	0.00E+00	8.20E-06
Co58	2.50E-04	2.29E-02	0.00E+00	2.32E-02
Co60	2.61E-05	8.71E-03	0.00E+00	8.74E-03
Fe59	2.70E-05	5.00E-05	0.00E+00	7.70E-05
Sr89	1.30E-04	2.85E-03	0.00E+00	2.98E-03
Sr90	5.22E-05	1.09E-03	0.00E+00	1.14E-03
Zr95	4.80E-08	1.00E-03	0.00E+00	1.00E-03
Nb95	1.80E-05	2.43E-03	0.00E+00	2.45E-03
Ru103	1.60E-05	6.10E-05	0.00E+00	7.70E-05
Ru106	2.70E-08	7.50E-05	0.00E+00	7.50E-05
Sb125	0.00E+00	6.09E-05	0.00E+00	6.09E-05
Cs134	2.53E-05	2.24E-03	0.00E+00	2.27E-03
Cs136	3.21E-05	4.80E-05	0.00E+00	8.01E-05
Cs137	5.58E-05	3.42E-03	0.00E+00	3.48E-03
Ba140	2.30E-07	4.00E-04	0.00E+00	4.00E-04
Ce141	1.30E-05	2.64E-05	0.00E+00	3.95E-05
C14*	4.30E+00	6.90E+00	0.00E+00	1.12E+01

*Carbon-14 production and gaseous effluent source term estimates were based on the Electric Power Research Institute (EPRI) methodology provided in EPRI Report 1021106, "Estimation of Carbon-14 in Nuclear Power Plant Gaseous Effluents," dated December 2010. The Carbon-14 production assumed for 365 EFPD has been determined to be 11.2 Ci; however, only 98 percent is considered released as gas and only the carbon dioxide form (i.e., 20 percent) of that is used in the gaseous dose calculations.

Table 4.11-10
Annual Projected Impact of TPC (2,496 TPBARs) on Effluent Dose to Maximally Exposed Members of the Public and Population Dose per Unit

Category	Non-TPC Dose	TPC Dose	Incremental Increase from TPC	NRC Annual Effluent Exposure Guideline
Annual Radioactive Gaseous Emissions				
Maximally Exposed Individual (mrem) Total Body	0.60	0.60	0	5.00 Total Body
Maximally Exposed Individual (mrem) Organ	8.85 (Bone)	11.3 (Bone)	2.45	15.00 Any Organ
50-mile Population Dose (person-rem)	11.3 (Thyroid)	19.1 (Thyroid)	7.8	NA
Annual Radioactive Liquid Emissions				
Maximally Exposed Individual (mrem) Total Body	0.34	0.39	0.05	3.00 Total Body
Maximally Exposed Individual (mrem) Organ	0.47 (Liver)	0.50 (Liver)	0.03	10.00 Any Organ
50-mile Population Dose (person-rem)	6.9 (Thyroid)	13.0 (Thyroid)	6.1	NA

Table 4.11-10 demonstrates that the increase in the tritium reactor coolant activity and resultant environmental releases would result in a minor increase to the offsite doses, which continue to remain below the NRC's guidance levels.

Annual dose equivalent limits for the normal operation at WBN site are prescribed in 40 CFR 190.10, “Standards for normal operations,” part (a). Using the revised realistic TPC source terms for 2,496 TPBARs, the offsite doses calculated for releases of radionuclides in liquid and gaseous effluents from the site operating with two TPC cores during normal operation plus direct radiation are summarized in Table 4.11-11. This table also lists the regulatory established dose limits.

Table 4.11-11 Annual Projected Impact of Two TPCs (2,496 TPBARs) on 40 CFR Part 190 Compliance		
Organ	Site Dose from Two TPCs (mrem/yr)	40 CFR 190 Limit(mrem/yr)
Whole Body	12.24	25
Thyroid	16.52	75
Critical Organ (Bone)	23.44	25

Table 4.11-11 demonstrates that the resultant environmental releases from tritium production at the site meet the Environmental Protection Agency limits. The decrease in the thyroid dose from Table 4.1-36 of Reference 4.11-4 is a result of two dairy farms that are no longer in production.

Tritium Impacts on Occupational Dose During Normal Operation

Because of weepage through valve stems and pump shaft seals, some coolant escapes into the containment and the auxiliary buildings. A portion of the RCS leakage flashes to steam/evaporates, thus contributing to the tritiated water vapor source term, and a fraction remains as liquid, becoming part of the liquid source term. The relative amount of leakage entering the gaseous and liquid phases is dependent upon the temperature and pressure at the point where the leakage occurs. Ten percent due to flashing and SFP evaporative losses is the assumed gaseous effluent fraction for dose impact modeling (NUREG-0017, Revision 1), whereas WBN effluent history indicates an average of approximately 5%. As tritiated water vapor is not removed by filtration or ion exchange, it will be released as gaseous effluent to the environment. A breaker-to-breaker run will potentially produce the maximum RCS tritium concentration, WBN Unit 1 Cycles 11 and 12 with 544 TPBARs were estimated to peak at just less than 7.0 µCi/gm. With the assumption of routine boron control and 2,496 TPBARs at 5 Ci/TPBAR/year, the average RCS tritium concentration, without any TPBAR failures, was determined in Interface Issue 5 to be 15.5 µCi/gm (Section 4.4 to this enclosure).

There is a strong correlation between the RCS tritium concentration and the containment airborne tritium concentration (station tritium dose). Containment tritium derived air concentration (DAC) values are a function of the RCS tritium activity, the transfer of tritium from the RCS to the containment atmosphere (leak rate), and the turnover/dilution of the containment atmosphere through periodic and continuous containment venting and purging. Consistent with WBN Unit 1 License Amendment 40 and the associated LAR (References 4.11-9 and 4.11-10, respectively), site-specific data collected during extended non-TPC operating cycles [i.e., WBN Unit 1 Cycle 3 and Sequoyah Nuclear Plant (SQN) Unit 1 Cycle 10, breaker-to-breaker Non-TPC cycles] have provided useful data to estimate the effect from tritium production on TVA pressurized water reactor (PWR) station radiological conditions. The RCS maximum tritium levels noted during the extended operating cycles were approximately 2.5 µCi/gm with a cycle RCS tritium mean of approximately 1.0 µCi/gm. The extended cycle tritium peak RCS tritium value of approximately 2.5 µCi/gm resulted in a containment peak tritium DAC-fraction of <0.15 for both WBN and SQN. The extended cycle tritium average RCS tritium value of approximately 1.0 µCi/gm resulted in a containment average DAC-fraction of about 0.08.

TVA determined that with no modifications to the current boron-control feed and bleed methodologies (i.e., approximately 366,000-gallon cycle letdown), the RCS average tritium value will be approximately 15.5 µCi/gm at a permeation rate of 5 Ci/TPBAR/year. This mean value would indicate an estimated average containment tritium DAC-fraction of:

*0.08 DAC-fraction / 1 µCi/gm H3 * 15.5 µCi/gm = 1.24 DAC-Fraction*

The estimated containment average tritium DAC-fraction equates to an effective dose rate of:

*1.24 DAC-fraction * 2.5 mrem/DAC-hour = 3.1 mrem/hour*

Because the primary radiological significance of exposure to tritium is in the form of internal exposure, a potential hazard arises when personnel are exposed to open processes that have been wetted with tritiated liquids. TVA used the site-specific data collected during recent extended operating cycles to evaluate the additional committed effective dose equivalent from possible increased tritium airborne activity from this potential hazard. The effect on station occupational exposure due to increased tritium concentration in the RCS was estimated based on the historical committed effective dose equivalent (CEDE) reported to the NRC. Based on data in NUREG-0713 volumes 21 through 28 (1999 through 2006), the average collective CEDE for WBN was approximately two person-rem per year. Conservatively assuming that this collective CEDE was entirely due to tritium, the expected increase utilizing the design basis tritium source term would result in the following bounding increase in CEDE:

*2.0 person-rem/year * 15.5 µCi/gm /1 µCi/gm H3 = 31 person-rem/year*

In NUREG-0713 volumes 29 through 42 (2007 through 2020), WBN did not report any collective CEDE. Therefore, because tritium is only one of many isotopes that contributed to the reported CEDE and recent performance has shown a noticeable decline in CEDE, the above estimated increase in dose is extremely conservative; the actual CEDE is expected to be much less.

Additionally, TVA has estimated the occupational dose received due to fuel and TPBAR handling activities. TVA's current estimate of the TPBAR cycle work scope includes pre-cycle preparation activities, post cycle hardware removal and handling activities, TPBAR consolidation (including equipment setup and disassembly), shipping activities, and the processing, packaging, and shipping of the irradiated components. Based on actual dose accrual, the average dose for these activities is 0.46 mrem/TPBAR. The result was conservatively rounded up to 1 mrem/TPBAR. TVA estimates that for a 2,496 TPBAR core, this additional TEDE is approximately 1.7 rem/year (2.5 rem per TPC cycle) for TPBAR handling and consolidation activities.

Therefore, an additional 32.7 rem/year is estimated for the increase in airborne activity and for fuel and TPBAR handling activities. The WBN three-year collective TEDE per reactor listed in NUREG-0713 Volume 42 for years 2018 -2020 was 26.46 rem. An additional annual average 32.7 tritium rem would raise the TEDE total to 59.16 rem; a value that remains within the 149 rem assessment total.

<u>Solid Radioactive Waste</u>

For normal one-unit TPC operation, the additional solid waste will be the base plate and thimble plug assemblies that remain after consolidation. WBN will temporarily store these items on-site. Offsite shipment and ultimate disposal is assumed in accordance with agreements between TVA and DOE. The estimated irradiated components associated with a 2,496 TPC with a 108-feed equilibrium core results in 216 base plates and 96 thimble plugs. When adjusted to reflect measured dose rate from a base plate with 24 thimble plugs following 113-day decay adjusted to 180 days is 11,419 Ci per cycle (180 day post irradiation decay) or an average of 7,613 Ci per year. This represents an increase from the WBN estimated value of 1,800 Ci per year to approximately 9,413 Ci per year. This increased activity is associated with metal activation products. The estimated disposal volume of this additional solid waste is approximately 96 cubic feet per TPC operating cycle or an average of 64 cubic feet per year.

This additional volume is an insignificant increase in the WBN annual estimated solid waste, from 32,854 cubic feet per year to 32,918 cubic feet per year. WBN's current estimate of the TPBAR cycle work scope includes pre-cycle preparation activities, post cycle removal and handling activities, TPBAR consolidation (including equipment setup and disassembly) and shipping activities, and the processing, packaging, and shipping of the irradiated components for and estimated total of 1,200 man-hours in a 0.33 mrem/hour radiation field.

References 4.11-9 and 4.11-10 also included an evaluation with the failure of two TPBARs, which resulted in the need to perform more feed and bleed operations. Therefore, an increase in the amount of resin was evaluated. As discussed previously, the radwaste design basis does not include consideration of two TPBAR failures; therefore, no additional feed and bleed operations are expected for WBN Unit 1 and WBN Unit 2, and no additional resin is evaluated. Thus, the tritium production solid radioactive waste environmental impact is consistent with the Reference 4.11-9 impact assessment and results in an insignificant increase to the WBN Unit 1 and WBN Unit 2 non-TPC waste.

<u>Spent Fuel Generation and Storage</u>

WBN Unit 1 License Amendment 40 assessed the environmental impact from the storage of additional spent fuel associated with the production of 2,304 TPBARs. The number of additional fresh fuel bundles per cycle due to tritium production was set to approximately 20. The proposed license amendment establishes 2,496 as the maximum number of TPBARs per cycle. This level of TPBAR irradiation will normally require four to eight additional fresh fuel bundles per cycle.

References

4.11-1 NRC letter to 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)

4.11-2 NRC letter to 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)

4.11-3 TVA letter to NRC, CNL-15-001, "Application to Revise Technical Specification 4.2.1, ‘Fuel Assemblies,’ (WBN-TS-15-03),” dated March 31, 2015 (ML15098A446)

4.11-4 TVA letter to NRC, 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)

4.11-5 TVA letter to NRC, TVA-WBN-TS-07-01, “Watts Bar Nuclear Plant (WBN) Unit 1 - Technical Specification Change 07-01, Revision of Number of Tritium Producing Burnable Absorber Rods (TPBARS) in the Reactor Core,” dated April 25, 2007 (ML071210604)

4.11-6 NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," April 1985 (ML112720411)

4.11-7 NUREG-0847, Supplement 24, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2,” September 2011 (ML11277A148)

4.11-8 TVA letter to NRC, “Watts Bar Nuclear Plant — Request for Additional Information (RAI) Regarding Radiological Impact (TAC No. MB1884),” dated May 23, 2002 (ML021490139)

4.11-9 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.11-10 TVA letter to NRC, “Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015,” dated August 20, 2001 (ML012390106 and ML012390115)

4.11-11 TVA letter to NRC, CNL-18-051, “Correction to 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 April 9, 2018 (ML18100A953)

4.12 TPBAR Interface Issue 15: Process and Effluent Radiological Monitoring and Sampling System

NUREG-1672, Section 2.11.5, "In Section 2.11.6 of the TPC topical report, DOE states that the current process and effluent radiological monitoring instrumentation and sampling systems that are in place at the reference plant, as well as at other operating PWR plants, include the capability for monitoring the tritium levels within the plant and in plant effluent pathways, and are adequate for use when the plant is operated with a TPC. On the basis of its review, the staff agrees with DOE that the existing capability for radiation monitoring is adequate for tritium levels at the reference plant. In response to the staff's RAI dated October 15, 1998, DOE stated that the details of the laboratory instrumentation and sampling frequencies and locations are plant dependent. Therefore, a plant-specific assessment of the candidate plant for the TPC will be required to provide such information. The staff has identified this as an interface item that 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 TPBAR Interface Issue 15 information provided for the WBN Unit 1 and 2 Tritium Production Program increase to 2,496 TPBAR is based on the applicable WBN Unit 1 and 2 precedent documents (References 4.12-1 to 4.12-9). For the 2,496 TPBAR increase, 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 (References 4.12-10 through 4.12-12).

The information in Reference 4.12-9 regarding Interface Issue 15 remains applicable to this license amendment request.

<u>Process and Effluent Monitoring and Sampling</u>

TVA previously performed an evaluation of the production of tritium using 1,792 TPBARs in WBN Unit 1 and 2 and determined that no additional sampling points were needed beyond those presently required by the WBN TS during normal plant operations and refueling operations with a TPC (References 4.12-2 and 4.12-9).

References

4.12-1 NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 — Issuance of Amendment to Irradiate Up to 2304 Tritium-Producing Burnable Absorber Rods in the Reactor Core (TAC No. MB1884)," dated September 23, 2002 (ML022540925)

4.12-2 NRC letter to 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)

4.12-3 TVA letter to NRC, "Watts Bar Nuclear Plant (WBN) – Unit 1 – Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015," dated August 20, 2001 (ML012390106 and ML012390115)

4.12-4 TVA letter to NRC, CNL-15-001, "Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies,' (WBN-TS-15-03)," dated March 31, 2015 (ML15098A446)

4.12-5 TVA letter to NRC, CNL-15-093, "Response to NRC Request to Supplement Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies' (WBN-TS-15-03) - Radiological Protection and Radiological Consequences," dated June 15, 2015 (ML15167A359)

4.12-6 TVA letter to NRC, "Watts Bar Nuclear Plant - Responses to RAI Regarding Tritium Production -Interface Issues 14 and 15 (TAC No. MB1884)," dated December 7, 2001 (ML013520461)

4.12-7 TVA letter to NRC, CNL-15-216, "Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies' (WBN-TS-15-03) (TAC No. MF6050) - Response to NRC Request for Additional Information - Radiation Protection and Consequence Branch," dated December 22, 2015 (ML16054A661)

4.12-8 TVA letter to NRC, CNL-16-030, "Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies' (WBN-TS-15-03) (TAC No. MF6050) - Supplement to Response to NRC Request for Additional Information - Radiation Protection and Consequence Branch," dated February 22, 2016 (ML16053A513)

4.12-9 TVA letter to NRC, 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)

4.12-10 TVA letter to NRC, "Groundwater Protection - Data Collection Questionnaire for Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants," dated August 4, 2006 (ML062280598)

4.12-11 NEI 07-07, "Industry Ground Water Protection Initiative - Final Guidance Document," August 2007 (ML072610036)

4.12-12 TVA letter to NRC, "Watts Bar Nuclear Plant 2021 Annual Radioactive Effluent Release Report," dated April 29, 2022 (ML22119A005)

4.13 TPBAR Interface Issue 16: Use of LOCTA_JR Code for LOCA Analyses

The current analysis of record (AOR) for the small-break LOCA (SBLOCA) and LBLOCA events for WBN Bar Units 1 and 2 utilizes the 2016 Westinghouse FSLOCA Evaluation Model (EM) and is supported by a TPBAR structural integrity analysis based on the FSLOCA EM. These analyses have been approved by the NRC (References 4.13-1 and 4.13-2). Neither the analysis for nuclear fuel using the FSLOCA EM nor the TPBAR structural integrity analysis explicitly uses the LOCTA_JR code. Therefore, Interface Issue 16 is not applicable to these analyses.

References

4.13-1 TVA letter to NRC, CNL-21-010, “Correction of Application to Implement the FULL SPECTRUM™[1] LOCA (FSLOCA™[1]) 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)

4.13-2 NRC letter to 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 (EPID L-2020-LLA-0005),” dated February 26, 2021 (ML21034A166 and ML21034A169)

4.14 TPBAR Interface Issue 17: ATWS Analysis

NUREG-1672, Section 2.15.7, "The staff agrees with the partial ATWS analysis conducted and the results obtained by DOE. However, this concurrence pertains only to the TPC topical report. The staff concludes that licensees seeking to utilize a TPC must submit a plant-specific application containing a full ATWS analysis, conducted in accordance with NRC regulations and approved standards. The staff has identified this as an interface item that 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 information in Reference 4.14-1 regarding Interface Issue 17 remains applicable to this license amendment request except as provided below.

The effect of TPBARs on the moderator temperature coefficient (MTC) was initially studied in the DOE-sponsored TPC Topical Report (Reference 4.14-2). In order to address the plant-specific interface issue related to anticipated transient without scram (ATWS) for WBN Unit 1, TVA submitted a plant-specific analysis (Reference 4.14-3) which was approved by the NRC (Reference 4.14-4). The analysis included a comparison of the full power MTCs as a function of cycle length was performed for a representative TPC and a WBN core without TPBARs. This analysis came to two conclusions: (1) that cycle-to-cycle variations in MTC were small and (2) the variability was attributed to controllable causes like loading pattern differences, burnable absorber inventories, required cycle energy variations, and prior cycle operating histories. Therefore, because the moderator feedback for the TPC designs were shown to be comparable to the designs without TPBARs, the ATWS responses would be comparable as well. To implement TPBARs in WBN Unit 2, TVA submitted an evaluation of the plant-specific interface issue related to ATWS (Reference 4.14-1), which was approved by the NRC (Reference 4.14-5). The ATWS issue was also previously considered for WBN Unit 2, as documented in Section 15.3.6 of Reference 4.14-6.

The one discernable trend that was noted in Reference 4.14-3 was that at the beginning of the cycle the TPC would exhibit a more negative MTC. The reason for this is that fixed burnable absorbers, like TPBARs, serve as a source of negative reactivity which limits the amount of soluble boron required to be in the RCS to maintain criticality. Because a lower soluble boron concentration leads to a more negative moderator feedback, the use of additional TPBARs will further decrease the core MTC and will add additional safety margin to the ATWS overpressurization event.

References

4.14-1 TVA letter to NRC, 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)

4.14-2 NDP-98-181, Revision 1, "Tritium Production Core (TPC) Topical Report," Westinghouse Electric Company, dated February 8, 1999 (ML16077A093)

4.14-3 TVA letter to NRC, "Watts Bar Nuclear Plant - Tritium Production Program - Anticipated Transients Without Scram (ATWS)," dated September 29, 2000 (ML003759282)

4.14-4 NRC letter to TVA, "Sequoyah Units 1 and 2, And Watts Bar, Unit 1, Re: Tritium Production Program - NUREG-1672 Interface Issue 17 - Anticipated Transient Without Scram Analyses (TAC Nos. MA9583 and MB0515)," dated March 16, 2001 (ML010750049)

4.14-5 NRC letter to 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)

4.14-6 NUREG-0847, Supplement 24, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2," September 2011 (ML11277A148)

4.15 Post-LOCA Subcriticality Evaluation

Previous WBN Unit 1 and WBN Unit 2 license amendments to increase the TPBAR quantities to be irradiated included a post-LOCA subcriticality evaluation of a representative core design (References 4.15-1 and 4.15-2). Insertion of TPBARs into WBN Units 1 and 2 presents the potential for a positive reactivity insertion following a LOCA in the event of TPBAR cladding rupture at high temperatures. In the earlier evaluations, TPBAR failure was assumed to occur, and loss of Li-6 material was conservatively accounted for in the post-LOCA subcriticality evaluation. The contribution of Li-6 loss from TPBARs during a LBLOCA reduced the subcriticality margin and accommodations were required to ensure this margin was maintained.

In Reference 4.15-3, NRC approved the use of the Westinghouse FSLOCA methodology for WBN Units 1 and 2, which is used to evaluate TPBAR structural integrity during a LBLOCA. The use of the FSLOCA Evaluation Model yields a reduction in the peak cladding temperature in analyses of LBLOCA and SBLOCA for WBN Units 1 and 2. The application of the new TPBAR stress analysis methodology demonstrates that TPBAR integrity will be maintained following a LBLOCA. As a result, the presence of intact TPBARs is credited in the post-LOCA criticality evaluation as a negative reactivity contribution. Consequently, post-LOCA subcriticality margin is increased. The standard reload methodology for a core containing 2,496 TPBARs confirms post-LOCA subcriticality is maintained.

<u>References</u>

4.15-1 NRC letter to 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)

4.15-2 NRC letter to 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)

4.15-3 NRC letter to 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 (EPID L-2020-LLA-0005)," dated February 26, 2021 (ML21034A166 and ML21034A169)

5.0 REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements and Criteria

The information in Section 5.1 of Reference 3 of Section 7 to this enclosure remains applicable to this license amendment.

5.2 Precedents

TVA has determined that this request is similar to the following WBN license amendments, which have been approved by the NRC for increasing the WBN TS limits on TPBARs.

- NRC letter to 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)

- NRC letter to 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)

Regarding the proposed change to the UFSAR in Attachment 6 to this enclosure, the NRC routinely approves license amendments, which make changes to dose consequence analyses of records. For example, the NRC issued an amendment to Susquehanna Steam Electric Station, which utilized new analysis codes, a new source term, new assumptions, and made other changes (Reference 5 of Section 7 to this enclosure). The source term used in Reference 5 was calculated using TRITON/ORIGEN-ARP in SCALE 6.2.3. The use of ORIGEN-ARP to calculate the core fission product inventory is also discussed in RG 1.183 and RG 1.195. As another example, the NRC approved Certificate of Compliance No. 1042 for TN Americas, which utilized a source term calculated using ORIGEN-ARP in SCALE 6.0 (Reference 6 of Section 7 to this enclosure). While these amendments are not directly applicable to the proposed WBN UFSAR revision, they do demonstrate the acceptability of updating the source term for accident analyses using newer versions of ORIGEN, including the use of ORIGEN-ARP.

5.3 No Significant Hazards Considerations Determination Analysis

TVA proposes to revise the current licensing basis of WBN Facility Operating License Nos. NPF-90 and NPF-96 by revising the WBN Units 1 and 2 TS 4.2.1 to increase the number of TPBARs that can be irradiated in the core to 2,496.

TVA also proposes a supporting change to WBN Units 1 and 2 TS 5.9.6. The LAR also revises WBN Unit 1 TS 5.9.6 to be consistent with WBN Unit 2 TS 5.9.6. WBN Units 1 and 2 TS 5.9.6 are also revised to add WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A, Rev. 0.

The proposed change also contains a revision to the WBN dual unit UFSAR that modifies 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.

TVA has concluded that these changes do not involve a significant hazards consideration. TVA's conclusion is based on its evaluation in accordance with 10 CFR 50.91(a)(1) of the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

1. *Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?*

   Response: No.

   The proposed change to WBN Units 1 and 2 TS 4.2.1 increases the limit on the number of TPBARs that can be irradiated in the core. The safety analyses demonstrated sufficient reactivity control after a postulated LOCA to maintain the reactor core subcritical. This conclusion will be verified for each core that contains TPBARs as part of the normal reload analysis. The TPBARs are not potential sources for accident generation and the modification of the number of TPBARs will not increase the potential for an accident. Therefore, the probability of an accident is not increased by the proposed changes. Because the reactor core remains subcritical after a postulated LOCA, the consequences of an accident are not increased by the proposed changes.

   Implementation of the analytical methods as proposed would continue to provide assurance that appropriate reactor coolant system pressure and temperature limits are established to preserve the integrity of the reactor coolant system. The proposed amendment is based on NRC-approved methodologies. Ensuring appropriate reactor coolant system pressure and temperature limits are established will not adversely affect a structure, system, or component of the plant, plant operations, design functions, or analysis that verifies the capability of a structure, system, or component to perform a design function. Because there are no adverse effects on systems, structures, or components (SSCs), the likelihood of a malfunction is not increased and consequences of previously evaluated accidents in the WBN dual unit UFSAR are not changed.

The proposed change to the source term used for design basis accident analyses is considered to be a departure from a method of evaluation described in the WBN dual unit UFSAR, but it does not modify any SSCs installed at WBN. The change in methods does not affect any initiator or precursor of any accident previously evaluated. Thus, the proposed change does not involve a significant increase in the probability of an accident previously evaluated. Further, radiological dose consequences remain within the limits of regulatory criteria.

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

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

Response: No.

The proposed change to WBN Units 1 and 2 TS 4.2.1 increases the limit on the number of TPBARs that can be irradiated in the core. The safety analyses demonstrated sufficient reactivity control after a postulated LOCA to maintain the reactor core subcritical. This conclusion will be verified for each core that contains TPBARs as part of the normal reload analysis.

Implementation of the analytical methods as proposed would continue to provide assurance that appropriate reactor coolant system pressure and temperature limits are established to preserve the integrity of the reactor coolant system. The proposed amendment is based on NRC-approved methodologies. The proposed amendment does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed amendment does not change the design of SSCs of the plant; or create new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. The proposed amendment would continue to ensure reactor coolant system integrity.

The proposed change to the source term used for design basis accident analyses is considered to be a departure from a method of evaluation described in the WBN dual unit UFSAR, but it does not modify any SSCs installed at WBN. The proposed change does not alter the design function or operation of any SSCs installed at WBN or result in new or different assumptions into the safety analyses. As such, no new failure modes are introduced. Radiological dose consequences remain within the limits of regulatory criteria.

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

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

Response: No.

The proposed change to WBN Units 1 and 2 TS 4.2.1 increases the limit on the number of TPBARs that can be irradiated in the core. The proposed change does not alter any setpoints utilized for the actuation of accident mitigation system or control functions. This conclusion will be verified for each core that contains TPBARs as part of the normal reload analysis.

Implementation of the analytical methods would continue to provide assurance that appropriate reactor coolant system pressure and temperature limits are established in accordance with NRC-approved methodologies. This ensures that the plant is operated within design limits and the margin of safety in the plant safety analysis is maintained.

The proposed change to the source term used for design basis accident analyses is considered a departure from a method of evaluation described in the WBN dual unit UFSAR. Radiological dose consequences remain within the limits of regulatory criteria. The margin of safety is considered to be that provided by meeting the applicable regulatory criteria.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.4 Conclusions

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

6.0 ENVIRONMENTAL CONSIDERATION

The environmental impacts of producing tritium in the TVA at WBN were initially assessed in a 1999 "Final Environmental Impact Statement (EIS) for the Production of Tritium in a Commercial Light Water Reactor" (DOE/EIS0288) prepared by the DOE. TVA was a cooperating agency in the preparation of this EIS. In accordance with 40 CFR 1506.3(c) of the Council on Environmental Quality's 1978 regulations, TVA independently reviewed the EIS prepared by DOE, found it to be adequate, and adopted the EIS. TVA's "Record of Decision and Adoption of Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor" was published in the Federal Register at 65 FR 26259 (May 5, 2000).

The DOE issued a Supplemental Environmental Impact Statement (SEIS) to update the environmental analyses in DOE's 1999 EIS for the Production of Tritium in a Commercial Light Water Reactor in 2016.[2] The SEIS was prepared to address impacts associated with the higher permeation rate for tritium from the TPBARs and DOE's revised estimate of the maximum number of TPBARs necessary to support the current tritium supply requirements. The SEIS considered a variety of alternatives, including irradiating up to 2,500 TPBARs at each WBN unit (up to 5,000 TPBARs in total). The DOE notes that although the TPBAR captures 99.96 percent of the tritium produced, it is still absorbed at a rate lower than was originally analyzed. The results of the analyses presented in the SEIS indicate there would be no significant increase in radiation exposure associated with

[2] See https://www.energy.gov/nepa/eis-0288-s1-production-tritium-commercial-light-water-reactor

TPBAR irradiation for facility workers or the public. For all analyzed alternatives (including TPBAR irradiation at both WBN units), estimated radiation exposures would remain well below regulatory criteria. In accordance with 40 CFR 1506.3(c) of the Council on Environmental Quality's 1978 regulations, TVA independently reviewed the 2016 SEIS prepared by DOE, found it to be adequate, and adopted the SEIS, as published in the Federal Register at 81 FR 11557 (March 4, 2016). TVA's Record of Decision on Production of Tritium in Commercial Light Water Reactors (including both WBN units) was published in the Federal Register at 82 FR 16653 (April 5, 2017).3

TVA provided information to NRC regarding the environmental impacts associated with tritium production from as many as 2,304 TPBARs to support WBN Unit 1 License Amendment 40 (Reference 7 of Section 7 to this enclosure). NRC used this information in their Environmental Assessment and Finding of No Significant Impact for WBN Unit 1 License Amendment 40 (Reference 8 of Section 7 to this enclosure).

TVA also provided updated information to NRC regarding the environmental impacts associated with tritium production from as many as 1,792 TPBARs (Reference 9 of Section 7 to this enclosure for WBN Unit 1 and Reference 3 of Section 7 to this enclosure for WBN Unit 2) to support WBN Unit 1 License Amendment 107 (Reference 1 of Section 7 to this enclosure) and WBN Unit 2 License Amendment 27 (Reference 2 of Section 7 to this enclosure), respectively. NRC used this information in their Environmental Assessment and Finding of No Significant Impact for WBN Unit 1 License Amendment 107 (Reference 10 of Section 7 to this enclosure) and WBN 2 License Amendment 27 (Reference 11 of Section 7 to this enclosure).

TVA has reviewed the 2016 SEIS to determine whether the analysis remains adequate under the National Environmental Policy Act (NEPA) for the proposal to irradiate 2,496 TPBARs at each WBN unit (up to 4,992 TPBARs total). In its review, TVA also considered whether any new information or circumstances relevant to environmental concerns had emerged since 2016 that would require additional environmental analysis. This review is documented in the TVA Determination of NEPA Adequacy (DNA) memorandum, which concluded that the SEIS adequately addresses the potential impacts associated with increasing production of tritium at WBN and that the 2016 SEIS does not require additional supplementation. TVA's review and documentation is consistent with requirements of the Council on Environmental Quality at 40 CFR 1502.9(d) and TVA NEPA procedures at 18 CFR 1318.101(d). In September 2022, DOE also documented their conclusion that the SEIS adequately addresses the proposed increase in tritium production at WBN and that no additional review is necessary.

Based on (1) the 1999 EIS and 2016 SEIS prepared by the DOE; (2) the TVA DNA memorandum; (3) the information provided to NRC (References 7 and 9 of Section 7 to this enclosure) for WBN Unit 1 License Amendments 40 and 107; (4) the information provided to NRC (Reference 3 of Section 7 to this enclosure) for WBN Unit 2 License Amendment 27; (5) the corresponding NRC Environmental Assessments and Findings of No Significant Impact (References 8, 10, and 11 of Section 7 to this enclosure); and (6) the updated evaluation information provided for this proposed amendment in Sections 4.4 and 4.11; the proposed amendment does not involve (i) a significant hazards consideration, (ii) a

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[3] See https://www.tva.com/environment/environmental-stewardship/environmental-reviews/nepa-detail/production-of-tritium-in-a-commercial-light-water-reactor-supplemental-environmental-impact-statement

significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), NRC will not need to prepare an environmental impact statement or environmental assessment in connection with the proposed amendment.

7.0 REFERENCES

1. NRC letter to 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)
2. NRC letter to 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)
3. TVA letter to NRC, 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)
4. NRC letter to 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 (EPID L-2020-LLA-0005)," dated February 26, 2021 (ML21034A166 and ML21034A169)
5. NRC letter to Susquehanna Nuclear, LLC, "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment Nos. 276 And 258 Re: Revise Technical Specification 5.5.2 to Modify the Design-Basis Loss-of-Coolant Accident Analysis (EPID L-2020-LLA-0000)," dated October 8, 2020 (ML20199G749)
6. NRC letter to TN Americas, LLC, "Issuance of Certificate of Compliance No. 1042, Initial Certificate, for the NUHOMS® EOS Storage System (CAC No. L25028)," dated May 3, 2017 (ML17116A277)
7. TVA letter to NRC, "Watts Bar Nuclear Plant — Request for Additional Information (RAI) Regarding Radiological Impact (TAC No. MB1884)," dated May 23, 2002 (ML021490139)
8. NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 — Environmental Assessment and Finding of No Significant Impact for Incore Irradiation Services for the U.S. Department of Energy's Tritium Production Program (TAC No. MB1884)," dated August 20, 2002 (ML022320905)

9. TVA letter to NRC, CNL-16-038, “Application to Revise Technical Specification 4.2.1, ‘Fuel Assemblies’ (WBN-TS-15-03) (TAC No. MF6050) – Radioactive Waste System Design Basis Source Term Supplement to Response to NRC Request for Additional Information - Radiation Protection and Consequence Branch,” dated March 31, 2016 (ML16095A064)

10. NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 — Environmental Assessment and Finding of No Significant Impact Related to License Amendment Request to Revise Technical Specification 4.2.1, ‘Fuel Assemblies’ (CAC No. MF6050),” dated June 23, 2016 (ML16138A020)

11. NRC letter to TVA, "Watts Bar Nuclear Plant, Units 1 and 2 — Environmental Assessment and Finding of No Significant Impact Related to 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 (EPID L-2017-LLA-0427),” dated February 6, 2019 (ML18332A014)

Enclosure 1

Attachment 1

Proposed TS Changes (Mark-Ups) for WBN Unit 1

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~~1792~~ 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~~. 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~~

(continued)

~~CFR 50.60, concerning use of Code Case N-514 to determine LTOP setpoints, dated June 20, 1997.~~

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)

Enclosure 1

Attachment 2

Proposed TS Changes (Mark-Ups) for WBN Unit 2

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~~1792~~ 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.

Enclosure 1

Attachment 3

Proposed TS Changes (Final Typed) for WBN Unit 1

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)

Enclosure 1

Attachment 4

Proposed TS Changes (Final Typed) for WBN Unit 2

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.

Enclosure 1

Attachment 5

Proposed TS Bases Changes (Markup) for WBN Units 1 and 2 (For Information Only)

BASES

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident(DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. References ~~1~~8 and 9 establish~~es~~ the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control and the thermal gradient through the vessel wall are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

(continued)

BASES

REFERENCES

1. Appendix "A" to RCS System Description N3-68-4001, "Watts Bar Unit 1 RCS Pressure and Temperature Limits Report."

2. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

3. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure."

4. ASTM E 185-82, "Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

5. Title 10, Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events."

8. WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

9. WCAP-18124-NP-A, Revision 0 “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018 and WCAP-18124-NP-A Revision 0 Supplement 1-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials," May 2022.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is when RCS pressure and temperature conditions are undergoing planned changes. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. Appendix "B" to RCS System Description N3-68-4001, "Watts Bar Unit 2 RCS Pressure and Temperature Limits Report."

2. Title 10, Code of Federal Regulations, Part 50, Appendix G, "Fracture Toughness Requirements."

3. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

4. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

5. Title 10, Code of Federal Regulations, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

7. ASME Boiler and Pressure Vessel Code, Section XI, Appendix E, "Evaluation of Unanticipated Operating Events."

8. WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004.

9. WCAP-18124-NP-A, Revision 0 “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018 and WCAP-18124-NP-A Revision 0 Supplement 1-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET – Supplement for Extended Beltline Materials," May 2022.

Enclosure 1

Attachment 6

Proposed WBN UFSAR Change (Markups)

Condition IV Events

1. Major Rupture of a Main Steam Line 15.4.2.1

2. Major Rupture of a Main Feedwater Pipe 15.4.2.2

3. Steam Generator Tube Rupture 15.4.3

4. Single Reactor Coolant Pump Locked Rotor 15.4.4

5. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) 15.4.6

15.1.7 Fission Product Inventories

15.1.7.1 Radioactivity in the Core

ORIGEN-S/ORIGEN-ARP modules within the SCALE 6.0[2]

~~Unit 1~~

The core fission product-inventory is calculated by the ~~ORIGEN~~[2] computer code. The inventories of fission products important from a health hazard point of view are given in Table 15.1-4. The isotopes included in Table 15.1-4 are the isotopes controlling from considerations of inhalation dose (iodines) and from direct dose due to immersion (noble gases).

~~Unit 2~~

~~The average core fission product-inventory is calculated by the ORIGEN-S Subcode within the SCALE-4.2 [2] computer code. The inventories of fission products important from a health hazard point of view are given in Table 15.1-4. The isotopes included in Table 15.1-4 are the isotopes controlling from considerations of inhalation dose (iodines) and from direct dose due to immersion (noble gases).~~

based on the core fission product inventories calculated by the ORIGEN-S/ORIGEN-ARP modules within the SCALE 6.0[2]

15.1.7.2 Radioactivity in the Fuel Pellet Clad Gap

~~Unit 1~~

,

The ~~calculation of the~~ maximum core fission product-inventories are also calculated ~~by the ORIGEN~~ computer code and are the basis for determining the gap activities used in single fuel assembly accident analyses. The gap activities are consistent with the guidance of Regulatory Guide ~~1.25~~[3]~~:~~ ~~10% of the total noble gases other than Kr-85 and 30% of Kr-85. For an accident analysis involving a fuel assembly, 10% of the total radioactive iodine in the rods at the time of the accident is also in the gap~~.

1.183

The radioactivity in the reactor coolant as well as in the volume control tank, pressurizer, and waste gas decay tanks are given in Chapter 11 along with the data on which these computations are based.

~~Unit 2~~

~~The calculation of the maximum core fission product inventories are also calculated by the ORIGEN-S computer code and are the basis for determining the gap activities used in single fuel assembly accident analyses. The gap activities are consistent with the guidance of Safety Guide 25 [3]: 10% of the total noble gases other than Kr-85 and 30% of Kr-85. For an accident analysis involving a fuel assembly, 10% of the total radioactive iodine in the rods at the time of the accident is also in the gap.~~

~~The radioactivity in the reactor coolant as well as in the volume control tank, pressurizer, and waste gas decay tanks are given in Chapter 11 along with the data on which these computations are based.~~

15.1.8 Residual Decay Heat

Residual heat in a subcritical core consists of:

1. Fission product decay energy,

2. Decay of neutron capture products, and

3. Residual fissions due to the effect of delayed neutrons.

These constituents are discussed separately in the following paragraphs.

15.1.8.1 Fission Product Decay Energy

For short times (10^3 seconds) after shutdown, data on yields of short half life isotopes is sparse. Very little experimental data is available for the X-ray contributions and even less for the β-ray contribution. Several authors have compiled the available data into a conservative estimate of fission product decay energy for short times after shutdown, notably Shure[7] and Dudziak.[8] Of these two selections, Shure's curve is the highest, and it is based on the data of Stehn and Clancy[10] and Obenshain and Foderaro.[11]

The fission product contribution to decay energy which has been assumed in the accident analyses is the curve of Shure increased by 20% for conservatism unless otherwise stated in the sections describing specific accidents. This curve with the 20% factor included is shown in Figure 15.1-6.

SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. I-III, Version 6 (ORNL/TM-2005/39), January 2009.

<u>REFERENCES</u>

1. Deleted in initial UFSAR.

2a. ~~SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. I-III, NUREG/CR-0200, Rev. 5 (ORNL/NUREG/CSD-2/R5), March 1997. (Unit 1)~~

Deleted.

2b. ~~SCALE-4.2: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Volumes I-III, NUREG/CR-0200, Rev. 5 (ORNL/NUREG/CSD-2/R5), March 1997 (ORIGEN-S Subsection) (Unit 2)~~

3. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents At Nuclear Power Reactors, July 2000

4. Toner, D. F. and Scott, J. S., "Fission-Product Release from UO", <u>Nuc.</u> <u>Safety 3</u> No. 2, 15-20, December 1961.

5. Belle, J., "Uranium Dioxide Properties and Nuclear Applications," Naval Reactors, Division of Reactor Development United States Atomic Energy Commission, 1961.

6. Booth. A. H., "A Suggested Method for Calculating the Diffusion of Radioactive Rare Gas Fission Products From UO Fuel Elements," DCI-27, 1957.

7. Shure, K., "Fission Product Decay Energy" in <u>Bettis Technical Review</u>, WAPD-BT-24, p. 1-17, December 1961.

8. Shure, K. and Dudziak, D. J., "Calculating Energy Released by Fission Products," <u>Trans. Am. Nucl. Soc. 4</u> (1) 30 (1961).

9. Deleted in initial UFSAR.

10. Stehn, J.R. and Clancy, E. F., "Fission-Product Radioactivity and Heat Generation" and "Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958," Volume 13, pp. 49-54, United Nations, Geneva, 1958.

11. Obershain, F. E. and Foderaro, A. H., "Energy from Fission Product Decay," WAPD-P-652, 1955.

12. Hargrove, H. G., "FACTRAN, a FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, December 1989.

13. Deleted in initial UFSAR.

14. Deleted. in initial UFSAR

15. Burnett, T. W. T., et al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary) April 1984.

Enclosure 2

WCAP-18769-NP, Revision 1
Watts Bar Units 1 & 2 Reactor Vessel Integrity Evaluations for the
2496 TPBAR Implementation Project

WCAP-18769-NP
Revision 1

February 2023

Watts Bar Units 1 & 2 Reactor Vessel Integrity Evaluations for the 2496 TPBAR Implementation Project

Westinghouse

WCAP-18769-NP
Revision 1

Watts Bar Units 1 & 2 Reactor Vessel Integrity Evaluations for the 2496 TPBAR Implementation Project

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February 2023

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RECORD OF REVISION

Revision 0: Original Issue

Revision 1: This WCAP was updated to reflect that the PT limits curves meet the requirements of WCAP-14040-A Revision 4 even though they were generated to WCAP-14040-A Revision 1. The limitations to the uncertainty of neutron exposure were also updated to allow the use of the fluence determination methodology on the extended beltline materials.

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EXECUTIVE SUMMARY

This report presents the evaluation of the Watts Bar Units 1 & 2 reactor pressure vessel with respect to reactor vessel integrity. Watts Bar Units 1 & 2 have been approved for 40 years of operation, however; the evaluations in this report are applicable for 32 and 48 effective full-power years. 40 years of operation (32 effective full-power years) is considered end-of-license and 60 years of operation (48 effective full-power years) is considered end-of-license-extension (applicable to Unit 1 only). Note that calculations for Watts Bar Unit 2 were only performed up to 32 effective full power years.

A summary of results for the Watts Bar Units 1 & 2 reactor vessel integrity evaluations are provided. Based on the results presented herein, it is concluded that the Watts Bar Units 1 & 2 reactor pressure vessels will continue to meet reactor pressure vessel integrity regulatory requirements through the end-of-license and end-of-license-extension.

Neutron Fluence

The reactor pressure vessel beltline and extended beltline neutron fluence values applicable through 40 and 60-year license periods were calculated for the Watts Bar Units 1 & 2 beltline and extended beltline materials. The analysis methodologies used to calculate the Watts Bar Units 1 & 2 reactor pressure vessel fluences satisfy the guidance set forth in Regulatory Guide 1.190. See Section 2 for more details.

Pressurized Thermal Shock and Emergency Response Guideline Limits

The RT_{PTS} values of all of the beltline and extended beltline materials in the Watts Bar Units 1 & 2 reactor pressure vessels are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds (per 10 CFR 50.61.b.2), through end-of-license and end-of-license-extension. Additionally, Watts Bar Unit 1 will remain in emergency response guideline Category II through end-of-license and end-of-license-extension, while Watts Bar Unit 2 will remain in Category I through end-of-license. See Section 6 for more details.

Upper-Shelf Energy

The upper-shelf energy values of all of the beltline and extended beltline materials in the Watts Bar Unit 1 reactor pressure vessel are projected to remain above the USE screening criterion of 50 ft-lb (per 10 CFR 50 Appendix G Section IV.1), through end-of-license (32 EFPY) and end-of-license-extension (48 EFPY) with one exception. The Watts Bar Unit 1 intermediate shell forging 05 has a USE value below the 50 ft-lb screening criterion for the end-of-license and end-of-license-extension. However, as previously outlined in WCAP-16760-NP, the upper-shelf energy remains above the 43 ft-lb lower bound as determined in the generic evaluation, WCAP-13587, Revision 1. See Section 7 for more details.

The upper-shelf energy values of all of the beltline and extended beltline materials in the Watts Bar Unit 2 reactor pressure vessel are projected to remain above the USE screening criterion of 50 ft-lb (per 10 CFR 50 Appendix G Section IV.1), through end-of-license (32 EFPY).

Determination of Pressure-Temperature Limit Curve Applicability

Adjusted reference temperatures values are calculated at 32 and 48 effective full-power years. The adjusted reference temperature values are used to perform an applicability check on the pressure-temperature limit curves calculated in WCAP-16761-NP for Watts Bar Unit 1 and WCAP-18191-NP for Watts Bar Unit 2. With the consideration of the updated fluence projections and revised Position 2.1 chemistry factor values, the applicability of the Watts Bar Units 1 & 2 pressure-temperature limit curves remain unchanged at 32 effective full-power years (Units 1 & 2) and 48 effective full-power years (Unit 1). See Section 8 for more details.

Surveillance Capsule Withdrawal Schedules

Recommended surveillance capsule withdrawal schedules were generated for Watts Bar Units 1 & 2. These schedules meet the recommendations of ASTM E185-82 as required by 10 CFR 50, Appendix H.

1 INTRODUCTION

The purpose of this report is to evaluate the Watts Bar Units 1 & 2 beltline and extended beltline materials with respect to reactor vessel integrity (RVI). These materials are evaluated to determine their RT_{PTS}, upper-shelf energy (USE), and adjusted reference temperature (ART) values at end of license (EOL) for Units 1 and 2 and end-of-license-extension (EOLE) for Unit 1 only, which respectively correspond to 32 and 48 effective full-power years (EFPY). The ART values will subsequently be used to validate the applicability period of the pressure-temperature (P-T) curves calculated in WCAP-16761-NP [Ref. 18] (32 and 48 EFPY) for Watts Bar Unit 1 and WCAP-18191-NP [Ref. 2] (32 EFPY) for Watts Bar Unit 2.

Reference nil-ductility transition temperature (RT_{NDT}) increases, and the USE decreases as the material is exposed to fast-neutron irradiation. To find the most limiting RT_{NDT} and USE at any time period in the reactor's life, Regulatory Guide 1.99 (RG 1.99), Revision 2 [Ref. 3] is used to calculate the ΔRT_{NDT} and USE percent decease due to the associated radiation exposure. The resulting limiting ART values are used to satisfy the requirements of 10 CFR Part 50.61 [Ref. 4], the "PTS Rule," and to calculate P-T limit curves in accordance with the requirements of 10 CFR Part 50, Appendix G [Ref. 5], as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code [Ref. 6]. (Note, the methodology to calculate ΔRT_{PTS} is stipulated in 10 CFR Part 50.61; however, it is identical to RG 1.99.) The resulting limiting USE values are used to satisfy the requirements of 10 CFR 50, Appendix G.

Historically, only those materials directly adjacent to the active core, commonly referred to as the traditional beltline, have been evaluated with respect to RVI. However, U.S. NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. 7] states that any materials exceeding 1.0×10^{17} n/cm^2 (E > 1.0 MeV) must be evaluated to determine the changes in fracture toughness. Any materials that have predicted fluence levels greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) are now commonly referred to as the extended beltline. Therefore, these materials are included in the Watts Bar Units 1 & 2 RVI evaluations (i.e., RT_{PTS}, USE, and ART values).

Section 2 of this report discusses the methodologies used to evaluate the neutron fluence and presents the neutron fluence values. Section 0 provides fracture toughness and material properties of the materials in the Watts Bar Units 1 & 2 reactor vessel beltline and extended beltline regions. Sections 4 and 5 identify relevant surveillance data and RG 1.99 chemistry factors (CF) that will be used in the evaluation within this report. Section 6 analyzes the RT_{PTS} values and emergency response guideline (ERG) limits. Section 7 analyzes the USE values. Section 8 calculates the EOL (Units 1 & 2) and EOLE (Unit 1) quarter thickness (1/4T) and three-quarter thickness (3/4T) ART values and evaluates the P-T limit curves applicability period.

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

This section describes discrete ordinates transport analyses performed for the Watts Bar Units 1 & 2 reactors. The purpose of these transport analyses was to characterize the neutron radiation environment within the reactor pressure vessel, surveillance capsules, and surrounding structures. In these analyses, fast neutron exposure parameters in terms of fast neutron (E > 1.0 MeV) fluence and iron atom displacements (dpa) were established on a plant-specific and fuel cycle-specific basis. A re-evaluation of dosimetry sensor sets withdrawn from the reactors is provided. Comparisons of the results from the dosimetry evaluations with the analytical predictions served to validate the plant-specific neutron transport calculations. These validated calculations form the basis for projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 48 EFPY.

The use of fast neutron (E > 1.0 MeV) fluence to correlate measured material property changes to the neutron exposure of the material has traditionally been accepted for the development of damage trend curves as well as for the implementation of trend curve data to assess the condition of the vessel. However, it has been suggested that an exposure model that accounts for differences in neutron energy spectra between surveillance capsule locations and positions within the vessel wall could lead to an improvement in the uncertainties associated with damage trend curves and improved accuracy in the evaluation of damage gradients through the reactor vessel wall.

Because of this potential shift away from a threshold fluence toward an energy-dependent damage function for data correlation, ASTM Standard Practice E853-18, "Standard Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Neutron Exposure Results" [Ref. 8] recommends reporting displacements per iron atom (dpa) along with fluence (E > 1.0 MeV) to provide a database for future reference. The energy-dependent dpa function to be used for this evaluation is specified in ASTM Standard Practice E693-94, "Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements per Atom" [Ref. 9]. The application of the dpa parameter to the assessment of embrittlement gradients through the thickness of the reactor vessel wall has been promulgated in Revision 2 to RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials" [Ref. 3].

The calculations and dosimetry evaluations described in this section were based on nuclear cross-section data derived from ENDF/B-VI. Furthermore, the neutron transport and dosimetry evaluation methodologies follow the guidance of RG 1.190 [Ref. 10]. The methods used to develop the calculated pressure vessel fluence are consistent with the NRC-approved methodology described in WCAP-18124-NP-A [Ref. 11] and WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A [Ref. 20].

2.2 DISCRETE ORDINATES ANALYSIS

For both Units 1 & 2, six irradiation capsules attached to the neutron pad were included in each reactor design. These surveillance capsules constitute the reactor vessel surveillance program for each unit. Each unit was equipped with capsules U, X, V, Y, W, and Z located at azimuthal angles of 56.0°, 236.0°, 58.5°, 238.5°, 124.0°, and 304.0°, respectively. A representative quadrant model of the reactor was developed for the transport analyses. In the quadrant model, the full-core azimuthal positions of these capsules correspond with the following quadrant-symmetric locations.

Capsule	Specimen Guide Type	Azimuthal Angle (°)	
		Full-Core	First Octant Equivalent
U	Dual	56	34
W	Single	124	34
X	Dual	236	34
Z	Single	304	34
V	Dual	58.5	31.5
Y	Dual	238.5	31.5

Watts Bar Units 1 & 2 reactors are sufficiently similar that they could be represented by a single, three-dimensional, quadrant-symmetric model. The submodels depicted in Figure 2-1 and Figure 2-2 were developed to represent quadrants with a dual-capsule specimen guide and a single-capsule specimen guide, respectively, each with the corresponding neutron pad. The stainless steel specimen containers are 1.182-inch by 1-inch and are approximately 56 inches in height. The containers are positioned axially such that the test specimens are centered on the core midplane, thus spanning the central 5 feet of the 12-foot high reactor core.

From a neutronic standpoint, the surveillance capsules and associated support structures are significant. The presence of these materials has a significant effect on both the spatial distribution of the neutron exposure rate and the neutron spectrum in the vicinity of the capsules. However, the capsules are far enough apart that they do not interfere with one another. It is important to include the surveillance specimens in the analytical model to accurately determine the neutron environment. Plant-specific three-dimensional, forward transport calculations were carried out to directly solve for the space- and energy-dependent neutron exposure rate, $\phi(r,\theta,z,E)$.

The model contained a representation of the reactor core, the reactor internals, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, the inlet and outlet nozzles, the vessel support structure, and the primary biological shield wall. Features of the reactors in the extended beltline regions were, such as the inlet and outlet nozzles, were also represented. The model formed the basis for the calculated results and enabled comparisons to the surveillance capsule dosimetry evaluations. In developing this analytical model, nominal design dimensions were employed for the various structural components.

In addition, water temperatures, and hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. The coolant densities were treated on a fuel-cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

A three-dimensional view of the RAPTOR-M3G model of the reactors is shown in Figure 2-3. The model extended radially from the centerline of the reactor core to a location interior to the primary biological shield and over an axial span from an elevation approximately seven and a half feet below the active fuel to five feet above the active fuel. Figure 2-4 shows section view of the reactor with surveillance capsule and bioshield structure.

Both RAPTOR-M3G submodels consisted of 284 radial mesh and 440 vertical mesh, with the single surveillance capsule submodel containing 199 azimuthal mesh and the dual surveillance capsule submodel containing 206 mesh. Mesh sizes were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion utilized in the RAPTOR-M3G calculations was set at a value of 0.001.

The core power distributions used in the plant-specific transport analysis for the first eighteen fuel cycles at Watts Bar Unit 1 and the first four fuel cycles at Watts Bar Unit 2 included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. Actual operating characteristics through each completed cycle were evaluated. Projections of future neutron exposure were based upon the example core loading pattern and expected operating characteristics for an equilibrium fuel cycle containing 2,496 tritium-producing burnable absorber rods (TPBAR) in the core. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle for each unit. Therefore, the results from the neutron transport calculations provided data in terms of fuel-cycle-averaged neutron exposure rate, which when multiplied by the appropriate fuel cycle length, generated the incremental fast neutron exposure for each fuel cycle. In constructing these core source distributions, the energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of individual fuel assemblies. From these assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined.

The transport calculations supporting this analysis were carried out using the RAPTOR-M3G discrete ordinates code and the BUGLE-96 cross-section library, as described in WCAP-18124-NP-A [Ref. 11]. The BUGLE-96 library provides a coupled 47-neutron, 20-gamma-group cross-section data set produced specifically for light-water reactor (LWR) applications. Anisotropic scattering was treated with a P_3 Legendre expansion. Consistent with the additional requirements for extended beltline region fluence analyses described in WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A [Ref. 20], angular discretization was modeled with an S_{16} order of angular quadrature. Energy- and space-dependent core power distributions, as well as system operating temperatures, were treated on a fuel-cycle-specific basis.

2.2.1 Watts Bar Unit 1 Transport Analysis

Selected results from the neutron transport analyses for Watts Bar Unit 1 are provided in Table 2-1 through Table 2-13. In Table 2-1, the calculated fast neutron (E > 1.0 MeV) fluence rates at the radial and azimuthal center of the surveillance capsule positions at core midplane are presented. Integrated fast neutron fluence at the center of the surveillance capsules is presented in Table 2-2. Iron dpa rates at the center of the surveillance capsules are shown in Table 2-4, with the integrated iron dpa presented in Table 2-5. These results, representative of the exposure of the material specimens, establish the calculated exposure of the surveillance capsules to-date and projected into the future based upon the equilibrium 2496 TPBAR core.

Updated lead factors for the surveillance capsules are provided in Table 2-3. The capsule lead factor is defined as the ratio of the calculated fast neutron fluence at the radial and azimuthal center of the surveillance capsule to the maximum calculated fast neutron fluence at the pressure vessel clad/base metal interface. In Table 2-3, the lead factors for capsules that have been withdrawn from the reactor (U, W, X, and Z) were based on the calculated fluence values for the irradiation period corresponding to the time of withdrawal for the individual capsules, which is indicated. The capsules that remain in the reactor, capsules V and Y, are standby capsules.

Neutron exposure data pertinent to selected pressure vessel weld materials are given in Table 2-6 and Table 2-7 for fast neutron (E > 1.0 MeV) fluence rate and fluence. Similar data for pressure vessel weld materials are provided in Table 2-8 and Table 2-9 for dpa/s and dpa. Neutron exposure data pertinent to selected pressure vessel forgings are given in Table 2-10 and Table 2-11 for fast neutron (E > 1.0 MeV) fluence rate and fluence. Similar data for pressure vessel forgings are provided in Table 2-12 and Table 2-13 for dpa/s and dpa. The data presented represent the maximum neutron exposure experienced by the RPV materials that will constitute inputs to the reactor vessel integrity analysis. The reported data considers both the inner and outer radius of the RPV base metal, and accounts for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core. In each case, the data are provided for each operating cycle. For any given fuel cycle, the location of the maximum neutron exposure rate may or may not coincide with the location of the maximum neutron exposure.

Neutron exposure data specific to the cladding/base metal interface is provided in Table 2-14 through Table 2-18. The axial and azimuthal maximum of the fast neutron fluence rate and fluence at the cladding/base metal interface are given in Table 2-14 and Table 2-15. The axial and azimuthal maximum of the iron atom displacement rate and iron atom displacements at the cladding/base metal interface are given in Table 2-16 and Table 2-17. Table 2-18 presents the parameters from Tables 2-14 through 2-17 at selected azimuthal locations about the cladding/base metal interface.

These data tabulations include both plant-specific and fuel-cycle-specific calculated neutron exposures at the end of Cycle 18 and projections to 32 and 48 EFPY. Projections of neutron exposure beyond the end of Cycle 18 are based on the example core loading pattern and expected operating characteristics for the equilibrium 2496 TPBAR core and the rated thermal power of 3459 MWt.

Table 2-1: Unit 1 – Calculated Fast Neutron (E > 1.0 MeV) Fluence Rate at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Fluence Rate (n/cm^2-s)		
		Dual Capsule		Single Capsule
		34°	31.5°	34°
1	1.20	1.21E+11	1.01E+11	1.20E+11
2	2.50	7.50E+10	6.43E+10	7.44E+10
3	3.88	7.48E+10	6.39E+10	7.43E+10
4	5.21	8.56E+10	7.28E+10	8.50E+10
5	6.62	6.64E+10	5.61E+10	6.60E+10
6	7.94	7.38E+10	6.20E+10	7.34E+10
7	9.29	8.38E+10	7.05E+10	8.33E+10
8	10.47	7.97E+10	6.60E+10	7.93E+10
9	11.88	7.70E+10	6.50E+10	7.65E+10
10	13.28	8.41E+10	7.12E+10	8.35E+10
11	14.54	7.98E+10	6.89E+10	7.93E+10
12	15.91	8.55E+10	7.20E+10	8.50E+10
13	17.25	9.03E+10	7.63E+10	8.97E+10
14	18.57	8.74E+10	7.35E+10	8.69E+10
15	19.91	8.15E+10	7.06E+10	8.09E+10
16	21.28	9.19E+10	7.71E+10	9.13E+10
17	22.66	9.84E+10	8.15E+10	9.78E+10
18	24.06	9.43E+10	7.93E+10	9.37E+10
Projections	--	8.66E+10	7.49E+10	8.60E+10

Table 2-2: Unit 1 – Calculated Fast Neutron (E > 1.0 MeV) Fluence at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Fluence (n/cm^2)		
		Dual Capsule		Single Capsule
		34°	31.5° (Capsules V and Y)	34°
1	1.20	4.60E+18 (U)	3.83E+18	4.57E+18
2	2.50	7.66E+18	6.45E+18	7.61E+18
3	3.88	1.09E+19	9.23E+18	1.08E+19 (W)
4	5.21	1.45E+19	1.23E+19	1.44E+19
5	6.62	1.75E+19 (X)	1.48E+19	1.74E+19
6	7.94	2.06E+19	1.74E+19	2.04E+19
7	9.29	2.41E+19	2.04E+19	2.40E+19 (Z)
8	10.47	2.71E+19	2.28E+19	2.69E+19
9	11.88	3.05E+19	2.57E+19	3.03E+19
10	13.28	3.42E+19	2.89E+19	3.40E+19
11	14.54	3.74E+19	3.16E+19	3.72E+19
12	15.91	4.11E+19	3.47E+19	4.08E+19
13	17.25	4.49E+19	3.79E+19	4.46E+19
14	18.57	4.86E+19	4.10E+19	4.83E+19
15	19.91	5.20E+19	4.40E+19	5.17E+19
16	21.28	5.60E+19	4.73E+19	5.56E+19
17	22.66	6.03E+19	5.09E+19	5.99E+19
18	24.06	6.44E+19	5.44E+19	6.40E+19
Projections	32.00	8.61E+19	7.31E+19	8.56E+19
	48.00	1.30E+20	1.11E+20	1.29E+20

Table 2-3: Unit 1 – Surveillance Capsule Lead Factors

Cycle	Cumulative Operating Time (EFPY)	Lead Factors		
		Dual Capsule		Single Capsule
		34°	31.5° (Capsules V and Y)	34°
1	1.20	4.87 (Capsule U)	4.05	4.84
2	2.50	4.84	4.07	4.80
3	3.88	4.81	4.07	4.78 (Capsule W)
4	5.21	4.85	4.11	4.82
5	6.62	4.83 (Capsule X)	4.09	4.80
6	7.94	--	4.07	4.78
7	9.29	--	4.05	4.76 (Capsule Z)
8	10.47	--	4.06	--
9	11.88	--	4.04	--
10	13.28	--	4.06	--
11	14.54	--	4.07	--
12	15.91	--	4.07	--
13	17.25	--	4.07	--
14	18.57	--	4.07	--
15	19.91	--	4.09	--
16	21.28	--	4.10	--
17	22.66	--	4.10	--
18	24.06	--	4.11	--
Projections	32.00	--	4.17	--
	48.00	--	4.06	--

Table 2-4: Unit 1 – Calculated Iron Atom Displacement Rate at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacement Rate (dpa/s)		
		Dual Capsule		Single Capsule
		34°	31.5°	34°
1	1.20	2.46E-10	2.02E-10	2.46E-10
2	2.50	1.51E-10	1.28E-10	1.50E-10
3	3.88	1.50E-10	1.26E-10	1.50E-10
4	5.21	1.72E-10	1.44E-10	1.72E-10
5	6.62	1.34E-10	1.11E-10	1.33E-10
6	7.94	1.49E-10	1.23E-10	1.48E-10
7	9.29	1.69E-10	1.40E-10	1.68E-10
8	10.47	1.61E-10	1.31E-10	1.60E-10
9	11.88	1.55E-10	1.29E-10	1.55E-10
10	13.28	1.69E-10	1.41E-10	1.69E-10
11	14.54	1.61E-10	1.37E-10	1.60E-10
12	15.91	1.73E-10	1.43E-10	1.72E-10
13	17.25	1.82E-10	1.52E-10	1.81E-10
14	18.57	1.76E-10	1.46E-10	1.76E-10
15	19.91	1.64E-10	1.40E-10	1.63E-10
16	21.28	1.86E-10	1.53E-10	1.85E-10
17	22.66	1.99E-10	1.63E-10	1.99E-10
18	24.06	1.91E-10	1.58E-10	1.90E-10
Projections	--	1.75E-10	1.49E-10	1.74E-10

Table 2-5: Unit 1 – Calculated Iron Atom Displacements at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacements (dpa)		
		Dual Capsule		Single Capsule
		34°	31.5° (Capsules V and Y)	34°
1	1.20	9.37E-03 (U)	7.67E-03	9.33E-03
2	2.50	1.55E-02	1.29E-02	1.55E-02
3	3.88	2.21E-02	1.84E-02	2.20E-02 (W)
4	5.21	2.93E-02	2.45E-02	2.92E-02
5	6.62	3.53E-02 (X)	2.94E-02	3.51E-02
6	7.94	4.15E-02	3.45E-02	4.13E-02
7	9.29	4.86E-02	4.05E-02	4.84E-02 (Z)
8	10.47	5.46E-02	4.54E-02	5.44E-02
9	11.88	6.16E-02	5.11E-02	6.13E-02
10	13.28	6.91E-02	5.74E-02	6.88E-02
11	14.54	7.54E-02	6.28E-02	7.51E-02
12	15.91	8.29E-02	6.90E-02	8.26E-02
13	17.25	9.06E-02	7.54E-02	9.02E-02
14	18.57	9.80E-02	8.15E-02	9.76E-02
15	19.91	1.05E-01	8.74E-02	1.04E-01
16	21.28	1.13E-01	9.40E-02	1.12E-01
17	22.66	1.22E-01	1.01E-01	1.21E-01
18	24.06	1.30E-01	1.08E-01	1.29E-01
Projections	32.00	1.74E-01	1.45E-01	1.73E-01
	48.00	2.62E-01	2.21E-01	2.61E-01

Table 2-6: Unit 1 – Calculated Maximum Fast Neutron Fluence Rate in Reactor Welds

Cycle	EFPY	Fluence Rate (n/cm^2-s)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	1.20	3.39E+07	1.58E+07	5.25E+08	2.40E+10	1.70E+09	7.81E+06
2	2.50	4.58E+07	2.16E+07	6.48E+08	1.47E+10	1.79E+09	6.83E+06
3	3.88	3.75E+07	1.79E+07	5.29E+08	1.47E+10	1.55E+09	6.34E+06
4	5.21	3.80E+07	1.82E+07	5.62E+08	1.68E+10	1.39E+09	5.98E+06
5	6.62	3.95E+07	1.91E+07	5.45E+08	1.36E+10	1.29E+09	5.10E+06
6	7.94	4.33E+07	2.07E+07	6.25E+08	1.51E+10	1.53E+09	5.81E+06
7	9.29	5.80E+07	2.76E+07	8.67E+08	1.67E+10	2.35E+09	7.63E+06
8	10.47	5.30E+07	2.52E+07	7.85E+08	1.59E+10	1.96E+09	6.49E+06
9	11.88	3.99E+07	1.92E+07	5.76E+08	1.60E+10	1.50E+09	6.09E+06
10	13.28	4.26E+07	1.96E+07	6.22E+08	1.65E+10	1.49E+09	6.07E+06
11	14.54	3.83E+07	1.80E+07	5.32E+08	1.55E+10	1.49E+09	6.22E+06
12	15.91	3.98E+07	1.88E+07	5.78E+08	1.70E+10	1.59E+09	6.57E+06
13	17.25	4.58E+07	2.15E+07	6.70E+08	1.79E+10	1.37E+09	6.22E+06
14	18.57	4.18E+07	2.02E+07	6.13E+08	1.75E+10	1.22E+09	5.85E+06
15	19.91	4.17E+07	2.01E+07	5.86E+08	1.62E+10	1.12E+09	5.50E+06
16	21.28	4.15E+07	1.96E+07	6.11E+08	1.76E+10	1.16E+09	5.74E+06
17	22.66	4.47E+07	2.08E+07	6.76E+08	1.96E+10	1.26E+09	6.26E+06
18	24.06	4.47E+07	2.09E+07	6.30E+08	1.85E+10	1.15E+09	5.99E+06
Projections		4.50E+07	2.10E+07	6.16E+08	1.84E+10	1.22E+09	6.03E+06

Table 2-7: Unit 1 – Calculated Maximum Fast Neutron Fluence in Reactor Welds

Cycle	EFPY	Fluence (n/cm^2)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	1.20	1.29E+15	5.99E+14	2.00E+16	9.14E+17	6.45E+16	2.97E+14
2	2.50	3.15E+15	1.48E+15	4.64E+16	1.51E+18	1.38E+17	5.73E+14
3	3.88	4.79E+15	2.26E+15	6.95E+16	2.15E+18	2.05E+17	8.50E+14
4	5.21	6.39E+15	3.02E+15	9.31E+16	2.86E+18	2.64E+17	1.10E+15
5	6.62	8.15E+15	3.88E+15	1.17E+17	3.47E+18	3.21E+17	1.33E+15
6	7.94	9.95E+15	4.74E+15	1.43E+17	4.10E+18	3.85E+17	1.57E+15
7	9.29	1.24E+16	5.91E+15	1.80E+17	4.81E+18	4.84E+17	1.89E+15
8	10.47	1.44E+16	6.85E+15	2.09E+17	5.40E+18	5.57E+17	2.14E+15
9	11.88	1.62E+16	7.70E+15	2.35E+17	6.11E+18	6.24E+17	2.41E+15
10	13.28	1.81E+16	8.57E+15	2.63E+17	6.84E+18	6.90E+17	2.67E+15
11	14.54	1.96E+16	9.28E+15	2.84E+17	7.45E+18	7.49E+17	2.92E+15
12	15.91	2.13E+16	1.01E+16	3.09E+17	8.18E+18	8.17E+17	3.21E+15
13	17.25	2.32E+16	1.10E+16	3.37E+17	8.93E+18	8.75E+17	3.47E+15
14	18.57	2.50E+16	1.18E+16	3.63E+17	9.67E+18	9.26E+17	3.71E+15
15	19.91	2.67E+16	1.27E+16	3.87E+17	1.03E+19	9.73E+17	3.94E+15
16	21.28	2.85E+16	1.35E+16	4.14E+17	1.11E+19	1.02E+18	4.18E+15
17	22.66	3.05E+16	1.44E+16	4.43E+17	1.19E+19	1.08E+18	4.45E+15
18	24.06	3.24E+16	1.54E+16	4.71E+17	1.27E+19	1.13E+18	4.72E+15
Projections	32.00	4.37E+16	2.06E+16	6.25E+17	1.68E+19	1.43E+18	6.23E+15
	48.00	6.64E+16	3.13E+16	9.37E+17	2.60E+19	2.05E+18	9.27E+15

Table 2-8: Unit 1 – Calculated Maximum Iron Atom Displacement Rate in Reactor Welds

Cycle	EFPY	Iron Atom Displacement Rate (dpa/s)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	1.20	1.63E-13	1.02E-13	8.72E-13	3.83E-11	2.72E-12	5.04E-14
2	2.50	1.52E-13	9.39E-14	1.06E-12	2.35E-11	2.86E-12	4.33E-14
3	3.88	1.36E-13	8.50E-14	8.70E-13	2.34E-11	2.48E-12	4.02E-14
4	5.21	1.35E-13	8.56E-14	9.22E-13	2.68E-11	2.22E-12	3.80E-14
5	6.62	1.22E-13	7.61E-14	8.91E-13	2.17E-11	2.06E-12	3.24E-14
6	7.94	1.35E-13	8.31E-14	1.02E-12	2.40E-11	2.44E-12	3.69E-14
7	9.29	1.68E-13	1.05E-13	1.41E-12	2.67E-11	3.72E-12	4.83E-14
8	10.47	1.55E-13	9.61E-14	1.28E-12	2.54E-11	3.10E-12	4.10E-14
9	11.88	1.38E-13	8.76E-14	9.44E-13	2.55E-11	2.39E-12	3.87E-14
10	13.28	1.48E-13	8.94E-14	1.02E-12	2.63E-11	2.38E-12	3.86E-14
11	14.54	1.38E-13	8.51E-14	8.74E-13	2.45E-11	2.38E-12	3.96E-14
12	15.91	1.46E-13	9.10E-14	9.48E-13	2.70E-11	2.54E-12	4.19E-14
13	17.25	1.62E-13	1.01E-13	1.10E-12	2.85E-11	2.19E-12	3.96E-14
14	18.57	1.48E-13	9.46E-14	1.00E-12	2.79E-11	1.96E-12	3.72E-14
15	19.91	1.43E-13	9.22E-14	9.60E-13	2.48E-11	1.80E-12	3.49E-14
16	21.28	1.50E-13	9.36E-14	1.00E-12	2.81E-11	1.86E-12	3.66E-14
17	22.66	1.66E-13	1.02E-13	1.11E-12	3.12E-11	2.02E-12	4.00E-14
18	24.06	1.66E-13	1.03E-13	1.03E-12	2.88E-11	1.86E-12	3.83E-14
Projections		1.62E-13	1.01E-13	1.01E-12	2.82E-11	1.96E-12	3.86E-14

Table 2-9: Unit 1 – Calculated Maximum Iron Atom Displacements in Reactor Welds

Cycle	EFPY	Iron Atom Displacements (dpa)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	1.20	6.18E-06	3.86E-06	3.31E-05	1.46E-03	1.04E-04	1.91E-06
2	2.50	1.24E-05	7.69E-06	7.64E-05	2.41E-03	2.20E-04	3.68E-06
3	3.88	1.83E-05	1.14E-05	1.14E-04	3.43E-03	3.28E-04	5.43E-06
4	5.21	2.40E-05	1.50E-05	1.53E-04	4.56E-03	4.22E-04	7.03E-06
5	6.62	2.94E-05	1.84E-05	1.93E-04	5.53E-03	5.14E-04	8.48E-06
6	7.94	3.50E-05	2.18E-05	2.35E-04	6.53E-03	6.15E-04	1.00E-05
7	9.29	4.22E-05	2.63E-05	2.95E-04	7.66E-03	7.73E-04	1.21E-05
8	10.47	4.80E-05	2.99E-05	3.43E-04	8.60E-03	8.89E-04	1.36E-05
9	11.88	5.41E-05	3.38E-05	3.85E-04	9.73E-03	9.95E-04	1.53E-05
10	13.28	6.07E-05	3.77E-05	4.30E-04	1.09E-02	1.10E-03	1.70E-05
11	14.54	6.61E-05	4.11E-05	4.65E-04	1.19E-02	1.19E-03	1.86E-05
12	15.91	7.24E-05	4.50E-05	5.06E-04	1.30E-02	1.30E-03	2.04E-05
13	17.25	7.93E-05	4.93E-05	5.52E-04	1.42E-02	1.40E-03	2.21E-05
14	18.57	8.55E-05	5.32E-05	5.94E-04	1.54E-02	1.48E-03	2.36E-05
15	19.91	9.15E-05	5.71E-05	6.34E-04	1.64E-02	1.55E-03	2.51E-05
16	21.28	9.80E-05	6.12E-05	6.78E-04	1.76E-02	1.63E-03	2.67E-05
17	22.66	1.05E-04	6.56E-05	7.26E-04	1.90E-02	1.72E-03	2.84E-05
18	24.06	1.13E-04	7.01E-05	7.72E-04	2.03E-02	1.80E-03	3.01E-05
Projections	32.00	1.53E-04	9.55E-05	1.02E-03	2.68E-02	2.29E-03	3.98E-05
	48.00	2.35E-04	1.47E-04	1.54E-03	4.00E-02	3.28E-03	5.92E-05

Table 2-10: Unit 1 – Calculated Maximum Fast Neutron Fluence Rate in Reactor Forgings

Cycle	EFPY	Fluence Rate (n/cm^2-s)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	1.20	2.33E+07	1.37E+07	4.38E+08	2.41E+10	2.49E+10	1.41E+09
2	2.50	3.19E+07	1.51E+07	5.47E+08	1.53E+10	1.63E+10	1.49E+09
3	3.88	2.63E+07	1.25E+07	4.46E+08	1.50E+10	1.58E+10	1.29E+09
4	5.21	2.65E+07	1.28E+07	4.74E+08	1.70E+10	1.73E+10	1.15E+09
5	6.62	2.78E+07	1.34E+07	4.63E+08	1.38E+10	1.40E+10	1.07E+09
6	7.94	3.03E+07	1.46E+07	5.28E+08	1.53E+10	1.57E+10	1.28E+09
7	9.29	4.04E+07	1.94E+07	7.33E+08	1.70E+10	1.78E+10	1.96E+09
8	10.47	3.71E+07	1.76E+07	6.62E+08	1.63E+10	1.62E+10	1.63E+09
9	11.88	2.78E+07	1.34E+07	4.87E+08	1.61E+10	1.68E+10	1.24E+09
10	13.28	2.97E+07	1.38E+07	5.24E+08	1.68E+10	1.69E+10	1.24E+09
11	14.54	2.67E+07	1.26E+07	4.49E+08	1.57E+10	1.67E+10	1.24E+09
12	15.91	2.77E+07	1.31E+07	4.87E+08	1.72E+10	1.79E+10	1.32E+09
13	17.25	3.18E+07	1.51E+07	5.64E+08	1.82E+10	1.84E+10	1.14E+09
14	18.57	2.91E+07	1.41E+07	5.16E+08	1.78E+10	1.81E+10	1.01E+09
15	19.91	2.90E+07	1.40E+07	4.95E+08	1.65E+10	1.69E+10	9.29E+08
16	21.28	2.89E+07	1.37E+07	5.13E+08	1.78E+10	1.83E+10	9.62E+08
17	22.66	3.10E+07	1.45E+07	5.67E+08	1.99E+10	2.04E+10	1.04E+09
18	24.06	3.10E+07	1.46E+07	5.27E+08	1.88E+10	1.93E+10	9.56E+08
Projections		3.11E+07	1.47E+07	5.18E+08	1.87E+10	1.93E+10	1.01E+09

Table 2-11: Unit 1 – Calculated Maximum Fast Neutron Fluence in Reactor Forgings

Cycle	EFPY	Fluence (n/cm^2)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	1.20	8.86E+14	5.19E+14	1.67E+16	9.16E+17	9.45E+17	5.34E+16
2	2.50	2.19E+15	1.13E+15	3.90E+16	1.53E+18	1.58E+18	1.14E+17
3	3.88	3.33E+15	1.68E+15	5.84E+16	2.18E+18	2.27E+18	1.71E+17
4	5.21	4.44E+15	2.22E+15	7.84E+16	2.90E+18	2.99E+18	2.19E+17
5	6.62	5.69E+15	2.82E+15	9.90E+16	3.51E+18	3.62E+18	2.67E+17
6	7.94	6.95E+15	3.43E+15	1.21E+17	4.15E+18	4.28E+18	3.20E+17
7	9.29	8.66E+15	4.25E+15	1.52E+17	4.87E+18	5.03E+18	4.03E+17
8	10.47	1.00E+16	4.91E+15	1.77E+17	5.47E+18	5.62E+18	4.64E+17
9	11.88	1.13E+16	5.51E+15	1.98E+17	6.19E+18	6.37E+18	5.19E+17
10	13.28	1.26E+16	6.12E+15	2.22E+17	6.93E+18	7.12E+18	5.74E+17
11	14.54	1.37E+16	6.61E+15	2.39E+17	7.55E+18	7.77E+18	6.23E+17
12	15.91	1.49E+16	7.18E+15	2.61E+17	8.29E+18	8.54E+18	6.80E+17
13	17.25	1.62E+16	7.82E+15	2.84E+17	9.06E+18	9.32E+18	7.28E+17
14	18.57	1.74E+16	8.41E+15	3.06E+17	9.80E+18	1.01E+19	7.71E+17
15	19.91	1.86E+16	9.00E+15	3.27E+17	1.05E+19	1.08E+19	8.10E+17
16	21.28	1.99E+16	9.59E+15	3.49E+17	1.12E+19	1.15E+19	8.51E+17
17	22.66	2.12E+16	1.02E+16	3.74E+17	1.21E+19	1.24E+19	8.96E+17
18	24.06	2.26E+16	1.09E+16	3.97E+17	1.29E+19	1.32E+19	9.38E+17
Projections	32.00	3.04E+16	1.45E+16	5.27E+17	1.71E+19	1.75E+19	1.19E+18
	48.00	4.61E+16	2.19E+16	7.88E+17	2.64E+19	2.73E+19	1.70E+18

Table 2-12: Unit 1 – Calculated Maximum Iron Atom Displacement Rate in Reactor Forgings

Cycle	EFPY	Iron Atom Displacement Rate (dpa/s)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	1.20	1.48E-13	9.46E-14	7.28E-13	3.83E-11	3.94E-11	2.27E-12
2	2.50	1.38E-13	8.73E-14	8.92E-13	2.42E-11	2.58E-11	2.40E-12
3	3.88	1.24E-13	7.91E-14	7.31E-13	2.37E-11	2.50E-11	2.08E-12
4	5.21	1.23E-13	7.96E-14	7.74E-13	2.70E-11	2.74E-11	1.86E-12
5	6.62	1.10E-13	7.08E-14	7.50E-13	2.19E-11	2.22E-11	1.72E-12
6	7.94	1.22E-13	7.73E-14	8.59E-13	2.43E-11	2.49E-11	2.04E-12
7	9.29	1.53E-13	9.72E-14	1.19E-12	2.69E-11	2.82E-11	3.12E-12
8	10.47	1.41E-13	8.93E-14	1.07E-12	2.58E-11	2.57E-11	2.60E-12
9	11.88	1.26E-13	8.14E-14	7.93E-13	2.56E-11	2.66E-11	2.00E-12
10	13.28	1.35E-13	8.32E-14	8.55E-13	2.67E-11	2.69E-11	1.99E-12
11	14.54	1.25E-13	7.92E-14	7.35E-13	2.46E-11	2.59E-11	1.99E-12
12	15.91	1.33E-13	8.47E-14	7.96E-13	2.73E-11	2.83E-11	2.13E-12
13	17.25	1.47E-13	9.36E-14	9.20E-13	2.89E-11	2.92E-11	1.83E-12
14	18.57	1.35E-13	8.80E-14	8.43E-13	2.82E-11	2.86E-11	1.64E-12
15	19.91	1.30E-13	8.57E-14	8.07E-13	2.52E-11	2.58E-11	1.50E-12
16	21.28	1.37E-13	8.71E-14	8.39E-13	2.83E-11	2.89E-11	1.56E-12
17	22.66	1.51E-13	9.49E-14	9.28E-13	3.15E-11	3.22E-11	1.69E-12
18	24.06	1.51E-13	9.55E-14	8.65E-13	2.90E-11	2.96E-11	1.55E-12
Projections		1.47E-13	9.41E-14	8.47E-13	2.85E-11	2.95E-11	1.64E-12

Table 2-13: Unit 1 – Calculated Maximum Iron Atom Displacements in Reactor Forgings

Cycle	EFPY	Iron Atom Displacements (dpa)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	1.20	5.63E-06	3.60E-06	2.77E-05	1.46E-03	1.50E-03	8.63E-05
2	2.50	1.12E-05	7.15E-06	6.40E-05	2.43E-03	2.51E-03	1.84E-04
3	3.88	1.67E-05	1.06E-05	9.59E-05	3.46E-03	3.60E-03	2.75E-04
4	5.21	2.18E-05	1.40E-05	1.29E-04	4.60E-03	4.75E-03	3.53E-04
5	6.62	2.68E-05	1.71E-05	1.62E-04	5.58E-03	5.74E-03	4.30E-04
6	7.94	3.19E-05	2.03E-05	1.98E-04	6.59E-03	6.77E-03	5.15E-04
7	9.29	3.83E-05	2.44E-05	2.48E-04	7.72E-03	7.97E-03	6.47E-04
8	10.47	4.36E-05	2.78E-05	2.88E-04	8.68E-03	8.91E-03	7.44E-04
9	11.88	4.92E-05	3.14E-05	3.23E-04	9.82E-03	1.01E-02	8.32E-04
10	13.28	5.52E-05	3.51E-05	3.61E-04	1.10E-02	1.13E-02	9.20E-04
11	14.54	6.01E-05	3.82E-05	3.90E-04	1.20E-02	1.23E-02	9.99E-04
12	15.91	6.59E-05	4.19E-05	4.25E-04	1.32E-02	1.35E-02	1.09E-03
13	17.25	7.21E-05	4.58E-05	4.63E-04	1.44E-02	1.48E-02	1.17E-03
14	18.57	7.77E-05	4.95E-05	4.99E-04	1.56E-02	1.60E-02	1.24E-03
15	19.91	8.32E-05	5.32E-05	5.33E-04	1.66E-02	1.70E-02	1.30E-03
16	21.28	8.91E-05	5.69E-05	5.69E-04	1.78E-02	1.83E-02	1.37E-03
17	22.66	9.57E-05	6.10E-05	6.09E-04	1.92E-02	1.97E-02	1.44E-03
18	24.06	1.02E-04	6.52E-05	6.47E-04	2.05E-02	2.10E-02	1.51E-03
Projections	32.00	1.39E-04	8.88E-05	8.60E-04	2.71E-02	2.78E-02	1.92E-03
	48.00	2.14E-04	1.36E-04	1.29E-03	4.04E-02	4.17E-02	2.75E-03

Table 2-14: Unit 1 – Calculated Maximum Fast (E > 1.0 MeV) Neutron Fluence Rate at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	1.20	1.20	2.49E+10	45	-20
2	1.29	2.50	1.63E+10	45	-126
3	1.38	3.88	1.58E+10	45	-126
4	1.33	5.21	1.73E+10	45	-18
5	1.41	6.62	1.40E+10	45	-18
6	1.32	7.94	1.57E+10	45	-74
7	1.34	9.29	1.78E+10	45	-74
8	1.18	10.47	1.63E+10	47	34
9	1.41	11.88	1.68E+10	44	-74
10	1.40	13.28	1.69E+10	45	-18
11	1.26	14.54	1.67E+10	68	-74
12	1.37	15.91	1.79E+10	45	-74
13	1.33	17.25	1.84E+10	45	-74
14	1.33	18.57	1.81E+10	45	-74
15	1.34	19.91	1.69E+10	22	-74
16	1.37	21.28	1.83E+10	47	-74
17	1.38	22.66	2.04E+10	45	-74
18	1.40	24.06	1.93E+10	22	-74
Projections		--	1.93E+10	22	-74
wrt: "with respect to"					

Table 2-15: Unit 1 – Calculated Maximum Fast (E > 1.0 MeV) Neutron Fluence at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	1.20	1.20	9.44E+17	45	-20
2	1.29	2.50	1.58E+18	45	-74
3	1.38	3.88	2.27E+18	45	-74
4	1.33	5.21	2.99E+18	45	-74
5	1.41	6.62	3.62E+18	45	-74
6	1.32	7.94	4.28E+18	45	-74
7	1.34	9.29	5.03E+18	45	-74
8	1.18	10.47	5.62E+18	45	-74
9	1.41	11.88	6.37E+18	45	-74
10	1.40	13.28	7.12E+18	45	-74
11	1.26	14.54	7.77E+18	45	-74
12	1.37	15.91	8.54E+18	45	-74
13	1.33	17.25	9.32E+18	45	-74
14	1.33	18.57	1.01E+19	45	-74
15	1.34	19.91	1.08E+19	45	-74
16	1.37	21.28	1.15E+19	45	-74
17	1.38	22.66	1.24E+19	45	-74
18	1.40	24.06	1.32E+19	45	-74
Projections		32.00	1.75E+19	22	-74
		48.00	2.73E+19	22	-74

Table 2-16: Unit 1 – Calculated Maximum Iron Atom Displacement Rate at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	1.20	1.20	3.94E-11	45	-20
2	1.29	2.50	2.58E-11	45	-126
3	1.38	3.88	2.50E-11	45	-124
4	1.33	5.21	2.74E-11	46	-18
5	1.41	6.62	2.22E-11	45	-18
6	1.32	7.94	2.49E-11	45	-74
7	1.34	9.29	2.82E-11	45	-74
8	1.18	10.47	2.58E-11	47	34
9	1.41	11.88	2.66E-11	44	-74
10	1.40	13.28	2.69E-11	46	-18
11	1.26	14.54	2.59E-11	45	-74
12	1.37	15.91	2.83E-11	45	-74
13	1.33	17.25	2.92E-11	45	-74
14	1.33	18.57	2.86E-11	45	-74
15	1.34	19.91	2.58E-11	22	-74
16	1.37	21.28	2.89E-11	47	-70
17	1.38	22.66	3.22E-11	45	-74
18	1.40	24.06	2.96E-11	47	-74
Projections		--	2.95E-11	22	-74
wrt: "with respect to"					

Table 2-17: Unit 1 – Calculated Maximum Iron Atom Displacements at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	1.20	1.20	1.50E-03	45	-20
2	1.29	2.50	2.51E-03	45	-74
3	1.38	3.88	3.60E-03	45	-74
4	1.33	5.21	4.75E-03	45	-74
5	1.41	6.62	5.74E-03	45	-74
6	1.32	7.94	6.77E-03	45	-74
7	1.34	9.29	7.97E-03	45	-74
8	1.18	10.47	8.91E-03	46	-74
9	1.41	11.88	1.01E-02	45	-74
10	1.40	13.28	1.13E-02	45	-74
11	1.26	14.54	1.23E-02	45	-74
12	1.37	15.91	1.35E-02	45	-74
13	1.33	17.25	1.48E-02	45	-74
14	1.33	18.57	1.60E-02	45	-74
15	1.34	19.91	1.70E-02	45	-74
16	1.37	21.28	1.83E-02	45	-74
17	1.38	22.66	1.97E-02	45	-74
18	1.40	24.06	2.10E-02	46	-74
Projections		32.00	2.78E-02	46	-74
		48.00	4.17E-02	22	-74

Table 2-18: Calculated Fast Neutron Fluence Rate at Selected Azimuthal Locations of the Cladding/base metal Interface 74 cm Below the Core Midplane

Cycle	EFPY	Fluence Rate (n/cm^2-s)						
		0°	15°	22°	30°	45°	67°	90°
18	24.06	1.12E+10	1.68E+10	1.93E+10	1.53E+10	1.86E+10	1.89E+10	1.12E+10
Projections		1.08E+10	1.65E+10	1.93E+10	1.52E+10	1.71E+10	1.88E+10	1.08E+10

Cycle	EFPY	Fluence (n/cm^2)						
		0°	15°	22°	30°	45°	67°	90°
18	24.06	7.08E+18	1.07E+19	1.27E+19	1.08E+19	1.32E+19	1.24E+19	7.08E+18
-	32.00	9.79E+18	1.48E+19	1.75E+19	1.46E+19	1.75E+19	1.71E+19	9.79E+18
-	48.00	1.52E+19	2.31E+19	2.73E+19	2.23E+19	2.61E+19	2.66E+19	1.53E+19

Cycle	EFPY	Iron Atom Displacement Rate (dpa/s)						
		0°	15°	22°	30°	45°	67°	90°
18	24.06	1.72E-11	2.58E-11	2.95E-11	2.38E-11	2.94E-11	2.88E-11	1.73E-11
Projections		1.67E-11	2.53E-11	2.95E-11	2.37E-11	2.71E-11	2.88E-11	1.67E-11

Cycle	EFPY	Iron Atom Displacements (dpa)						
		0°	15°	22°	30°	45°	67°	90°
18	24.06	1.10E-02	1.64E-02	1.94E-02	1.67E-02	2.10E-02	1.89E-02	1.10E-02
-	32.00	1.51E-02	2.27E-02	2.68E-02	2.27E-02	2.78E-02	2.61E-02	1.52E-02
-	48.00	2.36E-02	3.55E-02	4.17E-02	3.46E-02	4.14E-02	4.06E-02	2.36E-02

2.2.2 Watts Bar Unit 2 Transport Analysis

Selected results from the neutron transport analyses for Watts Bar Unit 2 are provided in Table 2-19 through Table 2-31. In Table 2-19, the calculated fast neutron (E > 1.0 MeV) fluence rates at the radial and azimuthal center of the surveillance capsule positions at core midplane are presented. Integrated fast neutron fluence at the center of the surveillance capsules is presented in Table 2-20. 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, representative of the exposure of the material specimens, establish the calculated exposure of the surveillance capsules to-date and projected into the future based on the equilibrium 2,496 TPBAR core.

Updated lead factors for the surveillance capsules are provided in Table 2-21. In Table 2-21, the lead factor for the capsule that has been withdrawn from the reactor (U) was based on the calculated fluence values for the irradiation period corresponding to the time of withdrawal of Capsule U after Cycle 2. Capsules V, W, X, Y and Z remain in the reactor.

Neutron exposure data pertinent to selected pressure vessel weld materials are given in Table 2-24 and Table 2-25 for fast neutron (E > 1.0 MeV) fluence rate and fluence. Similar data for pressure vessel weld materials are provided in Table 2-26 and Table 2-27 for dpa/s and dpa. Neutron exposure data pertinent to selected pressure vessel forgings are given in Table 2-28 and Table 2-29 for fast neutron (E > 1.0 MeV) fluence rate and fluence. Similar data for pressure vessel forgings are provided in Table 2-30 and Table 2-31 for dpa/s and dpa. The data presented represent the maximum neutron exposure experienced by the RPV materials that will constitute inputs to the reactor vessel integrity analysis. The reported data considers both the inner and outer radius of the RPV base metal, and accounts for the possibility of higher neutron exposure values occurring on the outer surface of the RPV (as compared to the inner surface) for materials that are distant from the active core. In each case, the data are provided for each operating cycle. For any given fuel cycle, the location of the maximum neutron exposure rate may or may not coincide with the location of the maximum neutron exposure.

Neutron exposure data specific to the cladding/base metal interface is provided in Table 2-32 through Table 2-36. The axial and azimuthal maximum of the fast neutron fluence rate and fluence at the cladding/base metal interface are given in Table 2-32 and Table 2-33. The axial and azimuthal maximum of the iron atom displacement rate and iron atom displacements at the cladding/base metal interface are given in Table 2-34 and Table 2-35. Table 2-36 presents the parameters from Table 2-32 through Table 2-35 at selected azimuthal locations about the cladding/base metal interface.

These data tabulations include both plant-specific and fuel-cycle-specific calculated neutron exposures at the end of Cycle 4 and projections to 32 and 48 EFPY. Projections of neutron exposure beyond the end of Cycle 4 are based on the example core loading pattern and expected operating characteristics for the equilibrium 2496 TPBAR core and the rated thermal power of 3459 MWt.

Table 2-19: Unit 2 – Calculated Fast Neutron (E > 1.0 MeV) Fluence Rate at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Fluence Rate (n/cm^2-s)		
		Dual Capsule		Single Capsule
		34°	31.5°	34°
1	0.74	1.13E+11	9.37E+10	1.12E+11
2	2.00	8.85E+10	7.38E+10	8.79E+10
3	3.35	8.27E+10	7.00E+10	8.21E+10
4	4.52	9.19E+10	7.67E+10	9.13E+10
Projections	--	8.71E+10	7.53E+10	8.65E+10

Table 2-20: Unit 2 – Calculated Fast Neutron (E > 1.0 MeV) Fluence at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Fluence (n/cm^2)		
		Dual Capsule		Single Capsule
		34° (Capsule X)	31.5° (Capsules V and Y)	34° (Capsules W and Z)
1	0.74	2.63E+18	2.19E+18	2.62E+18
2	2.00	6.14E+18 (U)	5.11E+18	6.10E+18
3	3.35	9.68E+18	8.11E+18	9.62E+18
4	4.52	1.31E+19	1.10E+19	1.30E+19
Projections	32.00	8.86E+19	7.62E+19	8.80E+19
	48.00	1.33E+20	1.14E+20	1.32E+20

Table 2-21: Unit 2 – Surveillance Capsule Lead Factors

Cycle	Cumulative Operating Time (EFPY)	Lead Factors		
		Dual Capsule		Single Capsule
		34˚ (Capsule X)	31.5˚ (Capsules V and Y)	34˚ (Capsules W and Z)
1	0.74	4.87	4.05	4.84
2	2.00	4.80 (Capsule U)	4.00	4.77
3	3.35	4.81	4.03	4.78
4	4.52	4.83	4.04	4.80
Projections	6.8	4.90	4.15	4.87
	13.8	4.71	4.03	4.68
	32.00	4.58	3.94	4.55
	48.00	4.55	3.92	4.52

Table 2-22: Unit 2 – Calculated Iron Atom Displacement Rate at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacement Rate (dpa/s)		
		Dual Capsule		Single Capsule
		34°	31.5°	34°
1	0.74	2.30E-10	1.88E-10	2.29E-10
2	2.00	1.79E-10	1.47E-10	1.78E-10
3	3.35	1.67E-10	1.39E-10	1.66E-10
4	4.52	1.86E-10	1.53E-10	1.85E-10
Projections	--	1.76E-10	1.50E-10	1.75E-10

Table 2-23: Unit 2 – Calculated Iron Atom Displacements at the Geometric Center of the Surveillance Capsules

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacements (dpa)		
		Dual Capsule		Single Capsule
		34° (Capsule X)	31.5° (Capsules V and Y)	34° (Capsules W and Z)
1	0.74	5.36E-03	4.39E-03	5.34E-03
2	2.00	1.24E-02 (U)	1.02E-02	1.24E-02
3	3.35	1.96E-02	1.62E-02	1.95E-02
4	4.52	2.65E-02	2.18E-02	2.64E-02
Projections	32.00	1.79E-01	1.52E-01	1.78E-01
	48.00	2.68E-01	2.27E-01	2.67E-01

Table 2-24: Unit 2 – Calculated Maximum Fast Neutron Fluence Rate in Reactor Welds

Cycle	EFPY	Fluence Rate (n/cm^2-s)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	0.74	3.98E+07	1.86E+07	6.45E+08	2.21E+10	3.39E+09	7.37E+06
2	2.00	3.85E+07	1.83E+07	5.90E+08	1.80E+10	2.84E+09	6.22E+06
3	3.35	4.11E+07	1.92E+07	6.15E+08	1.67E+10	2.52E+09	5.52E+06
4	4.52	3.89E+07	1.77E+07	6.25E+08	1.85E+10	2.85E+09	6.25E+06
Projections		4.48E+07	2.10E+07	6.38E+08	1.84E+10	2.62E+09	6.19E+06

Table 2-25: Unit 2 – Calculated Maximum Fast Neutron Fluence in Reactor Welds

Cycle	EFPY	Fluence (n/cm^2)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	0.74	9.29E+14	4.34E+14	1.51E+16	5.16E+17	7.91E+16	1.72E+14
2	2.00	2.46E+15	1.16E+15	3.84E+16	1.23E+18	1.91E+17	4.18E+14
3	3.35	4.21E+15	1.98E+15	6.48E+16	1.94E+18	3.00E+17	6.55E+14
4	4.52	5.65E+15	2.64E+15	8.79E+16	2.63E+18	4.05E+17	8.86E+14
Projections	32.00	4.45E+16	2.08E+16	6.41E+17	1.83E+19	2.67E+18	6.26E+15
	48.00	6.72E+16	3.14E+16	9.63E+17	2.76E+19	3.99E+18	9.38E+15

Table 2-26: Unit 2 – Calculated Maximum Iron Atom Displacement Rate in Reactor Welds

Cycle	EFPY	Iron Atom Displacement Rate (dpa/s)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	0.74	1.67E-13	1.05E-13	1.06E-12	3.53E-11	5.40E-12	4.75E-14
2	2.00	1.42E-13	8.91E-14	9.67E-13	2.87E-11	4.52E-12	3.97E-14
3	3.35	1.42E-13	8.72E-14	1.01E-12	2.65E-11	4.01E-12	3.51E-14
4	4.52	1.54E-13	9.53E-14	1.03E-12	2.95E-11	4.53E-12	3.99E-14
Projections		1.62E-13	1.01E-13	1.04E-12	2.82E-11	4.17E-12	3.97E-14

Table 2-27: Unit 2 – Calculated Maximum Iron Atom Displacements in Reactor Welds

Cycle	EFPY	Iron Atom Displacements (dpa)					
		Cold Leg to Upper Shell 06	Hot Leg to Upper Shell 06	Upper Shell 06 to Intermediate Shell 05	Intermediate Shell 05 to Lower Shell 04	Lower Shell 04 to Bottom Head Ring 03	Bottom Head Ring 03 to Bottom Head Peel 02
1	0.74	3.90E-06	2.44E-06	2.48E-05	8.23E-04	1.26E-04	1.11E-06
2	2.00	9.52E-06	5.97E-06	6.31E-05	1.96E-03	3.05E-04	2.68E-06
3	3.35	1.56E-05	9.71E-06	1.06E-04	3.09E-03	4.77E-04	4.18E-06
4	4.52	2.13E-05	1.32E-05	1.44E-04	4.18E-03	6.44E-04	5.66E-06
Projections	32.00	1.61E-04	1.01E-04	1.05E-03	2.81E-02	4.26E-03	4.01E-05
	48.00	2.43E-04	1.52E-04	1.58E-03	4.23E-02	6.37E-03	6.01E-05

Table 2-28: Unit 2 – Calculated Maximum Fast Neutron Fluence Rate in Reactor Forgings

Cycle	EFPY	Fluence Rate (n/cm^2-s)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	0.74	2.74E+07	1.41E+07	5.38E+08	2.22E+10	2.32E+10	3.06E+09
2	2.00	2.69E+07	1.28E+07	4.96E+08	1.83E+10	1.87E+10	2.57E+09
3	3.35	2.86E+07	1.35E+07	5.18E+08	1.70E+10	1.72E+10	2.29E+09
4	4.52	2.67E+07	1.29E+07	5.22E+08	1.88E+10	1.90E+10	2.57E+09
Projections		3.10E+07	1.46E+07	5.36E+08	1.87E+10	1.94E+10	2.36E+09

Table 2-29: Unit 2 – Calculated Maximum Fast Neutron Fluence in Reactor Forgings

Cycle	EFPY	Fluence (n/cm^2)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	0.74	6.40E+14	3.29E+14	1.26E+16	5.17E+17	5.41E+17	7.15E+16
2	2.00	1.70E+15	8.36E+14	3.22E+16	1.24E+18	1.28E+18	1.73E+17
3	3.35	2.93E+15	1.42E+15	5.44E+16	1.97E+18	2.01E+18	2.71E+17
4	4.52	3.91E+15	1.89E+15	7.37E+16	2.66E+18	2.71E+18	3.66E+17
Projections	32.00	3.08E+16	1.45E+16	5.38E+17	1.87E+19	1.93E+19	2.41E+18
	48.00	4.64E+16	2.19E+16	8.09E+17	2.82E+19	2.91E+19	3.61E+18

Table 2-30: Unit 2 – Calculated Maximum Iron Atom Displacement Rate in Reactor Forgings

Cycle	EFPY	Iron Atom Displacement Rate (dpa/s)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	0.74	1.52E-13	9.74E-14	8.48E-13	3.53E-11	3.68E-11	4.87E-12
2	2.00	1.29E-13	8.29E-14	7.82E-13	2.91E-11	2.96E-11	4.08E-12
3	3.35	1.29E-13	8.12E-14	8.17E-13	2.69E-11	2.72E-11	3.64E-12
4	4.52	1.40E-13	8.87E-14	8.24E-13	2.99E-11	3.01E-11	4.09E-12
Projections		1.47E-13	9.42E-14	8.44E-13	2.86E-11	2.96E-11	3.76E-12

Table 2-31: Unit 2 – Calculated Maximum Iron Atom Displacements in Reactor Forgings

Cycle	EFPY	Iron Atom Displacements (dpa)					
		Cold Leg	Hot Leg	Upper Shell 06	Intermediate Shell 05	Lower Shell 04	Bottom Head Ring 03
1	0.74	3.54E-06	2.27E-06	1.98E-05	8.23E-04	8.59E-04	1.14E-04
2	2.00	8.66E-06	5.56E-06	5.08E-05	1.97E-03	2.03E-03	2.75E-04
3	3.35	1.42E-05	9.04E-06	8.58E-05	3.13E-03	3.19E-03	4.31E-04
4	4.52	1.93E-05	1.23E-05	1.16E-04	4.23E-03	4.30E-03	5.82E-04
Projections	32.00	1.47E-04	9.40E-05	8.48E-04	2.86E-02	2.95E-02	3.84E-03
	48.00	2.21E-04	1.42E-04	1.27E-03	4.30E-02	4.45E-02	5.74E-03

Table 2-32: Unit 2 – Calculated Maximum Fast (E > 1.0 MeV) Neutron Fluence Rate at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	0.74	0.74	2.32E+10	45	-22
2	1.26	2.00	1.87E+10	45	-74
3	1.36	3.35	1.72E+10	45	-74
4	1.17	4.52	1.90E+10	46	-18
Projections		--	1.94E+10	22	-74
wrt: "with respect to"					

Table 2-33: Unit 2 – Calculated Maximum Fast (E > 1.0 MeV) Neutron Fluence at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	0.74	0.74	5.41E+17	45	-22
2	1.26	2.00	1.28E+18	45	-72
3	1.36	3.35	2.01E+18	45	-74
4	1.17	4.52	2.71E+18	45	-74
Projections		32.00	1.93E+19	22	-74
		48.00	2.91E+19	22	-74

Table 2-34: Unit 2 – Calculated Maximum Iron Atom Displacement Rate at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence Rate (n/cm^2-s)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	0.74	0.74	3.68E-11	45	-22
2	1.26	2.00	2.96E-11	45	-74
3	1.36	3.35	2.72E-11	45	-74
4	1.17	4.52	3.01E-11	46	-20
Projections		--	2.96E-11	22	-74
wrt: "with respect to"					

Table 2-35: Unit 2 – Calculated Maximum Iron Atom Displacements at the Reactor Pressure Vessel Cladding/Base Metal Interface

Cycle	Cycle Length (EFPY)	Total Time (EFPY)	Fluence (n/cm^2)	Azimuth (degrees)	Location wrt Core Midplane (cm)
1	0.74	0.74	8.59E-04	45	-22
2	1.26	2.00	2.03E-03	45	-70
3	1.36	3.35	3.19E-03	45	-72
4	1.17	4.52	4.30E-03	45	-72
Projections		32.00	2.95E-02	22	-74
		48.00	4.45E-02	22	-74

Table 2-36: Unit 2 – Calculated Fast Neutron Fluence Rate at Selected Azimuthal Locations of the cladding/base metal Interface 74 cm Below the Core Midplane

Cycle	EFPY	Fluence Rate (n/cm^2-s)						
		0°	15°	22°	30°	45°	67°	90°
4	4.52	9.87E+09	1.51E+10	1.76E+10	1.47E+10	1.88E+10	1.73E+10	9.96E+09
Projections		1.08E+10	1.65E+10	1.94E+10	1.53E+10	1.71E+10	1.89E+10	1.09E+10

Cycle	EFPY	Fluence (n/cm^2)						
		0°	15°	22°	30°	45°	67°	90°
4	4.52	1.40E+18	2.11E+18	2.50E+18	2.12E+18	2.71E+18	2.48E+18	1.41E+18
-	32.00	1.08E+19	1.65E+19	1.93E+19	1.54E+19	1.76E+19	1.89E+19	1.08E+19
-	48.00	1.63E+19	2.48E+19	2.92E+19	2.31E+19	2.62E+19	2.84E+19	1.63E+19

Cycle	EFPY	Iron Atom Displacement Rate (dpa/s)						
		0°	15°	22°	30°	45°	67°	90°
4	4.52	1.53E-11	2.31E-11	2.69E-11	2.29E-11	2.99E-11	2.65E-11	1.54E-11
Projections		1.68E-11	2.54E-11	2.96E-11	2.38E-11	2.72E-11	2.89E-11	1.68E-11

Cycle	EFPY	Iron Atom Displacements (dpa)						
		0°	15°	22°	30°	45°	67°	90°
4	4.52	2.16E-03	3.23E-03	3.82E-03	3.31E-03	4.29E-03	3.78E-03	2.18E-03
-	32.00	1.67E-02	2.52E-02	2.95E-02	2.39E-02	2.79E-02	2.88E-02	1.68E-02
-	48.00	2.52E-02	3.80E-02	4.45E-02	3.59E-02	4.16E-02	4.34E-02	2.52E-02

2.2.3 Power Distribution in the 2496 TPBAR Core

As discussed in Sections 2.2.1 and 2.2.2, the projections of future neutron exposure were based upon an equilibrium fuel cycle containing 2,496 TPBARs in the core. Additional TPBAR inserts were incorporated in the interior fuel assemblies as well as in fuel assemblies on the core periphery around the 45˚ azimuth. The net effect is an increase in the number of fuel assembly positions that would hold TPBAR inserts during a typical fuel cycle. The additional TPBARs in the equilibrium core further distribute power production to fuel assemblies that do not hold TPBAR inserts, including some fuel assemblies on the core periphery.

This power distribution effect was tempered by the sheer number of additional inserts and by the careful selection of the insert positions throughout the core. This effect is illustrated in Figure 2-5, which shows a comparison of the fuel assembly relative powers in the equilibrium 2496 TPBAR core with the fuel assembly relative powers of Unit 2 Cycle 7 from WCAP-18532-NP. Cycle 7 from WCAP-18532-NP accounted for TPBARs in the fuel as well as a power uprate from 3411 MWt to 3459 MWt and was used as the fluence projection cycle.

The net effect was such that the fluence rates observed near the surveillance capsules were reduced by just under 5 percent. Likewise, the fluence rates near the 0˚ and 90˚ cardinal axes were slightly increased. As in the previous analyses, the peak reactor vessel fluence occurred at the 22˚ azimuth. The distribution and quantity of the TPBAR inserts throughout the core was such that the peak fluence at the pressure vessel cladding/base metal interface showed a decrease of about 2 percent and was roughly comparable to previous analyses.

2.3 NEUTRON DOSIMETRY

The validity of the calculated neutron exposures reported in Section 2.2 is demonstrated by a direct comparison against measured sensor reaction rates and a least-squares evaluation performed for each of the capsule dosimetry sets for each unit. However, because the neutron dosimetry measurement data merely serve to validate the calculated results, only the direct comparisons of measured-to-calculated results for surveillance capsules analyzed are provided in this section. For completeness, an assessment based on both direct and least-squares evaluation comparisons for both units is documented in Appendix A.

As stated in Section 2.1, the transport analyses described herein were performed consistent with the NRC-approved methodology described in WCAP-18124-NP-A [Ref. 11], with additional analytical requirements associated with the extended beltline analysis as detailed in WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A [Ref. 20]. The Unit 1 dosimetry comparisons for capsules U, W, X and Z were reexamined to validate the application of the state-of-the-art, approved methodology described in WCAP-18124-NP-A and its Supplement 1 consistent with the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. 10].

For Watts Bar Unit 2, the direct comparison of measured versus calculated fast neutron threshold reaction rates for the sensors from Capsule U, which was withdrawn from Watts Bar Unit 2 at the end of the 2nd fuel cycle, was reported in WCAP-18518-NP. The dosimetry comparison for Capsule U was revisited to provide consistency between the dosimetry data and the model with the additional analytical requirements (associated with the extended beltline analysis as detailed in WCAP-18124-NP-A, Rev. 0 Supplement 1-NP-A) and geometry refinements applied.

2.4 CALCULATIONAL UNCERTAINTIES

The uncertainty associated with the calculated neutron exposure of the surveillance capsules and reactor pressure vessel is based on the recommended approach provided in RG 1.190. In particular, the qualification of the methodology was carried out in the following four stages:

1. **Simulator Benchmark Comparisons:** Comparisons of calculations with measurements from simulator benchmarks, including the pool critical assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL) and the VENUS-1 experiment.

2. **Operating Reactor and Calculational Benchmarks:** Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H.B. Robinson power reactor benchmark experiment. Also considered are comparisons of calculations performed with RAPTOR-M3G to results published in the NRC fluence calculation benchmark.

3. **Analytic Uncertainty Analysis:** An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant-specific transport calculations used in the neutron exposure assessments.

4. **Plant-Specific Benchmarking:** Comparisons of the plant-specific calculations with all available dosimetry results from the Watts Bar Units 1 & 2 surveillance program.

The first phase of the methods qualification (simulator benchmark comparisons) addressed the adequacy of basic transport calculation and dosimetry evaluation techniques and associated cross-sections. This phase, however, did not test the accuracy of commercial core neutron source calculations nor did it address uncertainties in operational or geometric variables that impact power reactor calculations. The second phase of the qualification (operating reactor and calculational benchmark comparisons) addressed uncertainties in these additional areas that are primarily methods-related and would tend to apply generically to all fast neutron exposure evaluations. The third phase of the qualification (analytical sensitivity study) identified the potential uncertainties introduced into the overall evaluation due to calculational methods approximations, as well as to a lack of knowledge relative to various plant-specific input parameters. The overall calculational uncertainty applicable to the Watts Bar Units 1 & 2 analyses was established from results of these three phases of the methods qualification.

The fourth phase of the uncertainty assessment (comparisons with plant-specific measurements) was used solely to demonstrate the validity of the transport calculations and to confirm the uncertainty estimates associated with the analytical results. The comparison was used only as a check and was not used in any way to modify the calculated surveillance capsule and pressure vessel neutron exposures previously described in Section 2.2. As such, the validation of the analytical model based on the measured plant dosimetry is completely described in Appendix A.

The following summarizes the uncertainties developed from the first three phases of the methodology qualification. Additional information pertinent to these evaluations is provided in Westinghouse Report WCAP-18124-NP-A [Ref. 11].

Description	Capsule and Vessel IR
Simulator Benchmark Comparisons	3%
H.B. Robinson Benchmark Comparisons	5%
Analytical Sensitivity Studies	11%
Additional Uncertainty for Factors not Explicitly Evaluated	5%
Net Calculational Uncertainty	**13%**

The net calculational uncertainty was determined by combining the individual components in quadrature. Therefore, the resultant uncertainty was treated as random, and no systematic bias was applied to the analytical results. The plant-specific measurement comparisons described in Appendix A support these uncertainty assessments for Watts Bar Unit 1 and Unit 2.

The NRC-issued Safety Evaluation for WCAP-18124-NP appears in Section A of WCAP-18124-NP-A [Ref. 11]. The NRC identified two “Limitations and Conditions” associated with the application of RAPTOR-M3G and FERRET, which are reproduced here for convenience:

1. Applicability of WCAP-18124-NP, Revision 0 is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to the response parameters of interest (e.g. pressure-temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the

method to components including, but not limited to, the RPV upper circumferential weld and the reactor coolant system inlet and outlet nozzles and reactor vessel internal components.

2. Least-squares adjustment is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the discrepancy should be disqualified.

Limitation #1 regarding the applicability of the methodology to the RPV extended beltline, which includes materials such as the RPV upper circumferential weld and the reactor coolant system inlet and outlet nozzles, was generically addressed in WCAP-18124-NP-A Revision 0 Supplement 1-NP-A Revision 0 [Reference 20]. The update to Limitation #1 allows the fluence determination methodology to be used for these materials. The transport analysis results described herein meet the geometric and analytical conditions enumerated in Section 7 of WCAP-18124-NP-A Revision 0, Supplement 1-NP-A . Limitation #1 regarding applicability of the methodology to the reactor vessel internal components continues to apply.

Limitation #2 applies in situations where the least-squares analysis is used to adjust the calculated values of neutron exposure. In the results reported in Appendix A, the least-squares analysis is provided only as a supplemental check on the results of the dosimetry evaluation. The least-squares analysis was *not* used to modify the calculated surveillance capsule or reactor pressure vessel neutron exposure. Therefore, Limitation #2 does not apply.

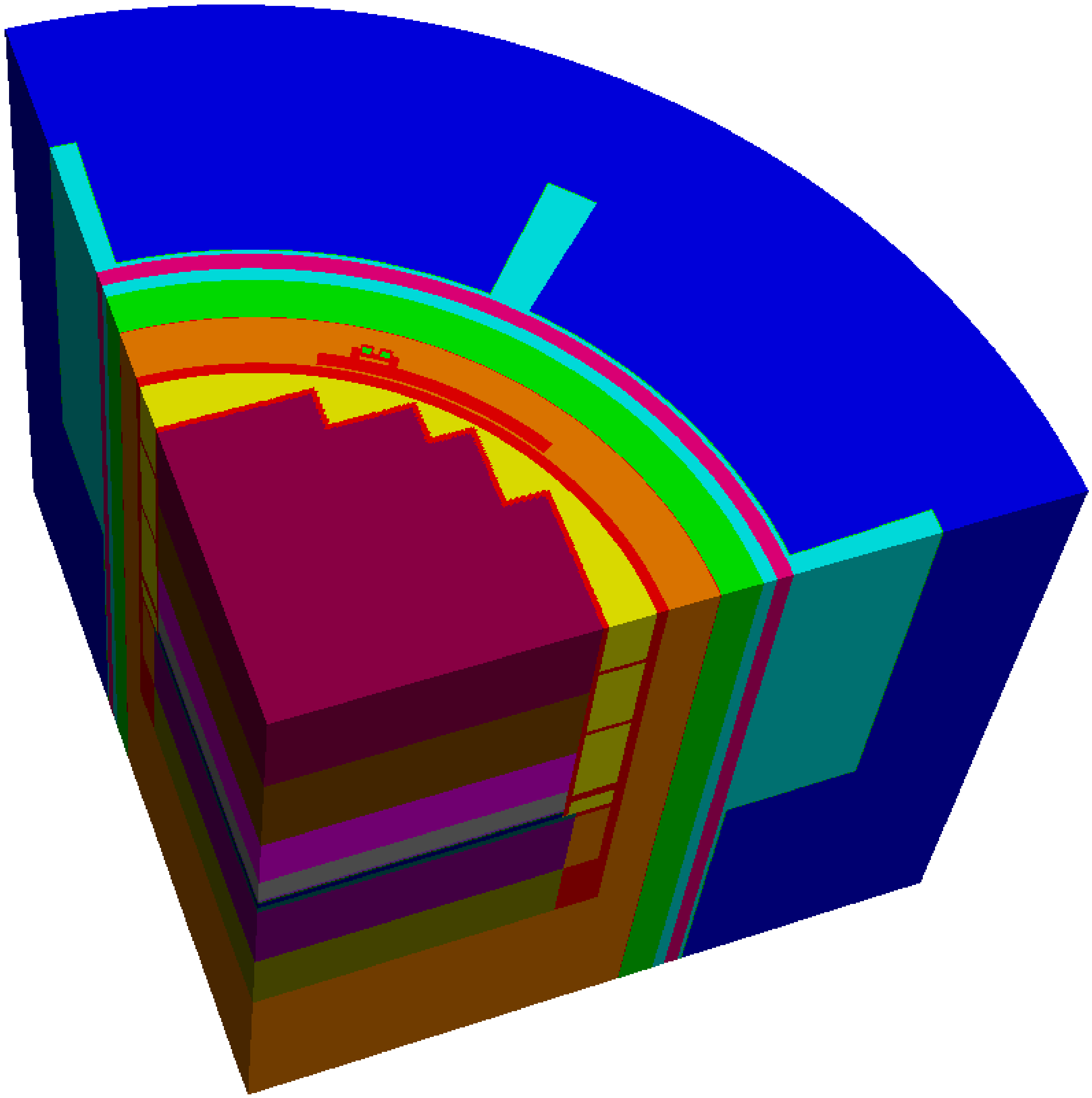

Figure 2-1 Three-Dimensional View of the Reactor Geometry Clipped at the Core Midplane Dual Surveillance Capsule Configuration

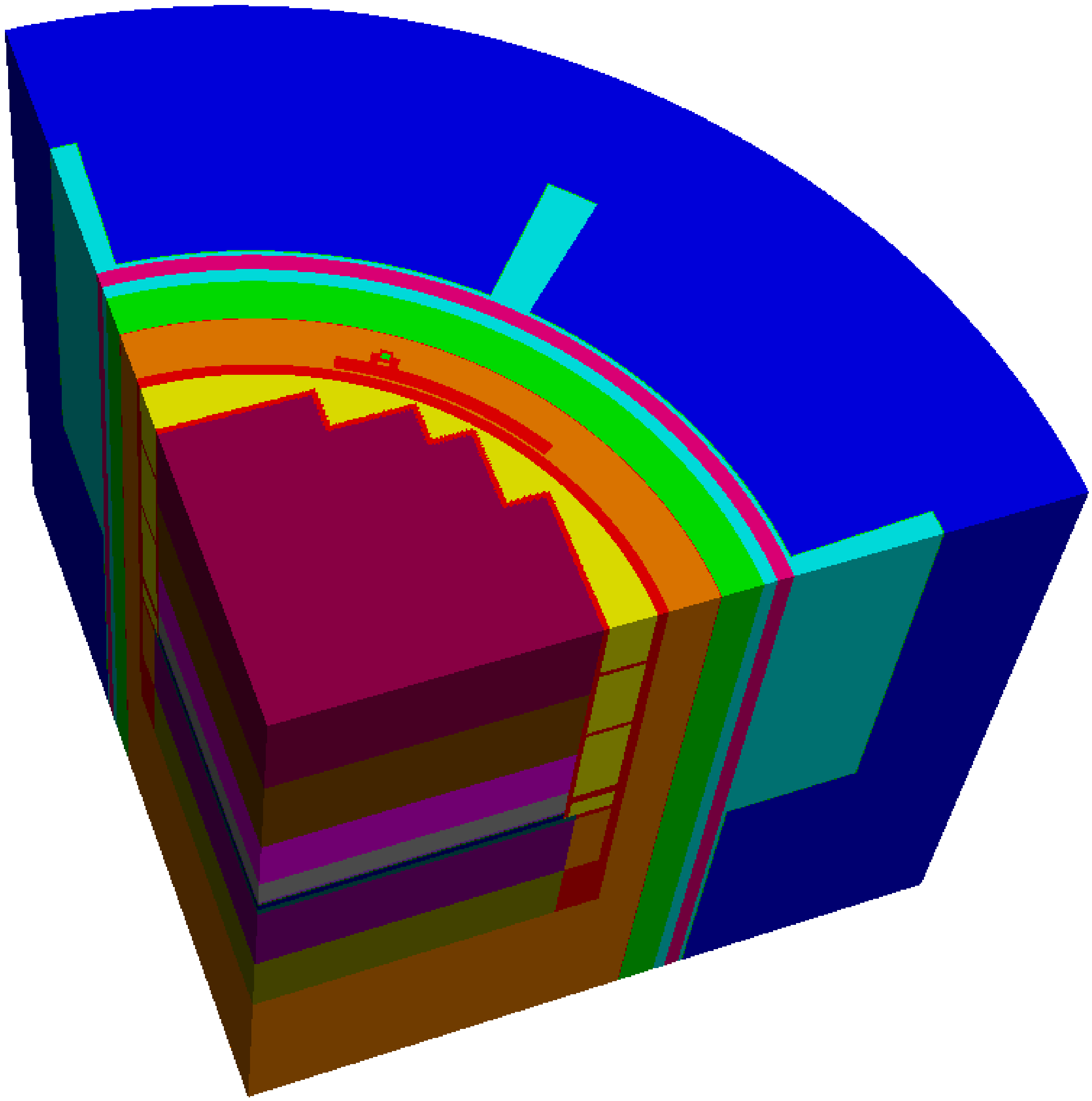

Figure 2-2 Three-Dimensional View of the Reactor Geometry Clipped at the Core Midplane Single Surveillance Capsule Configuration

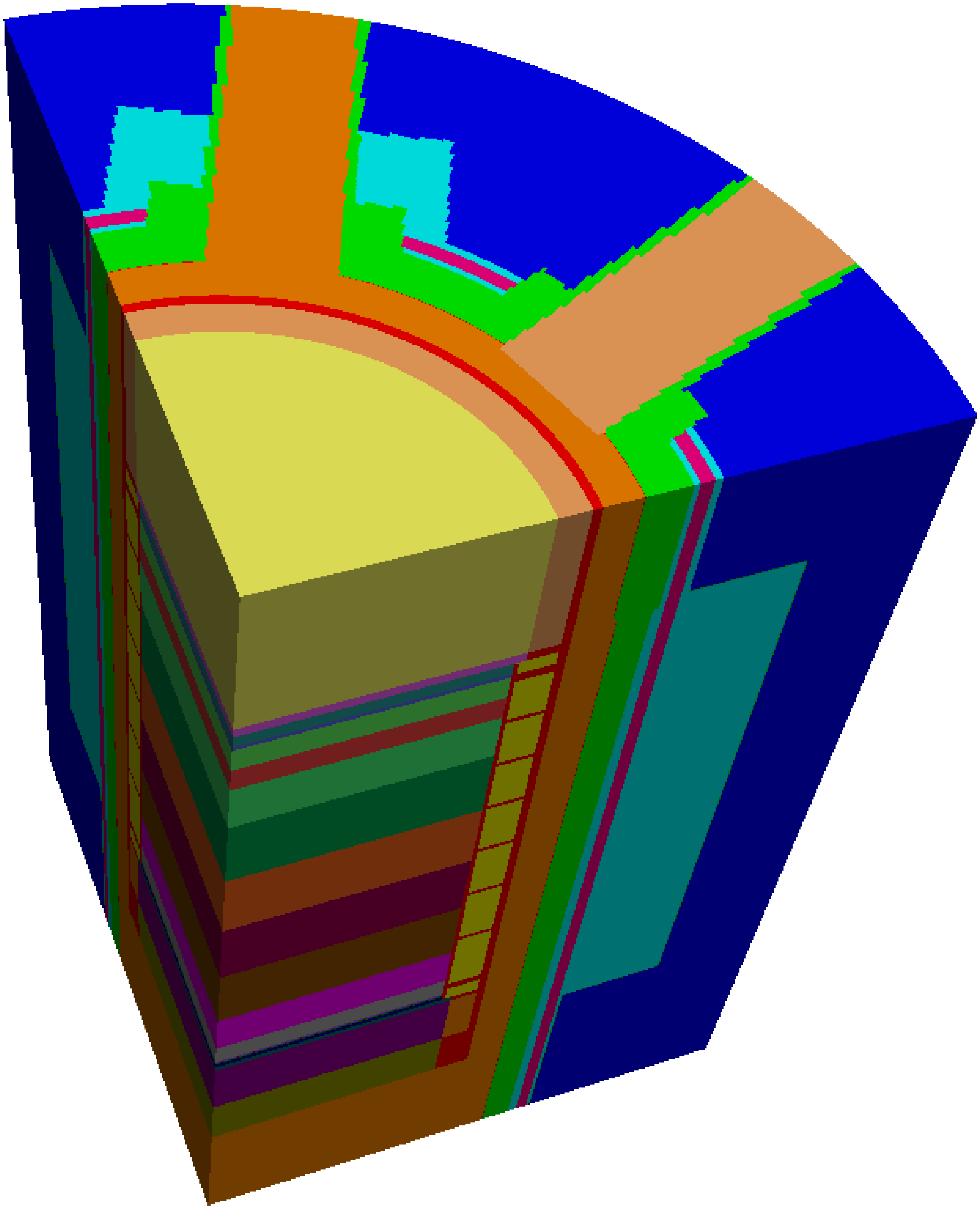

Figure 2-3 Three-Dimensional View of the Reactor Geometry from the Top of the Model (Midplane of the Inlet and Outlet Nozzles)

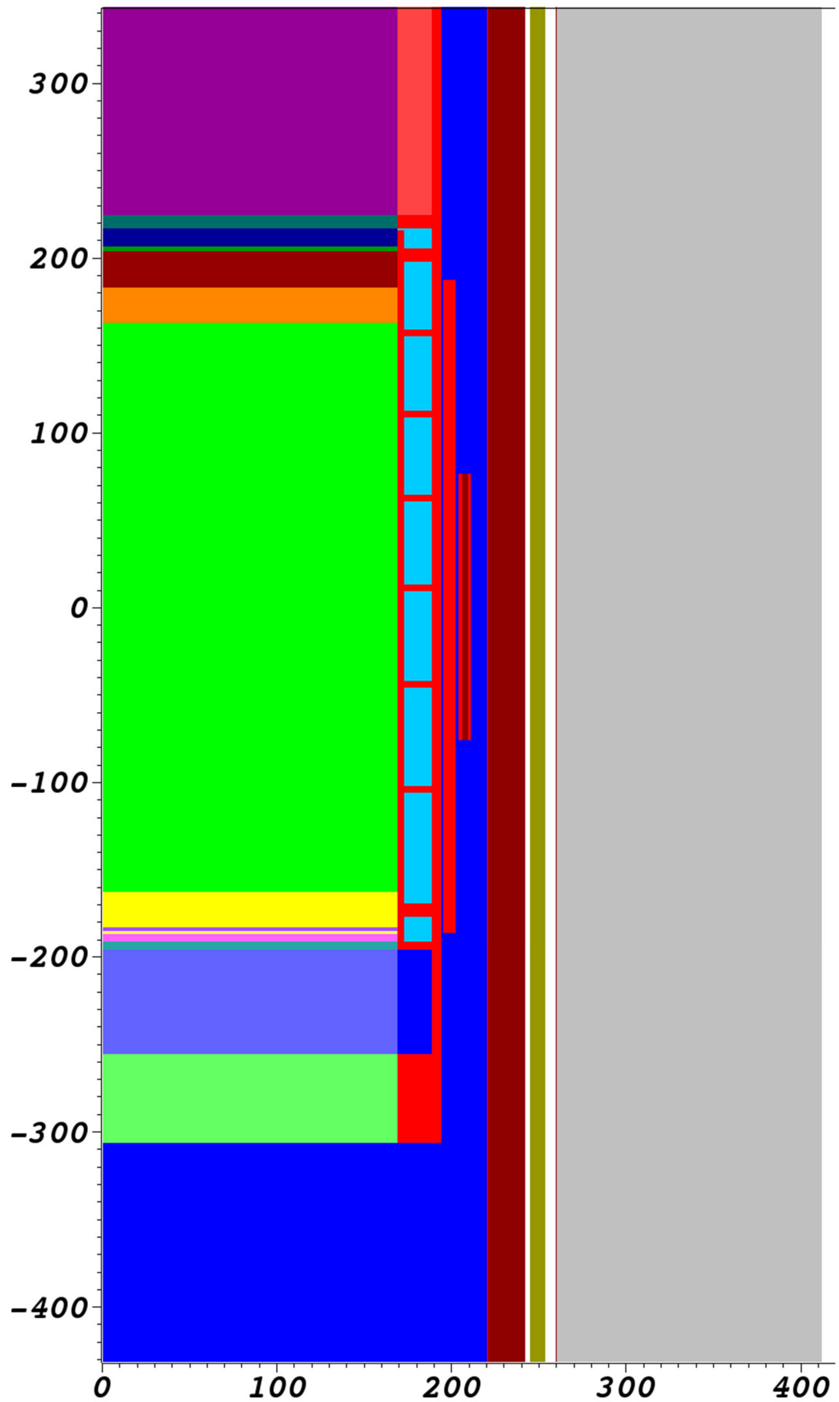

Figure 2-4 Watts Bar Model Section View of the Reactor Geometry at the First Octant Equivalent 34.0° Azimuthal Angle

1.046	0.994	0.968	0.979				
0.923	0.918	0.942	0.998	0.992	0.917		
0.949	0.915	1.014	0.990	0.920	0.844	0.917	
0.915	1.019	1.037	1.054	0.991	0.920	0.989	
1.038	1.048	1.086	1.075	1.054	0.989	0.995	0.977
1.039	1.077	1.038	1.084	1.037	1.013	0.940	0.966
1.095	1.112	1.077	1.048	1.022	0.916	0.915	0.986
1.155	1.094	1.039	1.038	0.915	0.949	0.923	1.046

Figure 2-5 Ratio of the Relative Fuel Assembly Power in the Equilibrium 2496 TPBAR Core to the Previous Fluence Projection for Watts Bar Unit 2

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for RVI are specified in 10 CFR 50, Appendix G [Ref. 5] and 10 CFR 50.61 [Ref. 4]. The beltline region of the reactor vessel is defined as the following in 10 CFR 50, Appendix G:

> *"… the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Watts Bar Units 1 & 2 beltline materials initially included the Intermediate Shell Forging 05, Lower Shell Forging 04, and the Intermediate to Lower Shell Circumferential Weld Seam W05. However, as described in NRC RIS 2014-11 [Ref. 7], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered to experience neutron embrittlement. The additional materials that exceed this fluence threshold are referred to as the "extended beltline" materials and are evaluated to ensure that the applicable neutron embrittlement effects are considered.

For Watts Bar Units 1 & 2, the extended beltline materials include Upper Shell Forging 06, Bottom Head Ring 03, and the circumferential welds connecting these forgings to the intermediate and lower shell forgings, respectively. (Note that for reactor vessel welds, the terms "girth" and "circumferential" are used interchangeably; herein, these welds shall be referred to as circumferential welds.) The fluence for both the inlet/outlet nozzle to upper shell welds are less than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at 32 and 48 EFPY for Watts Bar Units 1 & 2. Therefore, the materials of the inlet/outlet nozzle forgings and the associated welds to the upper shell do not need to be considered in the extended beltline evaluations. Figure 3-1 provides a schematic of the RPV which identifies the beltline and extended beltline regions.

A summary of the best-estimate copper (Cu) and nickel (Ni) contents in units of weight percent (wt%), as well as initial RT_{NDT}, σ_I, and USE values, for the Watts Bar Units 1 & 2 reactor vessel beltline and extended beltline materials are provided in Tables 3-1 and 3-2, respectively. Note, although not considered to be extended beltline materials, the inlet/outlet nozzle forgings are included in these tables. The impact of the inlet/outlet nozzle forgings were considered for Watts Bar Unit 2 in WCAP-18191-NP [Ref. 2] for their effect on the P-T limit curves per RIS 2014-11 [Ref. 7], thus; these material properties are included here for completeness. However, embrittlement does not need to be considered and the inlet/outlet nozzle forgings are excluded from subsequent sections of this report.

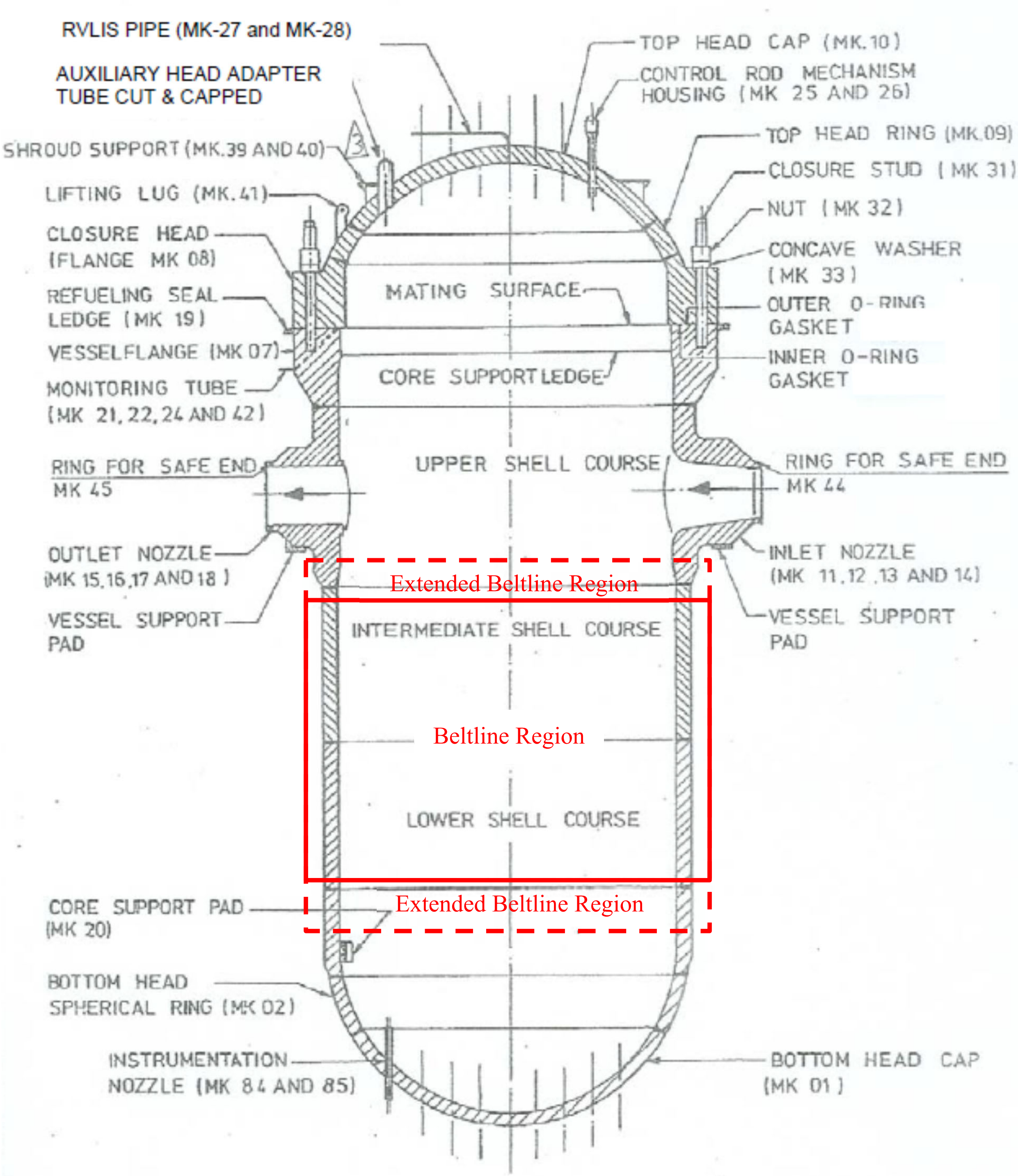

Figure 3-1 RPV Base Metal Material Identifications for Watts Bar Units 1 & 2

Note: Beltline and extended beltline regions are approximate and meant for illustrational purposes only.

Table 3-1 Watts Bar Unit 1 Reactor Vessel Beltline, Extended Beltline, and Surveillance Material Properties and Chemistry[(a)]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Reactor Vessel Beltline Materials									
Intermediate Shell Forging 05	527536	- - -	0.16	0.80	0.73	0.012	47	0	62[(e)]
Lower Shell Forging 04	528522	- - -	0.08	0.83	0.69	0.006	5	0	111[(e)]
Intermediate to Lower Shell Circumferential Weld Seam W05[(b)]	895075	LW320 (P46)	0.04	0.73	1.88	0.013	-43	0	134[(e)]
Reactor Vessel Extended Beltline Materials[(b)]									
Upper Shell Forging 06	411595	- - -	0.12	0.87	0.66	0.005	-22	0	99
Bottom Head Ring 03	528170	- - -	0.06	0.86	0.72	0.009	-40	0	105
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	LW320 (P23)	0.03	0.75	1.97	0.009	10[(c)]	0	98[(d)]
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	899680	LW320 (P23)	0.03	0.75	1.97	0.009	10[(c)]	0	98[(d)]
Reactor Vessel Non-Beltline Materials[(b)]									
Inlet Nozzle #11	527963	- - -	0.11	0.77	0.78	0.011	-4	0	> 73[(g)]
Inlet Nozzle #12	528095	- - -	0.06	0.85	0.68	0.011	-4	0	82
Inlet Nozzle #13	528097	- - -	0.05	0.86	0.75	0.009	5	0	> 82[(g)]
Inlet Nozzle #14	528207	- - -	0.06	0.82	0.70	0.013	-13	0	89
Outlet Nozzle # 15	536980	- - -	0.06	0.74	0.68	0.006	-22	0	> 88[(g)]
Outlet Nozzle # 16	526179	- - -	0.15	0.74	0.68	0.006	-31	0	> 81[(g)]
Outlet Nozzle # 17	526179	- - -	0.15	0.68	0.68	0.008	-4	0	92
Outlet Nozzle # 18	527963	- - -	0.11	0.76	0.78	0.012	-9.5	0	> 68[(g)]
Reactor Vessel Surveillance Materials									
Intermediate Shell Forging 05[(f)]	527536	- - -	- - -	- - -	- - -	- - -	- - -	- - -	- - -
Watts Bar Unit 1 Surveillance Weld[(f)]	895075	LW320 (P46)	0.03	0.75	- - -	- - -	- - -	- - -	- - -
Watts Bar Unit 2 Surveillance Weld	895075	LW320 (P46)	0.033	0.70	- - -	- - -	- - -	- - -	- - -

Table 3-1 Watts Bar Unit 1 Reactor Vessel Beltline, Extended Beltline, and Surveillance Material Properties and Chemistry(a)

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Catawba Unit 1 Surveillance Weld	895075	LW320 (P46)	0.05	0.73	- - -	- - -	- - -	- - -	- - -
McGuire Unit 2 Surveillance Weld	895075	LW320 (P46)	0.04	0.74	- - -	- - -	- - -	- - -	- - -

Notes:

(a) All chemistry values obtained from the WCAP-16761-NP [Ref. 18] and/or Watts Bar Unit 1 Certified Material Test Reports (CMTRs), unless otherwise noted. Although not required under current regulations, wt. % Mn and wt. % P values were provided for future use, if needed.

(b) The extended beltline and non-beltline forging $RT_{NDT(U)}$ and initial USE values were calculated as a part of this evaluation. The $RT_{NDT(U)}$ values are based on drop-weight data, tangentially oriented Charpy V-notch test data and NUREG-0800, BTP 5-3, Section B, Position 1.1(3)(a) and (b), unless otherwise noted. The initial USE values are the average of all impact energy values with ≥ 95% shear reduced to 65% of their original values to conservatively approximate transverse data per NUREG-0800, BTP 5-3, Section B, Position 1.2 [Ref. 12] methodology, unless otherwise noted.

(c) The initial RT_{NDT} was determined using NUREG-0800, BTP 5-3, Section B, Position 1.1(4) with the available measured impact energy data. Value is consistent with the initial RT_{NDT} value for weld Heat # 899680 as reported in WCAP-17455-NP [Ref. 25] for McGuire Unit 2, WCAP-17669-NP [Ref. 22] for Catawba Unit 1, and WCAP-18532-NP [Ref. 26] for Watts Bar Unit 2 [Ref. 2].

(d) The CMTRs for weld Heat # 899680 report only three impact energy values at a single test temperature (-12°C or 10.4°F) that did not reach greater than 55% shear. No other information is available for this weld heat. However, weld Heat # 895075 does have USE data and is a Rotterdam weld of the same flux type (Grau L.O., LW 320). Therefore, in absence of USE data for weld Heat # 899680, the weld Heat # 895075 test results from the first surveillance capsule, documented in WCAP-15046 [Ref. 23], is used to conservatively estimate the initial USE value for weld Heat # 899680. To ensure conservatism, the USE value from the first surveillance capsule is reduced by 25%.

(e) Initial USE value data is taken from UFSAR Table 5.2-11a.

(f) The Watts Bar Unit 1 surveillance forging material was made from reactor vessel Intermediate Shell Forging 05. Material properties for the surveillance forging material were taken to be identical to those of the vessel forging, since the surveillance material was cut from a prolongation of the actual vessel material. The Watts Bar Unit 1 surveillance weld material was made with the same weld heat, lot, and flux type as the Intermediate to Lower Shell Circumferential Weld (Heat # 895075 with Grau L.O. (LW320) flux, lot # P46) per WCAP-9298 [Ref. 24]. The chemistry values for the surveillance weld are the values in Table A-2 of WCAP-9298, Revision 3 [Ref. 24].

(g) Since Charpy data with ≥ 95% shear is not available for this material; this initial USE value is estimated based on the highest impact energy obtained with shear < 95%.

Table 3-2 Watts Bar Unit 2 Reactor Vessel Beltline, Extended Beltline, and Surveillance Material Properties and Chemistry[(a)]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
Reactor Vessel Beltline Materials									
Intermediate Shell Forging 05	527828	- - -	0.05	0.78	0.72	0.012	14	0	90
Lower Shell Forging 04	528658	- - -	0.05	0.81	0.72	0.006	5	0	105
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	LW320 (P46)	0.04[(b)]	0.73[(b)]	1.88[(b)]	0.013[(b)]	-50	0	127
Reactor Vessel Extended Beltline Materials									
Upper Shell Forging 06	411595	- - -	0.07	0.91	0.7	0.005	-14	0	94
Bottom Head Ring 03	5329	- - -	0.06	0.86	0.72	0.009	-40	0	105
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	LW320 (P23)	0.03	0.75	1.97	0.009	10[(d)]	0	101[(c)]
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	899680	LW320 (P23)	0.03	0.75	1.97	0.009	10[(d)]	0	101[(c)]
Reactor Vessel Non-Beltline Materials									
Inlet Nozzle #11	5328	- - -	0.05	0.83	0.75	0.008	-22	0	78
Inlet Nozzle #12	5330	- - -	0.06	0.85	0.77	0.011	-14	0	67
Inlet Nozzle #13	5331	- - -	0.06	0.82	0.75	0.010	-8	0	61
Inlet Nozzle #14	5335	- - -	0.04	0.79	0.77	0.009	-13	0	87
Outlet Nozzle # 15	5319	- - -	0.06	0.86	0.69	0.009	-22	0	90
Outlet Nozzle # 16	5324	- - -	0.06	0.84	0.71	0.011	-13	0	72
Outlet Nozzle # 17	5327	- - -	0.05	0.86	0.76	0.009	-39	0	84
Outlet Nozzle # 18	5334	- - -	0.04	0.80	0.77	0.009	-31	0	>84[(e)]
Reactor Vessel Surveillance Materials									
Intermediate Shell Forging 05	527828	- - -	- - -	- - -	- - -	- - -	- - -	- - -	- - -
Watts Bar Unit 2 Surveillance Weld	895075	LW320 (P46)	0.033	0.70	- - -	- - -	- - -	- - -	- - -
Catawba Unit 1 Surveillance Weld	895075	LW320 (P46)	0.05	0.73	- - -	- - -	- - -	- - -	- - -
Watts Bar Unit 1 Surveillance Weld	895075	LW320 (P46)	0.03	0.75	- - -	- - -	- - -	- - -	- - -

Table 3-2 Watts Bar Unit 2 Reactor Vessel Beltline, Extended Beltline, and Surveillance Material Properties and Chemistry[(a)]

Material Description	Heat Number	Flux Type (Lot)	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	$RT_{NDT(U)}$ (°F)	σ_I (°F)	Initial USE (ft-lb)
McGuire Unit 2 Surveillance Weld	895075	LW320 (P46)	0.04	0.74	- - -	- - -	- - -	- - -	- - -

Notes:

(a) Unless otherwise specified, all values are taken from WCAP-18191-NP [Ref. 2]. All values are based on information from the Watts Bar Unit 2 CMTRs and/or vessel fabrication records, unless noted otherwise. Although not required under current regulations, wt. % Mn and wt. % P values were provided for future use, if needed.

(b) The Watts Bar Unit 2 intermediate shell to lower shell circumferential weld (Heat # 895075) chemistry values are updated from the previous analysis of record (AOR), WCAP-18191-NP [Ref. 2], to be based on the average of all available data. This is also consistent with Heat # 895075 in Watts Bar Unit 1, Catawba Unit 1, and McGuire Unit 2.

(c) The CMTRs for weld Heat # 899680 report only three impact energy values at a single test temperature (-12°C or 10.4°F) that did not reach greater than 55% shear. No other information is available for this weld heat. However, weld Heat # 895075 does have USE data and is a Rotterdam weld of the same flux type (Grau L.O., LW 320). Therefore, in absence of USE data for weld Heat # 899680, the weld Heat # 895075 test results from the first surveillance capsule, documented in WCAP-18518 [Ref. 1], is used to conservatively estimate the initial USE value for weld Heat # 899680. To ensure conservatism, the USE value from the first surveillance capsule is conservatively reduced by 25%.

(d) The initial RT_{NDT} was determined using NUREG-0800, BTP 5-3, Section B, Position 1.1(4) with the available measured impact energy data. Value is consistent with the initial RT_{NDT} value for weld Heat # 899680 as reported in WCAP-17455-NP [Ref. 25] for McGuire Unit 2 and WCAP-17669-NP [Ref. 22] for Catawba Unit 1.

(e) Since Charpy data with ≥ 95% shear is not available for this material; this initial USE value is estimated based on the highest impact energy obtained with shear < 95%.

4 SURVEILLANCE DATA

Per RG 1.99, Revision 2 [Ref. 3], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, data from surveillance programs at other plants which include a Watts Bar Units 1 & 2 reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors. Data from a surveillance program at another plant is often called 'sister-plant' data.

The Watts Bar Units 1 & 2 surveillance capsules contain shell material from Intermediate Shell Forging 05 from their respective units and weld material from the intermediate shell to lower shell circumferential weld, Heat #895075, Flux Type Grau L.O. (LW320), Lot # P46 for both units. In addition, The Catawba Unit 1 and McGuire Unit 2 surveillance programs contain weld wire Heat # 895075, which was also used in the fabrication of the Watts Bar Units 1 & 2 intermediate to lower shell circumferential weld seam W05. Thus, the data from these surveillance programs are applicable to Watts Bar Units 1 & 2. Tables 4-1 and 4-2 summarize the surveillance data available from the Watts Bar Units 1 & 2 reactor vessel surveillance programs, as well as the Catawba Unit 1 and McGuire Unit 2 surveillance weld data.

Per RG 1.99, Revision 2, the use of surveillance data requires at least two credible data sets. Per Appendix B, the Watts Bar Unit 1 surveillance forging data for the intermediate shell forging 05 are deemed non-credible. Since Capsule U is the first capsule to be withdrawn from Watts Bar Unit 2, this criterion is not satisfied for the Unit 2 surveillance forging, as the surveillance data cannot be used to make embrittlement projections. Per Appendix B and Appendix C, the surveillance data for Heat # 895075, using all available data, was deemed credible for Watts Bar Units 1 & 2.

Table 4-1 Watts Bar Units 1 & 2 Intermediate Shell Forging Surveillance Capsule Data[(a)]

Unit	Material	Capsule	Capsule Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift Measured (°F)	Upper Shelf Energy Decrease Measured (%)
Unit 1	Intermediate Shell Forging 05 (Tangential)	U	0.46	98.3	19
		W	1.08	111.4	26
		X	1.75	94.7	20
		Z	2.40	144.5	23
	Intermediate Shell Forging 05 (Axial)	U	0.46	28.7	---
		W	1.08	79.0	3.2
		X	1.75	115.9	---
		Z	2.40	104.9	0
Unit 2	Intermediate Shell Forging 05 (Tangential)	U	0.614	26.7	26
	Intermediate Shell Forging 05 (Axial)	U	0.614	21.3	5

Notes:

(a) Information extracted from WCAP-16760-NP [Ref. 17] for Watts Bar Unit 1 and WCAP-18518-NP [Ref. 1] for Watts Bar Unit 2, unless otherwise specified.

(b) The fluence values are taken from Tables 2-2 and 2-20.

Table 4-2 Watts Bar Units 1 and 2, Catawba Unit 1, and McGuire Unit 2 Surveillance Capsule Data for Weld Heat # 895075[(a)]

Weld Metal Heat # 895075	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	Measured 30 ft-lb Transition Temperature Shift (°F)	Measured USE Decrease (%)	Cu Wt. %	Ni Wt. %	Position 1.1 CF Value (°F)[(c)]	Average Inlet Temperature During Period of Irradiation (°F)
Watts Bar 1	U	0.46	0	---	0.03	0.75	41	559[(d)]
	W	1.08	30.5	15[(b)]				
	X	1.75	25.8	---				
	Z	2.40	13.9	---				
Watts Bar 2	U	0.614	32.6	6	0.033	0.7	44.9	559[(d)]
Catawba 1	Z	0.286	1.91	---	0.05	0.73	68	562
	Y	1.29	17.79	---				
	V	2.27	26.50	---				
McGuire 2	V	0.302	38.51	---	0.04	0.74	54	557
	X	1.38	35.93	---				
	U	1.90	23.81	---				
	W	2.82	43.76	---				

Notes:

(a) All data except the Watts Bar Unit 1 and 2 fluences taken from WCAP-18518-NP [Ref. 1] unless otherwise noted. Fluence data is taken from Sections 2.2.1 and 2.2.2.

(b) Information taken from WCAP-16760-NP [Ref. 17]

(c) The Position 1.1 chemistry factor (CF) value for the surveillance welds were calculated using the Cu and Ni wt. % values and Table 1 of Regulatory Guide 1.99, Revision 2 [Ref. 3].

(d) Watts Bar Units 1 & 2 temperatures are determined by averaging (time-weighted) the inlet temperatures for all cycles prior to the capsule being removed.

5 CHEMISTRY FACTORS

The chemistry factors (CFs) were calculated using RG 1.99, Revision 2, Positions 1.1 and 2.1. Position 1.1 chemistry factors for each reactor vessel material are calculated using the best-estimate copper and nickel weight percent of the material and Tables 1 and 2 of RG 1.99, Revision 2. The best-estimate copper and nickel weight percent values for the Watts Bar Units 1 & 2 reactor vessel materials are provided in Tables 3-1 and 3-2. The Position 1.1 chemistry factors are summarized in Tables 5-3 and 5-4 for Watts Bar Units 1 & 2, respectively.

The Position 2.1 chemistry factors are calculated for the materials that have available surveillance data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, data from surveillance programs at other plants which include a Watts Bar Units 1 & 2 reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors. As discussed in Section 4, there is insufficient surveillance data to calculate a Position 2.1 CF for the surveillance forging for Watts Bar Unit 2; however, the Watts Bar Unit 2 Intermediate to Lower Shell Circumferential Weld W05 includes weld Heat # 895075, which does have surveillance data available from sister plants. The Watts Bar Units 1 & 2 sister-plant surveillance data is utilized in Tables 5-1 and 5-2 to calculate the Position 2.1 CF.

Adjustment of the measured ΔRT_{NDT} values are required per RG 1.99 [Ref. 3] due to chemistry differences between the surveillance welds and the Watts Bar Units 1 & 2 reactor vessel welds. The ratios between the RG 1.99, Position 1.1 chemistry factors of the surveillance welds and the vessel weld are shown below and are used as adjustment factors. Temperature adjustments are also considered in the Position 2.1 CF calculation in Table 5-1. The Position 2.1 CF is also included in Tables 5-3 and 5-4 for comparison with the Position 1.1 chemistry factor.

<u>Watts Bar Units 1 & 2 (Heat # 895075)</u>

Watts Bar Unit 1 data

$CF_{\text{Beltline Weld (Watts Bar Units 1 \& 2)}}$ = 54°F
$CF_{\text{Surv. Weld (Watts Bar Unit 1)}}$ = 41°F

Ratio = 54 ÷ 41 = 1.32
Applied to Watts Bar Unit 1 surveillance data for weld Heat # 895075

Watts Bar Unit 2 data

$CF_{\text{Beltline Weld (Watts Bar Units 1 \& 2)}}$ = 54°F
$CF_{\text{Surv. Weld (Watts Bar Unit 2)}}$ = 44.9°F

Ratio = 54 ÷ 44.9 = 1.20
Applied to Watts Bar Unit 2 surveillance data for weld Heat # 895075

Catawba Unit 1 data

$CF_{\text{Beltline Weld (Watts Bar Units 1 \& 2)}}$ = 54°F
$CF_{\text{Surv. Weld (Catawba Unit 1)}}$ = 68°F

Ratio = 54 ÷ 68 = 0.79
Applied to Catawba Unit 1 surveillance data for weld Heat # 895075

McGuire Unit 2 data

$CF_{\text{Beltline Weld (Watts Bar Units 1 \& 2)}}$ = 54°F

$CF_{\text{Surv. Weld (McGuire Unit 2)}}$ = 54°F

Ratio = 54 ÷ 54 = 1.00

Applied to McGuire Unit 2 surveillance data for weld Heat # 895075

Table 5-1 Calculation of Chemistry Factor for Weld Heat # 895075 Using All Available Surveillance Capsule Data[(a)]

Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Measured ΔRT_{NDT} (°F)	Adjusted ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Watts Bar Unit 1 Surveillance Weld	U	0.460	0.784	0	11.88	9.31	0.614
	W	1.08	1.022	30.5	52.14	53.26	1.044
	X	1.75	1.154	25.8	45.94	53.00	1.331
	Z	2.40	1.236	13.9	30.23	37.36	1.528
Watts Bar Unit 2 Surveillance Weld	U	0.614	0.863	32.6	49.92	43.10	0.745
Catawba Unit 1 Surveillance Weld	Z	0.286	0.658	1.91	10.99	7.23	0.433
	Y	1.29	1.071	17.79	23.53	25.20	1.147
	V	2.27	1.222	26.50	30.42	37.16	1.493
McGuire Unit 2 Surveillance Weld	V	0.302	0.672	38.51	45.51	30.58	0.452
	X	1.38	1.089	35.93	42.93	46.77	1.187
	U	1.90	1.176	23.81	30.81	36.22	1.382
	W	2.82	1.276	43.76	50.76	64.76	1.628
					SUM:	443.97	12.983
	$CF_{Heat\ \#\ 895075} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (443.97) \div (12.983) =$ **34.2°F**						

Notes:

(a) Unless otherwise noted, the data are taken from Tables 4-1 and 4-2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) The surveillance weld measured ΔRT_{NDT} values have been adjusted for chemistry and irradiation temperature as follows:

$$\text{Adjusted } \Delta RT_{NDT} = (\Delta RT_{NDT,\ Measured} + \text{temp. adjustment}) \times (CF_{vessel\ weld} \div CF_{surv.\ weld}).$$

The temperature adjustments are based on a time-weighted average temperature of the Watts Bar Units 1 & 2 reactor vessel, which is equal to 557°F over the life of the plant. In addition, the potential of a T_{avg} reduction of 7°F will be taken into consideration; thus, a value of 550°F will be used for the reactor vessel temperature for the adjustments. The Watts Bar Units 1 & 2 capsule irradiation temperatures are 559°F per Table 4-2, and the sister plant capsule irradiation temperatures are also provided in Table 4-2.

- For Watts Bar Unit 1 the CF ratio is 1.32 and temp. adjustment is 9°F (559°F - 550°F).
- For Watts Bar Unit 2, the CF ratio is 1.20 and temp. adjustment is 9°F (559°F - 550°F).
- For Catawba Unit 1, the CF ratio is 0.79 and temp. adjustment is 12°F (562°F - 550°F).
- For McGuire Unit 1 the CF ratio is 1.00 and temp. adjustment is 7°F (557°F - 550°F).

Table 5-2 Calculation of Watts Bar Unit 1 Chemistry Factor Value for Intermediate Shell Forging 05[(a)]

Material	Capsule	Capsule Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 05 (Tangential)	U	0.46	0.784	98.3	77.0	0.614
	W	1.08	1.022	111.4	113.8	1.044
	X	1.75	1.154	94.7	109.3	1.331
	Z	2.40	1.236	144.5	178.6	1.528
Intermediate Shell Forging 05 (Axial)	U	0.46	0.784	28.7	22.5	0.614
	W	1.08	1.022	79.0	80.7	1.044
	X	1.75	1.154	115.9	133.7	1.331
	Z	2.40	1.236	104.9	129.7	1.528
				SUM:	845.3	9.034
	$CF_{\text{Heat \# 895075}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (845.28) \div (9.034) =$ **93.6°F**					

Notes:

(a) Unless otherwise noted, the data are taken from Tables 4-1 and 4-2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) ΔRT_{NDT} values are measured 30 ft-lb shift values.

Table 5-3 Position 1.1 and 2.1 Chemistry Factors for Watts Bar Unit 1

Material	Chemistry Factor	
	Position 1.1[(a)] (°F)	Position 2.1 (°F)
Reactor Vessel Beltline Materials		
Intermediate Shell Forging 05	123.0	93.6[(b)]
Lower Shell Forging 04	51.0	- - -
Intermediate to Lower Shell Circumferential Weld	54.0	34.2[(c)]
Reactor Vessel Extended Beltline Materials		
Upper Shell Forging 06	86.0	---
Bottom Head Ring 03	37.0	---
Upper to Intermediate Shell Circumferential Weld	41.0	---
Lower Shell to Bottom Head Ring Weld	41.0	---
Reactor Vessel Surveillance Materials		
Surveillance Weld Material – Watts Bar Unit 1 (Heat # 895075)	41.0	---
Surveillance Weld Material – Watts Bar Unit 2 (Heat # 895075)	44.9	---
Surveillance Weld Material – Catawba Unit 1 (Heat # 895075)	68.0	---
Surveillance Weld Material – McGuire Unit 2 (Heat # 895075)	54.0	---

Notes:

(a) All values are based on Tables 1 and 2 of RG 1.99, Revision 2 (Position 1.1) using the Cu and Ni weight percent values given in Table 3-1.

(b) Value is from Table 5-2.

(c) Value is from Table 5-1.

Table 5-4 Position 1.1 and 2.1 Chemistry Factors for Watts Bar Unit 2

Material	Chemistry Factor	
	Position 1.1[(a)] (°F)	Position 2.1 (°F)
Reactor Vessel Beltline Materials		
Intermediate Shell Forging 05	31.0	---
Lower Shell Forging 04	31.0	---
Intermediate to Lower Shell Circumferential Weld	54.0	34.2[(b)]
Reactor Vessel Extended Beltline Materials		
Upper Shell Forging 06	44.0	---
Bottom Head Ring 03	37.0	---
Upper to Intermediate Shell Circumferential Weld	41.0	---
Lower Shell to Bottom Head Ring Weld	41.0	---
Reactor Vessel Surveillance Materials		
Surveillance Weld Material – Watts Bar Unit 2 (Heat # 895075)	44.9	---
Surveillance Weld Material – Catawba Unit 1 (Heat # 895075)	68.0	---
Surveillance Weld Material – Watts Bar Unit 1 (Heat # 895075)	41.0	---
Surveillance Weld Material – McGuire Unit 2 (Heat # 895075)	54.0	---

Notes:

(a) All values are based on Tables 1 and 2 of RG 1.99, Revision 2 (Position 1.1) using the Cu and Ni weight percent values given in Table 3-2.

(b) Value is from Table 5-1.

6 PRESSURIZED THERMAL SHOCK EVALUATION

6.1 RT_{PTS} CALCULATIONS

Pressurized thermal shock (PTS) may occur during a severe system transient such as a loss-of-coolant accident (LOCA) or steam line break. Such transients may challenge the integrity of the reactor pressure vessel (RPV) under the following conditions: severe overcooling of the inside surface of the vessel wall followed by high re-pressurization, significant degradation of vessel material toughness caused by radiation embrittlement, and the presence of a critical-size defect anywhere within the vessel wall.

In 1985, the U.S. NRC issued a formal ruling on PTS (10 CFR 50.61 [Ref. 4]) that established screening criteria on pressurized water reactor (PWR) vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of license, termed RT_{PTS}. RT_{PTS} screening values were set by the U.S. NRC for beltline axial welds, forgings or plates, and for beltline circumferential weld seams for plant operation to the end-of-plant license. All domestic PWR vessels have been required to evaluate vessel embrittlement in accordance with the criteria through the end of license. The U.S. NRC revised 10 CFR 50.61 in 1991 and 1995 to change the procedure for calculating radiation embrittlement. These revisions make the procedure for calculating the reference temperature for pressurized thermal shock (RT_{PTS}) values consistent with the methods given in RG 1.99, Revision 2 [Ref. 3].

These accepted methods were used with the clad/base metal interface fluence values of Section 2 to calculate the following RT_{PTS} values for the Watts Bar Units 1 & 2 RPV materials at 32 EFPY (EOL) and 48 EFPY (EOLE). Note that the Watts Bar Unit 2 RT_{PTS} values were only calculated up to EOL. The EOL and EOLE RT_{PTS} calculations are summarized in Tables 6-1 through 6-3.

PTS Conclusion

The Watts Bar Unit 1 limiting RT_{PTS} values for base metal or longitudinal weld materials are 222.2°F (32 EFPY) and 235.9°F (48 EFPY), which correspond to the Intermediate Shell Forging 05 (using Position 1.1). The limiting RT_{PTS} values for circumferentially oriented welds are 50.4°F (32 EFPY) and 57.2°F (48 EFPY), which correspond to the Lower Shell to Bottom Head Ring Circumferential Weld Seam W04 (using Position 1.1). The Watts Bar Unit 2 limiting RT_{PTS} value for base metal or longitudinal weld materials is 84.3°F (32 EFPY), which corresponds to the Intermediate Shell Forging 05 (using Position 1.1). The limiting RT_{PTS} value for circumferentially oriented welds is 62.5°F (32 EFPY), which corresponds to the Lower Shell to Bottom Head Ring Circumferential Weld Seam W04 (using Position 1.1).

All of the beltline and extended beltline reactor vessel materials for Watts Bar Units 1 & 2 are projected to remain below the RT_{PTS} screening criteria values of 270°F for plates, forgings, and longitudinal welds, and 300°F for circumferentially oriented welds (per 10 CFR 50.61) at 32 and 48 EFPY. Note that the Watts Bar Unit 1 RT_{PTS} values were only calculated up to EOL (32 EFPY).

Table 6-1 RT_{PTS} Calculations for Watts Bar Unit 1 Reactor Vessel Materials at EOL (32 EFPY)

Material	CF[a]	Surface Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surf. FF[c]	$RT_{NDT(U)}$[d] (°F)	Predicted ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[e] (°F)	M (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials									
Intermediate Shell Forging 05	123.0	1.71	1.148	47	141.2	0.0	17.0	34.0	222.2
Using non-credible surveillance data	93.6	1.71	1.148	47	107.4	0.0	17.0	34.0	188.4
Lower Shell Forging 04	51.0	1.75	1.154	5	58.8	0.0	17.0	34.0	97.8
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	54.0	1.68	1.143	-43	61.7	0.0	28.0	56.0	74.7
Using credible surveillance data[f]	34.2	1.68	1.143	-43	39.1	0.0	14.0	28.0	24.1
Reactor Vessel Extended Beltline Materials									
Upper Shell Forging 06	86.0	0.0527	0.301	-22	25.9	0.0	12.9	25.9	29.8
Bottom Head Ring 03	37.0	0.119	0.453	-40	16.7	0.0	8.4	16.7	-6.5
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	41.0	0.0625	0.330	10	13.5	0.0	6.8	13.5	37.0
Lower Shell to Bottom Head Ring Weld Seam W04 (Heat # 899680)	41.0	0.143	0.492	10	20.2	0.0	10.1	20.2	50.4

Notes:

(a) Chemistry factors are taken from Table 5-3.

(b) Fluence values taken from Tables 2-7 and 2-11.

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(d) $RTNDT_{(U)}$ values taken from Table 3-1.

(e) Per 10 CFR 50.61, the base metal σ_Δ = 17°F when surveillance data are non-credible or not used to determine the CF, and the base metal σ_Δ = 8.5°F when credible surveillance data are used. Also, per 10 CFR 50.61, the weld metal σ_Δ = 28°F when surveillance data are non-credible or not used to determine the CF, and the weld metal σ_Δ = 14°F when credible surveillance data are used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(f) The credibility evaluation for the surveillance weld data in Appendix B determined that the surveillance weld data for Heat # 895075 are deemed credible and the data for the Intermediate Shell Forging are deemed non-credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 6-2 RT_{PTS} Calculations for Watts Bar Unit 1 Reactor Vessel Materials at EOLE (48 EFPY)

Material	CF[(a)]	Surface Fluence (x 10^{19} n/cm², E > 1.0 MeV)	Surf. FF[(c)]	$RT_{NDT(U)}$ [(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(e)] (°F)	M (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials									
Intermediate Shell Forging 05	123.0	2.64	1.260	47	154.9	0.0	17.0	34.0	235.9
Using non-credible surveillance data	93.6	2.64	1.260	47	117.9	0.0	17.0	34.0	198.9
Lower Shell Forging 04	51.0	2.73	1.268	5	64.7	0.0	17.0	34.0	103.7
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	54.0	2.60	1.256	-43	67.8	0.0	28.0	56.0	80.8
Using credible surveillance data[(f)]	34.2	2.60	1.256	-43	43.0	0.0	14.0	28.0	28.0
Reactor Vessel Extended Beltline Materials									
Upper Shell Forging 06	86.0	0.0788	0.371	-22	31.9	0.0	15.9	31.9	41.8
Bottom Head Ring 03	37.0	0.170	0.531	-40	19.7	0.0	9.8	19.7	-0.7
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	41.0	0.0937	0.404	10	16.6	0.0	8.3	16.6	43.1
Lower Shell to Bottom Head Ring Weld Seam W04 (Heat # 899680)	41.0	0.205	0.575	10	23.6	0.0	11.8	23.6	57.2

Notes:

(a) Chemistry factors are taken from Table 5-3.

(b) Fluence values taken from Tables 2-7 and 2-11.

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(d) $RTNDT_{(U)}$ values taken from Table 3-1.

(e) Per 10 CFR 50.61, the base metal σ_Δ = 17°F when surveillance data are non-credible or not used to determine the CF, and the base metal σ_Δ = 8.5°F when credible surveillance data are used. Also, per 10 CFR 50.61, the weld metal σ_Δ = 28°F when surveillance data are non-credible or not used to determine the CF, and the weld metal σ_Δ = 14°F when credible surveillance data are used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(f) The credibility evaluation for the surveillance weld data in Appendix B determined that the surveillance weld data for Heat # 895075 are deemed credible and the data for the Intermediate Shell Forging are deemed non-credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 6-3 RT_{PTS} Calculations for Watts Bar Unit 2 Reactor Vessel Materials at EOL (32 EFPY)

Material	CF[(a)]	Surface Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Surf. FF[(c)]	$RT_{NDT(U)}$[(d)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_U (°F)	σ_Δ[(e)] (°F)	M (°F)	RT_{PTS} (°F)
Reactor Vessel Beltline Materials									
Intermediate Shell Forging 05	31.0	1.87	1.171	14	36.3	0.0	17.0	34.0	84.3
Lower Shell Forging 04	31.0	1.93	1.180	5	36.6	0.0	17.0	34.0	75.6
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	54.0	1.83	1.166	-50	62.9	0.0	28.0	56.0	68.9
Using credible surveillance data[(f)]	34.2	1.83	1.166	-50	33.9	0.0	14.0	28.0	17.9
Reactor Vessel Extended Beltline Materials									
Upper Shell Forging 06	44.0	0.0538	0.304	-14	13.4	0.0	6.7	13.4	12.8
Bottom Head Ring 03	37.0	0.241	0.615	-40	22.7	0.0	11.4	22.7	5.5
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	41.0	0.0641	0.334	10	13.7	0.0	6.8	13.7	37.4
Lower Shell to Bottom Head Ring Weld Seam W04 (Heat # 899680)	41.0	0.267	0.641	10	26.3	0.0	13.1	26.3	62.5

Notes:

(a) Chemistry factors are taken from Table 5-4.

(b) Fluence values taken from Tables 2-25 and 2-29.

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(d) $RT_{NDT(U)}$ values taken from Table 3-2.

(e) Per 10 CFR 50.61, the base metal σ_Δ = 17°F when surveillance data are non-credible or not used to determine the CF, and the base metal σ_Δ = 8.5°F when credible surveillance data are used. Also, per 10 CFR 50.61, the weld metal σ_Δ = 28°F when surveillance data are non-credible or not used to determine the CF, and the weld metal σ_Δ = 14°F when credible surveillance data are used. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(f) The credibility evaluation in Appendix C for the surveillance weld data determined that the surveillance weld data for Heat # 895075 are deemed credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF.

6.2 EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

The emergency response guideline (ERG) limits, HF04BG [Ref. 14], were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}, which is equivalent to the RT_{PTS} values calculated in Section 6.1. These generic categories were conservatively generated for the Westinghouse Owners Group (WOG) to be applicable to all Westinghouse plants.

The highest value of RT_{NDT} for which the generic category ERG limits were developed is 250°F for a longitudinal flaw and 300°F for a circumferential flaw. Therefore, if the limiting vessel material has an RT_{NDT} that exceeds 250°F for a longitudinal flaw or 300°F for a circumferential flaw, plant-specific ERG P-T limits must be developed.

The ERG category is determined by the magnitude of the limiting RT_{NDT} value, i.e., RT_{PTS}. The material with the highest RT_{NDT} defines the limiting material. The material with the highest RT_{NDT} for Watts Bar Units 1 & 2 is the Intermediate Shell Forging 05 material for both units. Tables 6-4 and 6-5 identify ERG category limits and the limiting material RT_{NDT} value at 32 and 48 EFPY for Watts Bar Units 1 & 2, respectively. Note that calculations were only performed up to EOL (32 EFPY) for Watts Bar Unit 2.

Table 6-4 Evaluation of Watts Bar Unit 1 ERG Limit Category

ERG Pressure-Temperature Limits	
Applicable RT_{NDT} Value[(a)]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category IIIb
Limiting RT_{NDT} Value[(b)]	
EFPY and Limiting Reactor Vessel Material	**RT_{NDT} Value @ EOL**
32 (Intermediate Shell Forging 05)	222.2
48 (Intermediate Shell Forging 05)	235.9

Notes:

(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.

(b) Values taken from Tables 6-1 and 6-2.

Table 6-5 Evaluation of Watts Bar Unit 2 ERG Limit Category

ERG Pressure-Temperature Limits	
Applicable RT_{NDT} Value[a]	**ERG P-T Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category IIIb
Limiting RT_{NDT} Value[b]	
EFPY and Limiting Reactor Vessel Material	**RT_{NDT} Value @ EOL**
32 (Intermediate Shell Forging 05)	84.3

Notes:

(a) Longitudinally oriented flaws are applicable only up to 250°F; circumferentially oriented flaws are applicable up to 300°F.

(b) Value taken from Table 6-3.

Per the ERG limit guidance document [Ref. 14], some vessels do not change categories for operation through the end of license. However, when a vessel does change ERG categories between the beginning and end of operation, a plant-specific assessment must be performed to determine at what operating time the category changes. Thus, the ERG classification need not be changed until the operating cycle during which the maximum vessel value of actual or estimated real-time RT_{NDT} exceeds the limit on its current ERG category.

Conclusion of ERG P-T Limit Categorization

Per Table 6-4, the limiting Watts Bar Unit 1 limiting material (Intermediate Shell Forging 05) has an RT_{NDT} between 200°F and 250°F through 32 and 48 EFPY. Therefore, as of 17 EFPY, Watts Bar Unit 1 is in ERG Category II and will remain in Category II through EOL and EOLE.

Per Table 6-5, the Watts Bar Unit 2 limiting material (Intermediate Shell Forging 05) has an RT_{NDT} of less than 200°F through 32 EFPY. Therefore, Watts Bar Unit 2 is in ERG Category I through EOL.

7 UPPER-SHELF ENERGY EVALUATION

Charpy USE is associated with the determination of acceptable RPV toughness during the licensed operating period when the vessel is exposed to additional irradiation. The requirements on USE are included in 10 CFR 50, Appendix G [Ref. 5]. 10 CFR 50, Appendix G requires utilities to submit an analysis at least three years prior to the time that the USE of any RPV material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

There are two methods that can be used to predict the decrease in USE with irradiation, depending on the availability of credible surveillance capsule data as defined in RG 1.99, Revision 2 [Ref. 3]. For vessel beltline materials that are not in the surveillance program or have non-credible data, the Charpy USE (Position 1.2) is assumed to decrease as a function of fluence and copper content, as indicated in Figure 2 of RG 1.99, Revision 2.

When two or more credible surveillance sets become available from the reactor, they may be used to determine the Charpy USE of the surveillance material. The surveillance data are then used in conjunction with the RG 1.99 to predict the change in USE (Position 2.2) of the RPV material due to irradiation. Since Capsule U is the first capsule withdrawn from Watts Bar Unit 2, this method is not applicable to Unit 2.

The 32 EFPY (EOL) and 48 EFPY (EOLE) Position 1.2 USE values of the vessel materials can be predicted using the corresponding 1/4T fluence projection, the copper content of the materials, and Figure 2 in RG 1.99, Revision 2. The projected USE values were calculated to determine if the Watts Bar Units 1 & 2 beltline and extended beltline materials remain above the 50 ft-lb criterion at EOL and EOLE. Note that the Watts Bar Unit 2 values were only calculated up to EOL (32 EFPY). These calculations are summarized in Tables 7-1 through 7-3.

USE Conclusion

As presented in Tables 7-1 and 7-2, the Watts Bar Unit 1 beltline and extended beltline materials, except for the intermediate shell forging 05, are expected to have an upper shelf energy (USE) greater than 50 ft-lb (per 10 CFR 50, Appendix G [Ref. 5]) at 32 and 48 EFPY.

As previously discussed in WCAP-16760-NP [Ref. 17], Westinghouse completed a generic evaluation in WCAP-13587 [Ref. 21] to demonstrate margin to safety, relative to USE. As identified in WCAP-13587, Table 3-5, the minimum acceptable USE for a 4-loop plant is 43 ft-lb. As presented in Tables 7-1 and 7-2, the EOL (32 EFPY) and EOLE (48 EFPY) USE values for the Watts Bar Unit 1 intermediate shell forging 05 are greater than 43 ft-lb. Thus, the bounding evaluation in WCAP-13587 shows that the Watts Bar Unit 1 intermediate shell forging 05 will maintain an equivalent margin (through EOLE), with respect to USE per the requirements of 10 CFR Part 50, Appendix G.

As presented in Table 7-3, all Watts Bar Unit 2 beltline and extended beltline materials are projected to remain above the USE screening criterion of 50 ft-lb (per 10 CFR 50, Appendix G [Ref. 5]) at 32 EFPY.

Table 7-1 Watts Bar Unit 1 Predicted USE Values at 32 EFPY

Material	Weight % Cu[(a)]	1/4T EOL Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	Projected EOL USE (ft-lb)
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	0.16	1.03	62	26	46
With surveillance data				26[(d)]	46
Lower Shell Forging 04	0.08	1.05	111	19	90
Intermediate to Lower Shell Circumferential Weld Seam W05	0.04	1.01	134	19	109
With surveillance data				15[(d)]	114
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.12	0.0317	99	10	89
Bottom Head Ring 03	0.06	0.0716	105	11	94
Upper to Intermediate Shell Circumferential Weld Seam W06	0.03	0.0376	98	9	89
Lower Shell to Bottom Head Ring Weld Seam W04	0.03	0.0861	98	11	87

Notes:

(a) Copper weight percent values and unirradiated USE values were taken from Table 3-1. For the predicted USE decrease determinations, the base metal and weld Cu weight percentages were conservatively rounded up to the nearest line in RG 1.99, Revision 2, Figure 2.

(b) Values taken from Table 8-3 of this report.

(c) Unless otherwise specified, values were calculated using Figure 2 from RG 1.99, Revision 2, Position 1.2.

(d) Calculated using Figure 2 from RG 1.99, Revision 2, Position 2.2. These results should be used in preference to Position 1.2.

Table 7-2 Watts Bar Unit 1 Predicted USE Values at 48 EFPY

Material	Weight % Cu(a)	1/4T EOL Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE(a) (ft-lb)	Projected USE Decrease(c) (%)	Projected EOL USE (ft-lb)
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	0.16	1.59	62	28	45
With surveillance data				28(d)	45
Lower Shell Forging 04	0.08	1.64	111	21	88
Intermediate to Lower Shell Circumferential Weld Seam W05	0.04	1.56	134	21	106
With surveillance data				17(d)	111
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.12	0.0474	99	11	88
Bottom Head Ring 03	0.06	0.102	105	11	93
Upper to Intermediate Shell Circumferential Weld Seam W06	0.03	0.0564	98	10	88
Lower Shell to Bottom Head Ring Weld Seam W04	0.03	0.123	98	12	86

Notes:

(a) Copper weight percent values and unirradiated USE values were taken from Table 3-1. For the predicted USE decrease determinations, the base metal and weld Cu weight percentages were conservatively rounded up to the nearest line in RG 1.99, Revision 2, Figure 2.

(b) Values taken from Table 8-4 of this report.

(c) Unless otherwise specified, values were calculated using Figure 2 from RG 1.99, Revision 2, Position 1.2.

(d) Calculated using Figure 2 from RG 1.99, Revision 2, Position 2.2. These results should be used in preference to Position 1.2.

Table 7-3 Watts Bar Unit 2 Predicted USE Values at 32 EFPY

Material	Weight % Cu[(a)]	1/4T EOL Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	Projected EOL USE (ft-lb)
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	0.05	1.13	90	20	72
Lower Shell Forging 04	0.05	1.16	105	20	84
Intermediate to Lower Shell Circumferential Weld Seam W05	0.04	1.10	127	20	102
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.07	0.0324	94	9	86
Bottom Head Ring 03	0.06	0.145	105	12	92
Upper to Intermediate Shell Circumferential Weld Seam W06	0.03	0.0386	101	9	92
Lower Shell to Bottom Head Ring Weld Seam W04	0.03	0.161	101	13	88

Notes:

(a) Copper weight percent values and unirradiated USE values were taken from Table 3-2. For the predicted USE decrease determinations, the base metal and weld Cu weight percentages were conservatively rounded up to the nearest line in RG 1.99, Revision 2, Figure 2.

(b) Values taken from Table 8-5 of this report.

(c) Calculated using Figure 2 from RG 1.99, Revision 2, Position 1.2.

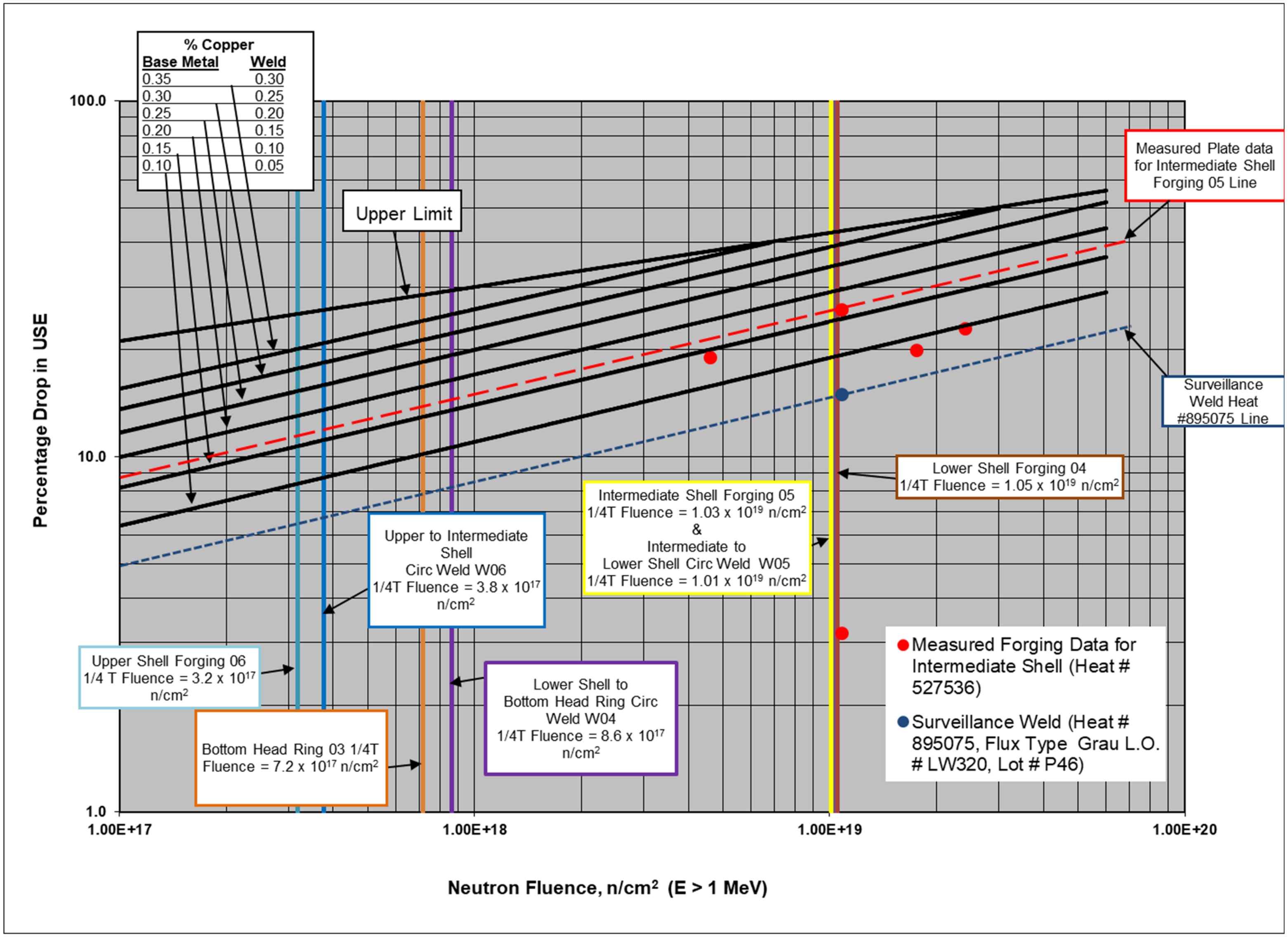

Figure 7-1
Watts Bar Unit 1 Regulatory Guide 1.99 Revision 2
Predicted Decrease in USE at 32 EFPY as a Function of Copper and Fluence

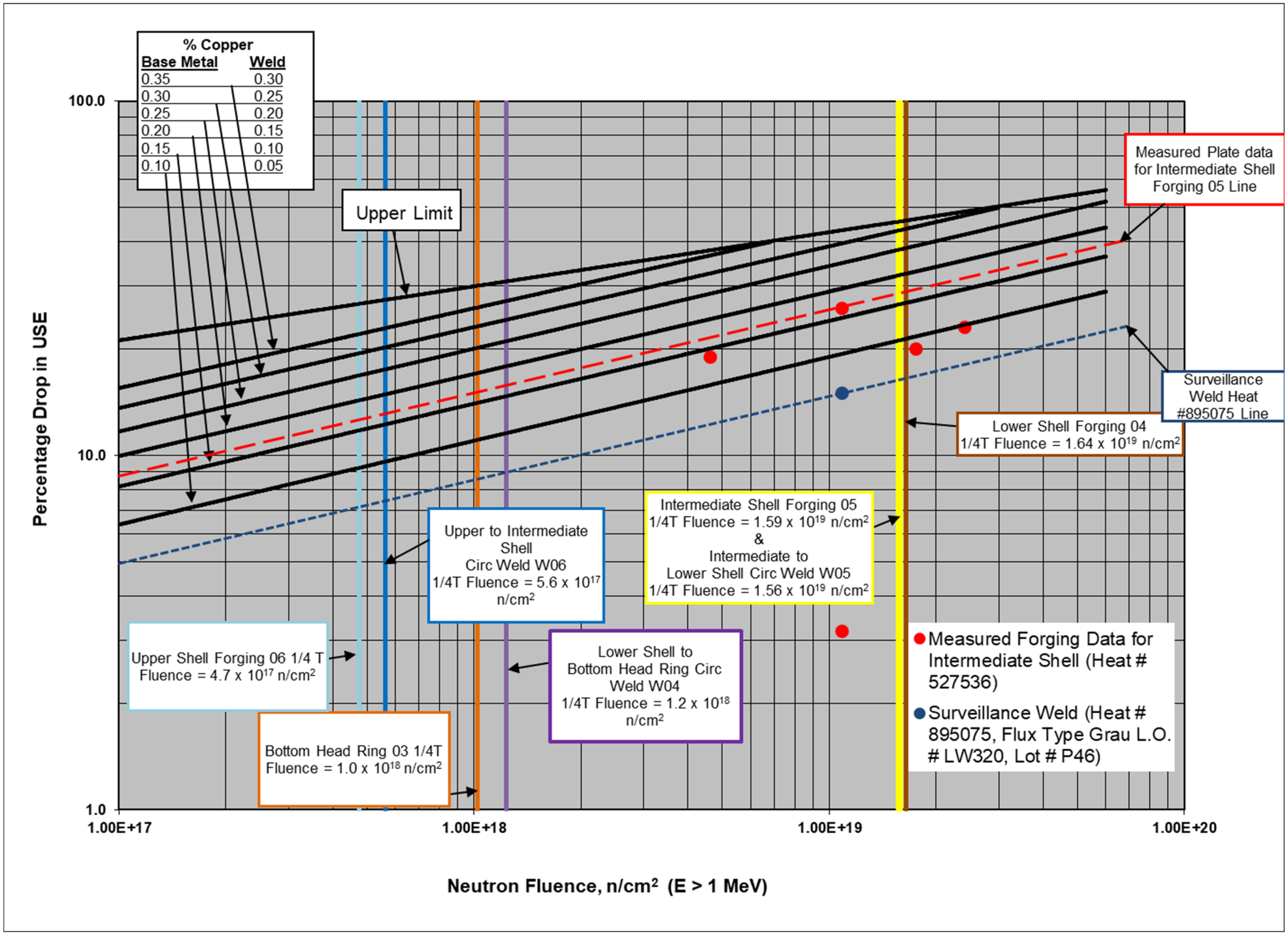

Figure 7-2
Watts Bar Unit 1 Regulatory Guide 1.99 Revision 2
Predicted Decrease in USE at 48 EFPY as a Function of Copper and Fluence

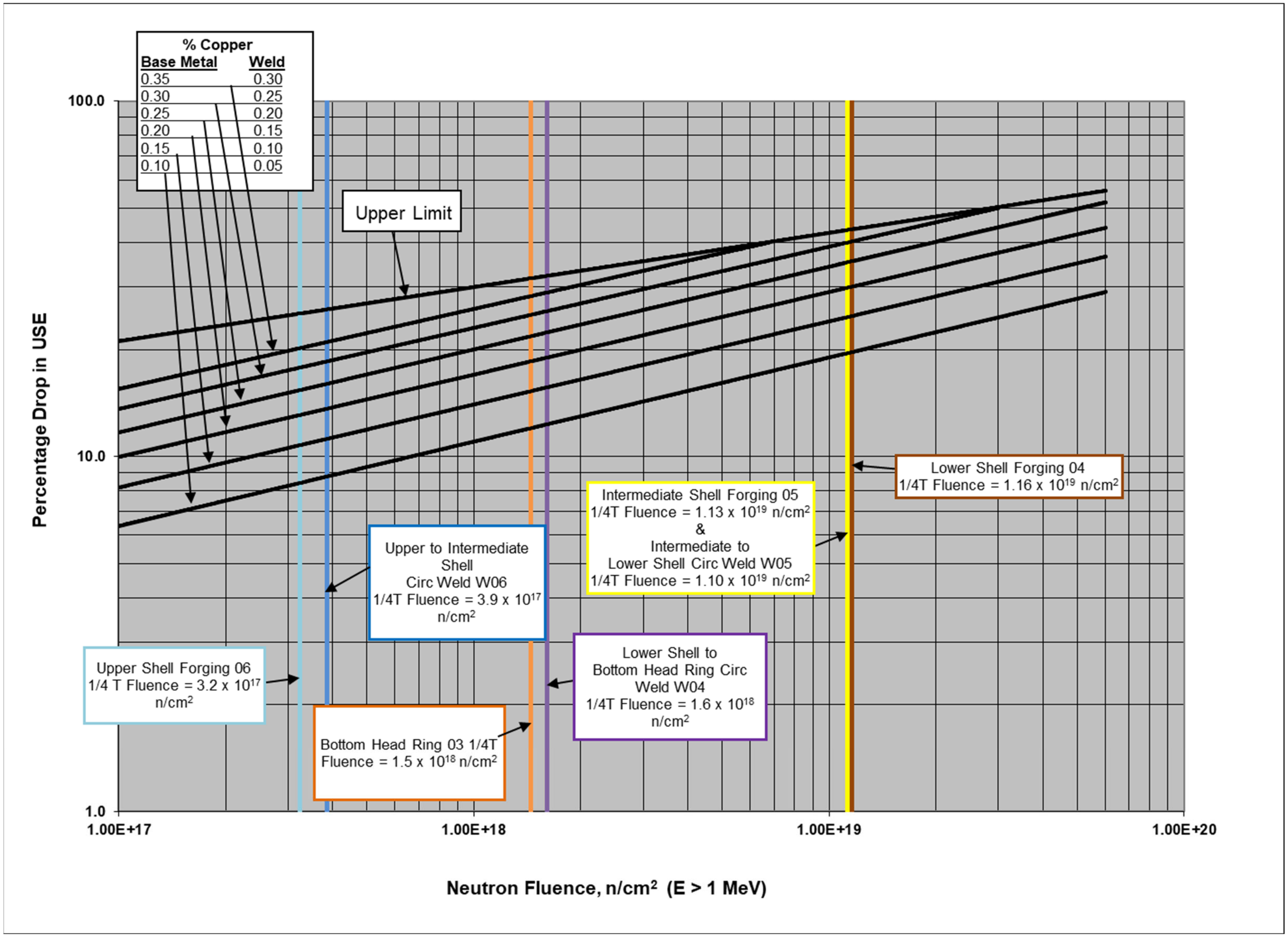

Figure 7-3
Watts Bar Unit 2 Regulatory Guide 1.99 Revision 2
Predicted Decrease in USE at 32 EFPY as a Function of Copper and Fluence

8 APPLICABILITY DETERMINATION OF PRESSURE-TEMPERATURE LIMIT CURVES

Heatup and cooldown limit curves, also known as pressure-temperature (P-T) limit curves, are calculated in accordance with the requirements of 10 CFR Part 50, Appendix G [Ref. 5], as augmented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code [Ref. 6], using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the RPV. The most limiting RT_{NDT} of the material in the RPV is determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}) per RG 1.99 [Ref. 3].

P-T limit curves for normal heatup and cooldown of the primary reactor coolant system for Watts Bar Unit 1 were developed in WCAP-16761-NP [Ref. 18] for 32 and 48 EFPY. These heatup and cooldown curves were generated using the methodology documented in WCAP-14040, Revision 1. (Note, these curves were determined to meet the requirements of WCAP-14040 Revision 4). The P-T limit curves for Watts Bar Unit 2 were developed in WCAP-18191-NP [Ref. 2] for 32 EFPY. These heatup and cooldown curves were generated using the methodology documented in WCAP-14040, Revision 4. As part of the review for the increase in TPBARs, the fluence projections were updated and are presented in Sections 2.2.1 and 2.2.2 for Watts Bar Units 1 & 2, respectively. The RG 1.99 [Ref. 3] methodology used to develop the limiting ART values in WCAP-16761-NP is consistent with the methodology used in this analysis.

As a result of updated fluence data, applicability checks of the current Watts Bar Units 1 & 2 P-T limit curves are appropriate. To confirm the Watts Bar Unit 1 32 and 48 EFPY P-T limits curves and the Watts Bar Unit 2 32 EFPY P-T limit curves, the updated reactor vessel material ART values from both the beltline and extended beltline must be shown to be less-than or equal-to the limiting beltline material ART values used in development of the current P-T limit curves.

The RG 1.99 methodology was used along with the fluence values documented in Section 2 to calculate ART values. Tables 8-3 through 8-5 provide the surface, 1/4T, and 3/4T fluence and fluence factor (FF) values for Watts Bar Units 1 & 2 which are needed to calculate ART values.

The ART calculations are summarized in Tables 8-6 through 8-11. For Watts Bar Unit 1 the limiting material is the Intermediate Shell Forging 05 with ART values of 205.0°F (32 EFPY) and 219.7°F (48 EFPY) at the 1/4T location and 170.4°F (32 EFPY) and 185.0 (48 EFPY) at the 3/4T location. All values were calculated using RG 1.99, Revision 2, Position 1.1.

For Watts Bar Unit 2 the limiting material is the Intermediate Shell Forging 05 with ART values of 78.0°F at the 1/4T location and 60.6°F at the 3/4T location. Both values were calculated using RG 1.99, Revision 2, Position 1.1.

Existing P-T Limit Curves Applicability Conclusions

Comparison of the limiting ART values to those used in calculation of the 32 and 48 EFPY P-T limit curves for Watts Bar Unit 1 and the 32 EFPY P-T limit curves for Watts Bar Unit 2 are contained in Tables 8-1 and 8-2. The revised limiting ART values in all cases are less than the limiting ART values used to develop the current P-T limit curves. Since the ART values have decreased in the updated analyses, there is no impact on the applicability of the 32 and 48 EFPY P-T limit curves for Watts Bar Unit 1 or the 32 EFPY P-T limit curves for Watts Bar Unit 2.

Although the reactor vessel nozzles are not a part of the extended beltline, NRC RIS 2014-11 requires that the nozzle materials be evaluated for their potential effect on P-T limit curves due to the higher stresses in the nozzle corner region. These higher stresses can potentially result in more restrictive P-T limits, even if the RT_{NDT} for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries. The concerns of RIS 2014-11 have been addressed generically for the U.S. PWR operating fleet in PWROG-15109-NP-A [Ref. 15]. The results of PWROG-15109-NP-A demonstrate that P-T limit curves developed with current NRC-approved methods (e.g., WCAP-14040-A [Ref. 16]) bound the generic nozzle P-T limit curves. This document has been approved by the NRC as an acceptable means to address the concerns of RIS 2014-11. The results and conclusions of PWROG-15109-NP-A are applicable as long as the plant-specific Watts Bar Units 1 & 2 fluence of the nozzle corners remain less than the screening criterion of 4.28 x 10^{17} n/cm^2, as described in PWROG-15109-NP-A. Section 2 of this report demonstrates the Watts Bar Units 1 & 2 adherence to this screening criterion at 32 and 48 EFPY, thus PWROG-15109-NP-A is applicable.

Even though no additional work is required to address the effects of nozzle P-T curves, it is worth mentioning that Watts Bar Unit 2 previously had a plant specific analysis performed in WCAP-18191-NP [Ref. 2] which demonstrate the beltline P-T limit curves remain bounding. Since the nozzles remain below the RIS 2014-11 [Ref. 7] threshold of 1 x 10^{17} n/cm^2, the effects of embrittlement do not need to be considered; thus, the nozzle P-T curves evaluated in WCAP-18191-NP remain applicable.

Table 8-1 Summary of the Limiting ART Values for Watts Bar Unit 1

	1/4T Limiting ART (°F)		3/4T Limiting ART (°F)	
	P-T Limit Curves AOR (WCAP-16761-NP)	Maximum ART Value from Tables 8-6 and 8-8	P-T Limit Curves AOR (WCAP-16761-NP)	Maximum ART Value from Tables 8-7 and 8-9
32 EFPY	205.74	205.0	171.15	170.4
48 EFPY	220.01	219.7	185.26	185.0
Limiting Material	Intermediate Shell Forging 05			

Table 8-2 Summary of the Limiting ART Values for Watts Bar Unit 2

	1/4T Limiting ART (°F)		3/4T Limiting ART (°F)	
	P-T Limit Curves AOR (WCAP-18191-NP)	Maximum ART Value from Table 8-10	P-T Limit Curves AOR (WCAP-18191-NP)	Maximum ART Value from Table 8-11
32 EFPY	88.0	78.0	71.0	60.6
Limiting Material	Intermediate Shell Forging 05			

Table 8-3 Watts Bar Unit 1 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 32 EFPY

Material	Surface Fluence[a] (x 10^{19} n/cm^2, E> 1.0 MeV)	1/4T Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[c]	3/4T Fluence[b] (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF[c]
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	1.71	1.03	1.008	0.373	0.727
Lower Shell Forging 04	1.75	1.05	1.014	0.381	0.733
Intermediate to Lower Shell Circumferential Weld Seam W05	1.68	1.01	1.003	0.366	0.722
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.0527	0.0317	0.227	0.0115	0.120
Bottom Head Ring 03	0.119	0.0716	0.353	0.0259	0.201
Upper to Intermediate Shell Circumferential Weld Seam W06	0.0625	0.0376	0.250	0.0136	0.135
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	0.143	0.0861	0.387	0.0312	0.225

Notes:

(a) The surface fluence values for the reactor vessel materials were taken from Table 2-7 and 2-11.

(b) 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (8.465 inches) and equation $f = f_{surf} * e^{-0.24\,(x)}$ from RG 1.99, Revision 2, where x = the depth into the vessel wall (inches).

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

Table 8-4 Watts Bar Unit 1 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 48 EFPY

Material	Surface Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(c)]	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF[(c)]
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	2.64	1.59	1.128	0.575	0.845
Lower Shell Forging 04	2.73	1.64	1.137	0.595	0.855
Intermediate to Lower Shell Circumferential Weld Seam W05	2.60	1.57	1.124	0.567	0.841
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.0788	0.0474	0.284	0.0172	0.156
Bottom Head Ring 03	0.170	0.102	0.421	0.0370	0.248
Upper to Intermediate Shell Circumferential Weld Seam W06	0.0937	0.0564	0.312	0.0204	0.174
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	0.205	0.123	0.460	0.0447	0.275

Notes:

(a) The surface fluence values for the reactor vessel materials were taken from Table 2-7 and 2-11.

(b) 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (8.465 inches) and equation $f = f_{surf} * e^{-0.24\,(x)}$ from RG 1.99, Revision 2, where x = the depth into the vessel wall (inches).

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

Table 8-5 Watts Bar Unit 2 Fluence and Fluence Factor Values for the Surface, 1/4T, and 3/4T Locations at 32 EFPY

Material	Surface Fluence[(a)] (x 10^{19} n/cm^2, E> 1.0 MeV)	1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[(c)]	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E> 1.0 MeV)	3/4T FF[(c)]
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	1.87	1.13	1.033	0.407	0.751
Lower Shell Forging 04	1.93	1.16	1.042	0.421	0.759
Intermediate to Lower Shell Circumferential Weld Seam W05	1.83	1.10	1.027	0.399	0.745
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.0538	0.0324	0.230	0.0117	0.122
Bottom Head Ring 03	0.241	0.145	0.495	0.0525	0.301
Upper to Intermediate Shell Circumferential Weld Seam W06	0.0641	0.0386	0.254	0.0140	0.137
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	0.267	0.161	0.518	0.0582	0.317

Notes:

(a) The surface fluence values for the reactor vessel materials were taken from Table 2-25 and 2-29.

(b) 1/4T and 3/4T fluence values were calculated from the surface fluence, the reactor vessel beltline thickness (8.465 inches) and equation $f = f_{surf} * e^{-0.24\,(x)}$ from RG 1.99, Revision 2, where x = the depth into the vessel wall (inches).

(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

Table 8-6 Calculation of the Watts Bar Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at 32 EFPY

Material	R.G. 1.99, Rev. 2 Position	CF(a)	1/4T Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(d) (°F)	M (°F)	ART (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	1.1	123.0	1.03	1.008	47	124.0	0.0	17.0	34.0	**205.0**
Using non-credible surveillance data(e)	2.1	93.6	1.03	1.008	47	94.3	0.0	17.0	34.0	175.3
Lower Shell Forging 04	1.1	51.0	1.05	1.014	5	51.7	0.0	17.0	34.0	90.7
Intermediate to Lower Shell Circumferential Weld Seam W05	1.1	54.0	1.01	1.003	-43	54.2	0.0	27.1	54.2	65.3
Using credible surveillance data(e)	2.1	34.2	1.01	1.003	-43	34.3	0.0	14.0	28.0	19.3
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	1.1	86.0	0.0317	0.227	-22	19.5	0.0	9.8	19.5	17.0
Bottom Head Ring 03	1.1	37.0	0.0716	0.353	-40	13.1	0.0	6.5	13.1	-13.8
Upper to Intermediate Shell Circumferential Weld Seam W06	1.1	41.0	0.0376	0.250	10	10.3	0.0	5.1	10.3	30.5
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	1.1	41.0	0.0861	0.387	10	15.9	0.0	7.9	15.9	41.8

Notes:

(a) Chemistry factors are taken from Table 5-3.

(b) Fluence and fluence factors taken from Table 8-3.

(c) $RT_{NDT(U)}$ values taken from Table 3-1.

(d) Per the guidance of RG 1.99, Revision 2 [Ref. 3], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per RG 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(e) The credibility evaluation for the surveillance weld data in Appendix B determined that the surveillance weld data for Heat # 895075 are deemed credible and the data for the Intermediate Shell Forging are deemed non-credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 8-7 Calculation of the Watts Bar Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at 32 EFPY

Material	R.G. 1.99, Rev. 2 Position	CF(a)	3/4T Fluence(b) (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(d) (°F)	M (°F)	ART (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	1.1	123.0	0.373	0.727	47	89.4	0.0	17.0	34.0	**170.4**
Using non-credible surveillance data(e)	2.1	93.6	0.373	0.727	47	68.1	0.0	17.0	34.0	149.1
Lower Shell Forging 04	1.1	51.0	0.381	0.733	5	37.4	0.0	17.0	34.0	76.4
Intermediate to Lower Shell Circumferential Weld Seam W05	1.1	54.0	0.366	0.722	-43	39.0	0.0	19.5	39.0	35.0
Using credible surveillance data(e)	2.1	34.2	0.366	0.722	-43	24.7	0.0	12.4	24.7	6.4
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	1.1	86.0	0.0115	0.120	-22	10.4	0.0	5.2	10.4	-1.3
Bottom Head Ring 03	1.1	37.0	0.0259	0.201	-40	7.5	0.0	3.7	7.5	-25.1
Upper to Intermediate Shell Circumferential Weld Seam W06	1.1	41.0	0.0136	0.135	10	5.5	0.0	2.8	5.5	21.0
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	1.1	41.0	0.0312	0.225	10	9.2	0.0	4.6	9.2	28.4

Notes:

(a) Chemistry factors are taken from Table 5-3.

(b) Fluence and fluence factors taken from Table 8-3.

(c) $RT_{NDT(U)}$ values taken from Table 3-1.

(d) Per the guidance of RG 1.99, Revision 2 [Ref. 3], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per RG 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(e) The credibility evaluation for the surveillance weld data in Appendix B determined that the surveillance weld data for Heat # 895075 are deemed credible and the data for the Intermediate Shell Forging are deemed non-credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 8-8 Calculation of the Watts Bar Unit 1 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at 48 EFPY

Material	R.G. 1.99, Rev. 2 Position	CF[a]	1/4T Fluence[b] (x 10^{19} n/cm^2, E > 1.0 MeV)	1/4T FF[b]	$RT_{NDT(U)}$[c] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[d] (°F)	M (°F)	ART (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	1.1	123.0	1.59	1.128	47	138.7	0.0	17.0	34.0	**219.7**
Using non-credible surveillance data[e]	2.1	93.6	1.59	1.128	47	105.6	0.0	17.0	34.0	186.6
Lower Shell Forging 04	1.1	51.0	1.64	1.137	5	58.0	0.0	17.0	34.0	97.0
Intermediate to Lower Shell Circumferential Weld Seam W05	1.1	54.0	1.56	1.124	-43	60.7	0.0	28.0	56.0	73.7
Using credible surveillance data[e]	2.1	34.2	1.56	1.124	-43	38.4	0.0	14.0	28.0	23.4
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	1.1	86.0	0.0474	0.284	-22	24.5	0.0	12.2	24.5	26.9
Bottom Head Ring 03	1.1	37.0	0.102	0.421	-40	15.6	0.0	7.8	15.6	-8.8
Upper to Intermediate Shell Circumferential Weld Seam W06	1.1	41.0	0.0564	0.312	10	12.8	0.0	6.4	12.8	35.6
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	1.1	41.0	0.123	0.460	10	18.9	0.0	9.4	18.9	47.7

Notes:

(a) Chemistry factors are taken from Table 5-3.

(b) Fluence and fluence factors taken from Table 8-4.

(c) $RT_{NDT(U)}$ values taken from Table 3-1.

(d) Per the guidance of RG 1.99, Revision 2 [Ref. 3], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per RG 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(e) The credibility evaluation for the surveillance weld data in Appendix B determined that the surveillance weld data for Heat # 895075 are deemed credible and the data for the Intermediate Shell Forging are deemed non-credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 8-9 Calculation of the Watts Bar Unit 1 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at 48 EFPY

Material	R.G. 1.99, Rev. 2 Position	CF[(a)]	3/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	M (°F)	ART (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	1.1	123.0	0.575	0.845	47	104.0	0.0	17.0	34.0	**185.0**
Using non-credible surveillance data[(e)]	2.1	93.6	0.575	0.845	47	79.1	0.0	17.0	34.0	160.1
Lower Shell Forging 04	1.1	51.0	0.595	0.855	5	43.6	0.0	17.0	34.0	82.6
Intermediate to Lower Shell Circumferential Weld Seam W05	1.1	54.0	0.567	0.841	-43	45.4	0.0	22.7	45.4	47.8
Using credible surveillance data[(e)]	2.1	34.2	0.567	0.841	-43	28.8	0.0	14.0	28.0	13.8
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	1.1	86.0	0.0172	0.156	-22	13.4	0.0	6.7	13.4	4.9
Bottom Head Ring 03	1.1	37.0	0.0370	0.248	-40	9.2	0.0	4.6	9.2	-21.7
Upper to Intermediate Shell Circumferential Weld Seam W06	1.1	41.0	0.0204	0.174	10	7.1	0.0	3.6	7.1	24.3
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	1.1	41.0	0.0447	0.275	10	11.3	0.0	5.6	11.3	32.6

Notes:

(a) Chemistry factors are taken from Table 5-3.

(b) Fluence and fluence factors taken from Table 8-4.

(c) $RT_{NDT(U)}$ values taken from Table 3-1.

(d) Per the guidance of RG 1.99, Revision 2 [Ref. 3], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per RG 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(e) The credibility evaluation for the surveillance weld data in Appendix B determined that the surveillance weld data for Heat # 895075 are deemed credible and the data for the Intermediate Shell Forging are deemed non-credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 8-10 Calculation of the Watts Bar Unit 2 ART Values at the 1/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at 32 EFPY

Material	R.G. 1.99, Rev. 2 Position	CF(a)	1/4T Fluence(b) (x 10^{19} n/cm², E > 1.0 MeV)	1/4T FF(b)	$RT_{NDT(U)}$(c) (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ(d) (°F)	M (°F)	ART (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	1.1	31.0	1.13	1.033	14	32.0	0.0	16.0	32.0	**78.0**
Lower Shell Forging 04	1.1	31.0	1.16	1.042	5	32.3	0.0	16.1	32.3	69.6
Intermediate to Lower Shell Circumferential Weld Seam W05	1.1	54.0	1.10	1.027	-50	55.5	0.0	27.7	55.5	60.9
Using credible surveillance data(e)	2.1	34.2	1.10	1.027	-50	35.1	0.0	14.0	28.0	13.1
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	1.1	44.0	0.0324	0.230	-14	10.1	0.0	5.1	10.1	6.2
Bottom Head Ring 03	1.1	37.0	0.145	0.495	-40	18.3	0.0	9.2	18.3	-3.3
Upper to Intermediate Shell Circumferential Weld Seam W06	1.1	41.0	0.0386	0.254	10	10.4	0.0	5.2	10.4	30.8
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	1.1	41.0	0.161	0.518	10	21.3	0.0	10.6	21.3	52.5

Notes:

(a) Chemistry factors are taken from Table 5-4.

(b) Fluence and fluence factors taken from Table 8-5.

(c) $RT_{NDT(U)}$ values taken from Table 3-2.

(d) Per the guidance of RG 1.99, Revision 2 [Ref. 3], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per RG 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed 0.5*ΔRT_{NDT} for either base metals or welds, with or without surveillance data.

(e) The credibility evaluation for the surveillance weld data in Appendix C determined that the surveillance weld data for Heat # 895075 are deemed credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

Table 8-11 Calculation of the Watts Bar Unit 2 ART Values at the 3/4T Location for the Reactor Vessel Beltline and Extended Beltline Materials at 32 EFPY

Material	R.G. 1.99, Rev. 2 Position	CF[(a)]	3/4T Fluence[(b)] (x 10^{19} n/cm², E > 1.0 MeV)	3/4T FF[(b)]	$RT_{NDT(U)}$[(c)] (°F)	Predicted ΔRT_{NDT} (°F)	σ_I (°F)	σ_Δ[(d)] (°F)	M (°F)	ART (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	1.1	31.0	0.407	0.751	14	23.3	0.0	11.6	23.3	**60.6**
Lower Shell Forging 04	1.1	31.0	0.421	0.759	5	23.5	0.0	11.8	23.5	52.1
Intermediate to Lower Shell Circumferential Weld Seam W05	1.1	54.0	0.399	0.745	-50	40.2	0.0	20.1	40.2	30.5
Using credible surveillance data[(e)]	2.1	34.2	0.399	0.745	-50	25.5	0.0	12.7	25.5	1.0
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	1.1	44.0	0.0117	0.122	-14	5.4	0.0	2.7	5.4	-3.3
Bottom Head Ring 03	1.1	37.0	0.0525	0.301	-40	11.1	0.0	5.6	11.1	-17.8
Upper to Intermediate Shell Circumferential Weld Seam W06	1.1	41.0	0.0140	0.137	10	5.6	0.0	2.8	5.6	21.2
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	1.1	41.0	0.0582	0.317	10	13.0	0.0	6.5	13.0	36.0

Notes:

(a) Chemistry factors are taken from Table 5-4.

(b) Fluence and fluence factors taken from Table 8-5.

(c) $RT_{NDT(U)}$ values taken from Table 3-2.

(d) Per the guidance of RG 1.99, Revision 2 [Ref. 3], the base metal σ_Δ = 17°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the base metal σ_Δ = 8.5°F for Position 2.1 with credible surveillance data. Also, per RG 1.99, Revision 2, the weld metal σ_Δ = 28°F for Position 1.1 and Position 2.1 with non-credible surveillance data, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$ for either base metals or welds, with or without surveillance data.

(e) The credibility evaluation for the surveillance weld data in Appendix C determined that the surveillance weld data for Heat # 895075 are deemed credible. Therefore, the Position 2.1 CF can be used with a reduced margin term in lieu of the Position 1.1 CF for Heat # 895075.

9 SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULES

The following surveillance capsule removal schedules (Tables 9-1 and 9-2) meet the recommendations of ASTM E185-82 [Ref. 27] as required by 10 CFR 50, Appendix H [Ref. 28] with consideration of NUREG-1801, Revision 2 [Ref. 29]. Should a capsule be unavailable, a radiologically equivalent capsule, i.e., equivalent lead factor, may be utilized with the appropriate regulatory approval.

It is noted that Watts Bar Unit 1 Capsule V is scheduled to be pulled and tested in the Spring of 2023 after the completion of Cycle 18, which is projected to be at 24.1 EFPY with a fluence of 5.44 x 10^{19} n/cm^2. This planned withdrawal is in-line with the recommendations herein. The results from the planned Capsule V withdrawal will provide Watts Bar Unit 1 surveillance data equivalent to a reactor vessel fluence of 90 EFPY.

Table 9-1
Watts Bar Unit 1 Recommended Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Capsule Lead Factor[(a)]	Removal Time (EFPY)[(b)]	Capsule Fluence (n/cm^2, E > 1.0 MeV)[(a)]
U	56°	4.87	1.20	4.6 x 10^{18}
W	124°	4.78	3.88	1.08 x 10^{19}
X	236°	4.83	6.62	1.75 x 10^{19}
Z	304°	4.76	9.29	2.40 x 10^{19}
V	58.5°	4.11[(c)]	24.1[(d)]	5.44 x 10^{19}
Y	238.5°	4.06[(c)]	[(e)]	Standby

Notes:

(a) Capsule lead factors and fluence values are taken from Section 2.2.1.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Capsule V lead factor is that projected at the end-of-cycle (EOC) 18, the anticipated withdrawal date. Capsule Y lead factor is calculated at 48 EFPY.

(d) Projected EFPY at the EOC 18, the anticipated withdrawal date of Capsule V. This removal ensures the capsule exposure remains below two times the peak reactor pressure vessel (RPV) neutron fluence (2.73 x 10^{19}) at 60 years of operation (48 EFPY).

(e) Capsule Y shall remain inserted in the reactor vessel on standby until needed to fulfill future 10 CFR 50, Appendix H or license renewal requirements.

Table 9-2
Watts Bar Unit 2 Recommended Surveillance Capsule Withdrawal Schedule

Capsule	Capsule Location	Capsule Lead Factor[(a)]	Removal Time (EFPY)[(b)]	Capsule Fluence (n/cm^2, E > 1.0 MeV)[(a)]
U	Dual 34°	4.80	2.00 EFPY (EOC 2)	0.614×10^{19}
W	Single 34°	4.87	6.8 EFPY[(c)] (EOC 6)	1.93×10^{19}
X	Dual 34°	~4.8	6.8 EFPY to 13.8 EFPY[(d)]	1.93×10^{19} to 3.86×10^{19} [(d)]
Z	Single 34°	4.55	Standby[(e)]	Standby[(e)]
V	Dual 31.5°	3.94	Standby[(e)]	Standby[(e)]
Y	Dual 31.5°	3.94	Standby[(e)]	Standby[(e)]

Notes:

(a) Capsule lead factors and fluence values are taken from Section 2.2.2.

(b) Effective Full Power Years (EFPY) from plant startup.

(c) Capsule W should be withdrawn at the outage nearest to but following 6.8 EFPY of operation.

(d) Capsule X should be removed between 10.4 EFPY and 13.8 EFPY if possible. Capsule X must be withdrawn between 6.8 EFPY and 13.8 EFPY in order to satisfy the recommendations of the third capsule for EOL per ASTM E185-82. However, if the capsule is removed after 10.4 EFPY (but still before 13.8 EFPY), this capsule will satisfy the requirements of the third capsule for both EOL and EOLE per ASTM E185-82 and NUREG-1801, Revision 2. Thus, if possible, the capsule should be pulled between 10.4 EFPY and 13.8 EFPY, but the capsule must be pulled between 6.8 EFPY and 13.8 EFPY. The removal EFPY of the third capsule should be revisited at a later date, such as after Capsule W is removed.

(e) Capsules Z, V, and Y should remain in the reactor. If additional metallurgical data is needed, withdrawal and testing of these capsules should be considered. Per ASTM E185-82 and NUREG 1801, Revision 2, it is recommended that the capsules be removed prior to reaching a fluence of two times the peak fluence at EOL. In the event that Capsule W cannot be withdrawn, Capsule Z may be removed instead, during the same outage, to satisfy the ASTM E185-82 requirements for the second withdrawn capsule.

10 REFERENCES

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4. Code of Federal Regulations, 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,” U.S. Nuclear Regulatory Commission, Federal Register, November 29, 2019.
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APPENDIX A : VALIDATION OF THE RADIATION TRANSPORT MODELS BASED ON NEUTRON DOSIMETRY MEASUREMENTS

A.1 NEUTRON DOSIMETRY

Comparisons of measured dosimetry results to both the calculated and least-squares adjusted values for Watts Bar Unit 1 and Unit 2 surveillance capsule dosimetry are provided in this appendix. The sensor sets were analyzed in accordance with the dosimetry evaluation methodology described in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" [Ref. A-1]. One of the main purposes for providing this material is to demonstrate that the overall measurements agree with the calculated and least-squares adjusted values to within ± 20% as specified by Regulatory Guide 1.190, thus serving to validate the calculated neutron exposures previously reported in Section 2.2.

A.1.1 Sensor Reaction Rate Determinations

In this section, the results of the evaluations of withdrawn capsules are presented. The capsule designations unique to each unit, location within each reactor unit, and time of withdrawal are as follows:

Unit 1			
Capsule	**Azimuthal Location**	**Withdrawal Time**	**Irradiation Time (EFPY)**
U	34° Dual	End of Cycle 1	1.20
W	34° Single	End of Cycle 3	3.88
X	34° Dual	End of Cycle 5	6.62
Z	34° Single	End of Cycle 7	9.29

Unit 2			
Capsule	**Azimuthal Location**	**Withdrawal Time**	**Irradiation Time (EFPY)**
U	34º Dual	End of Cycle 2	2.00

The passive neutron sensors included in these evaluations are summarized as follows:

Sensor Material	Reaction Of Interest	Capsules				
		Unit 1				Unit 2
		U	W	X	Z	U
Copper	Cu-63 (n,α) Co-60	X	X	X	X	X
Iron	Fe-54 (n,p) Mn-54	X	X	X	X	X
Nickel	Ni-58 (n,p) Co-58	X	X	X	X	X
Uranium-238	U-238 (n,f) Cs-137	X	X	X	X	X
Neptunium-237	Np-237 (n,f) Cs-137	X	X	X	X	X
Cobalt-Aluminum[(a)]	Co-59 (n,γ) Co-60	X	X	X	X	X

Notes:

a) The cobalt-aluminum and uranium sensors include both bare and cadmium-covered sensors.

The dosimetry monitors were located at the radial center of the material test specimen array. As such, gradient corrections were not required for these reaction rates. Pertinent physical and nuclear characteristics of the passive neutron sensors analyzed are as follows:

Reaction of Interest	Atomic Weight (g/g-atom)	Target Atom Fraction	Product Half-life (days)	Fission Yield (%)	90% Response Range[(a)] (MeV)
Cu-63 (n,α) Co-60	63.546	0.6917	1925.28	-	4.53–11.0
Fe-54 (n,p) Mn-54	55.845	0.05845	312.13	-	2.27–7.54
Ni-58 (n,p) Co-58	58.693	0.68077	70.86	-	1.98–7.51
U-238 (n,f) Cs-137	238.051	1.00	10975.76	6.0045[(b)]	1.44–6.69
Np-237 (n,f) Cs-137	237.048	1.00	10975.76	6.2129[(b)]	0.68-5.61
Co-59 (n,γ) Co-60	58.933	0.0015	1925.28	-	non-threshold

Note: (a) Energies between which 90% of activity is produced (U-235 fission spectrum) [Ref. A-5]

(b) For Unit 2 Capsule U dosimetry, fission yields of 6.02% for U-238 and 6.27% for Np-237 were used. The impact on analytical results is negligible.

The use of passive monitors does not yield a direct measure of the energy-dependent neutron exposure rate at the point of interest. Rather, the activation or fission process is a measure of the integrated effect that the time- and energy-dependent neutron exposure rate has on the target material over the course of the irradiation period. An accurate assessment of the average neutron exposure rate incident on the various monitors may be derived from the activation measurements only if the irradiation parameters are well known. In particular, the following variables are of interest:

- the measured specific activity of each monitor,
- the physical characteristics of each monitor,
- the operating history of the reactor,
- the energy response of each monitor, and
- the neutron energy spectrum at the monitor location.

The radiometric counting followed established ASTM procedures.

The irradiation history of the reactor over the relevant irradiation period for each capsule was based on the monthly power generation of the respective operating unit from initial reactor criticality through the end of the pertinent dosimetry evaluation period. For the sensor sets utilized in the surveillance capsules, the half-lives of the product isotopes are long enough that a monthly histogram describing reactor operation has proven to be an adequate representation for use in radioactive decay corrections for the reactions of interest in the exposure evaluations. The irradiation history applicable to capsules U, W, X and Z of Watts Bar Unit 1 is given in Table A-1. The irradiation history applicable to capsule U of Watts Bar Unit 2 is given in Table A-2.

Having the measured specific activities, the physical characteristics of the sensors, and the operating history of the reactor, reaction rates referenced to full-power operation were determined from the following equation:

$$R = \frac{A}{N_0 F Y \sum \frac{P_j}{P_{ref}} C_j \left[1 - e^{-\lambda t_j}\right]\left[e^{-\lambda t_{d,j}}\right]}$$

where:

R	=	Reaction rate averaged over the irradiation period and referenced to operation at a core power level of P_{ref} (rps/nucleus).
A	=	Measured specific activity (dps/g).
N_0	=	Number of target element atoms per gram of sensor.
F	=	Atom fraction of the target isotope in the target element.
Y	=	Number of product atoms produced per reaction.
P_j	=	Average core power level during irradiation period j (MW).

P_{ref} = Maximum or reference power level of the reactor (MW).

C_j = Calculated ratio of ϕ (E > 1.0 MeV) during irradiation period j to the time weighted average ϕ (E > 1.0 MeV) over the entire irradiation period.

λ = Decay constant of the product isotope (1/sec).

t_j = Length of irradiation period j (sec).

$t_{d,j}$ = Decay time following irradiation period j (sec).

The summation is carried out over the total number of monthly intervals comprising the irradiation period.

In the equation describing the reaction rate calculation, the ratio $[P_j]/[P_{ref}]$ accounts for month-by-month variation of reactor core power level within any given fuel cycle as well as over multiple fuel cycles. The ratio C_j, which was calculated for each fuel cycle using the transport methodology discussed in Section 2.2, accounts for the change in sensor reaction rates caused by variations in exposure rate level induced by changes in core spatial power distributions from fuel cycle to fuel cycle. For a single-cycle irradiation, C_j is normally taken to be 1.0. However, for multiple-cycle irradiations, the additional C_j term should be employed. The impact of changing exposure rate levels for constant power operation can be quite significant for sensor sets that have been irradiated for many cycles in a reactor that has transitioned from non-low-leakage to low-leakage fuel management or for sensor sets contained in surveillance capsules that have been moved from one capsule location to another. The fuel-cycle-specific neutron exposure rates are used to compute the cycle-specific C_j values at the radial and azimuthal center of the respective capsules at core midplane.

Prior to using the measured reaction rates in the least-squares evaluations of the dosimetry sensor sets, additional corrections were made to the U-238 measurements to account for the presence of U-235 impurities in the sensors, as well as to adjust for the build-in of plutonium isotopes over the course of the irradiation. Corrections were also made to the U-238 and Np-237 sensor reaction rates to account for gamma-ray-induced fission reactions that occurred over the course of the surveillance capsule irradiations. The correction factors corresponding to the Watts Bar Units 1 & 2 fission sensor reaction rates are summarized as follows:

Correction	**Unit 1**				**Unit 2**
	Capsule U	**Capsule W**	**Capsule X**	**Capsule Z**	**Capsule U**
U-235 Impurity/Pu Build-in	0.8663	0.8425	0.8175	0.7963	0.8605
U-238 (γ,f)	0.9648	0.9684	0.9659	0.9686	0.9653
Net U-238 Correction	0.8358	0.8159	0.7896	0.7713	0.8306
Np-237 (γ,f)	0.9902	0.9912	0.9903	0.9912	0.9903

The correction factors were applied in a multiplicative fashion to the decay-corrected cadmium-covered uranium fission sensor reaction rates.

Results of the sensor reaction rate determinations for Watts Bar Unit 1 are given in Table A-3 through Table A-6. Results of the sensor reaction rate determinations for Watts Bar Unit 2 are given in Table A-7. In these tables, the measured specific activities, decay-corrected saturated specific activities, and computed reaction rates for each sensor are listed.

A.1.2 Least-Squares Evaluation of Sensor Sets

Least-squares adjustment methods provide the capability of combining the measurement data with the corresponding neutron transport calculations resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best-estimates for key exposure parameters such as fluence rate (E > 1.0 MeV) or dpa/s along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross-sections, and the calculated neutron energy spectrum within their respective uncertainties. For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\varphi_g \pm \delta_{\varphi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, ϕ_g, through the multigroup dosimeter reaction cross-sections, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the Watts Bar Unit 1 and Unit 2 dosimetry, the FERRET code [Ref. A-2] was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine the best-estimate values of exposure parameters (fluence rate (E > 1.0 MeV) and dpa) and their associated uncertainties.

The application of the least-squares methodology requires the following input:

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.

2. The measured reaction rates and associated uncertainty for each sensor contained in the multiple foil set.

3. The energy-dependent dosimetry reaction cross-sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For each operating unit, the calculated neutron spectrum was obtained from the results of plant-specific neutron transport calculations described in Section 2.2. The sensor reaction rates were derived from the measured specific activities using the procedures described in Section A.1.1. The dosimetry reaction cross-sections and uncertainties were obtained from the SNLRML dosimetry cross-section library [Ref. A-3].

The uncertainties associated with the measured reaction rates, dosimetry cross-sections, and calculated neutron spectrum were input to the least-squares procedure in the form of variances and covariances. The assignment of the input uncertainties followed the guidance provided in ASTM Standard E944, "Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance" [Ref. A-4].

The following provides a summary of the uncertainties associated with the least-squares evaluation of the Watts Bar Unit 1 and Unit 2 surveillance capsule sensor sets.

Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, irradiation history corrections, and corrections for competing reactions. A high level of accuracy in the reaction rate determinations is ensured by utilizing laboratory procedures that conform to the ASTM National Consensus Standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input to the least-squares evaluation:

Reaction	**Uncertainty**
^{63}Cu (n,α) ^{60}Co	5%
^{54}Fe (n,p) ^{54}Mn	5%
^{58}Ni (n,p) ^{58}Co	5%
^{59}Co (n,γ) ^{60}Co	5%
^{238}U (n,f) FP	10%
^{237}Np (n,f) FP	10%

These uncertainties are given at the 1σ level.

Dosimetry Cross-Section Uncertainties

The reaction rate cross-sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross-sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross-sections and uncertainties are provided in a fine multigroup structure for use in least-squares adjustment applications. These cross-sections were compiled from recent cross-section evaluations, and they have been tested for accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination, as well as in the fluence and energy characterization of 14 MeV neutron sources.

For sensors included in the Watts Bar Unit 1 and Unit 2 surveillance programs, the following uncertainties in the fission spectrum averaged cross-sections are provided in the SNLRML documentation package.

Reaction	Uncertainty
Cu-63 (n,α) Co-60	4.08–4.16%
Fe-54 (n,p) Mn-54	3.05–3.11%
Ni-58 (n,p) Co-58	4.49–4.56%
Co-59 (n,γ) Co-60	0.79–3.59%
U-238 (n,f)	0.54–0.64%
Np-237 (n,f)	10.32-10.97%

These tabulated ranges provide an indication of the dosimetry cross-section uncertainties associated with the sensor sets used in LWR irradiations.

Calculated Neutron Spectrum

The neutron spectra inputs to the least-squares adjustment procedure were obtained directly from the results of plant-specific transport calculations for each surveillance capsule irradiation period and location. The spectrum for each capsule was input in an absolute sense (rather than as simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements.

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks and the dosimetry cross-section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties R_g and $R_{g'}$ specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta\, e^{-H}$$

Where:

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when $g = g'$, and is 0.0 otherwise.

The set of parameters defining the input covariance matrix for the calculated spectra for each plant was as follows:

Exposure Rate Normalization Uncertainty (R_n)	15%
Exposure Rate Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%
(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Exposure Rate Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

A.1.3 Comparisons of Measurements and Calculations

Results of the least-squares evaluations for Watts Bar Unit 1 are provided in Table A-8 through Table A-11. Results of the least-squares evaluations for Watts Bar Unit 2 are provided in Table A-12. In these tables, measured, calculated, and best-estimate values for sensor reaction rates are given. Also provided in these tabulations are ratios of the measured reaction rates to both the calculated and least-squares adjusted reaction rates. These ratios of measured-to-calculated (M/C) and measured-to-best estimate (M/BE) illustrate the consistency of the fit of the calculated neutron energy spectra to the measured reaction rates both before and after adjustment. Additionally, comparisons of the calculated and best-estimate values of neutron fluence rate (E > 1.0 MeV) and iron atom displacement rate are tabulated along with the best estimate-to-calculated (BE/C) ratios observed for each of the capsules.

The data comparisons provided in Table A-8 through Table A-11 and Table A-12 show that the adjustments to the calculated spectra for Watts Bar Units 1 and 2, respectively, are relatively small and within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections. Further, these results indicate that the use of the least-squares evaluation results in a reduction in the uncertainties associated with the exposure of the surveillance capsules. From Section 2.2, the calculational uncertainty is specified as 13% at the 1σ level.

Further comparisons of the measurement results with calculations for Watts Bar Unit 1 are given in Table A-13 and Table A-14. Similar comparisons for Watts Bar Unit 2 are given in Table A-15 and Table A-16. In Table A-13 and Table A-15, calculations of individual threshold sensor reaction rates are compared directly with the corresponding measurements. These threshold reaction rate comparisons provide a good evaluation of the accuracy of the fast neutron portion of the calculated energy spectra. In Table A-14 and Table A-16, calculations of fast neutron exposure rates in terms of fast neutron (E > 1.0 MeV) fluence rate and dpa/s are compared with the best-estimate results obtained from the least-squares evaluation of the capsule dosimetry results. These comparisons yield consistent and similar results with all measurement-to-calculation comparisons falling within the 20% limits specified as the acceptance criteria in Regulatory Guide 1.190.

In the case of the direct comparison of the measured and calculated sensor reaction rates for Watts Bar Unit 1, for the individual threshold foils considered in the least-squares analysis, the M/C comparisons of the fast neutron threshold reactions range from 0.91 to 1.27. The overall average M/C ratio is 1.04 with an associated standard deviation of 11.0%. In the case of the comparison of the best-estimate and calculated fast neutron exposure parameters for Unit 1, the BE/C comparisons are 1.01 and 1.02 for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, respectively.

In the case of the direct comparison of the measured and calculated sensor reaction rates for Watts Bar Unit 2, for the individual threshold foils considered in the least-squares analysis, the M/C comparisons of the fast neutron threshold reactions range from 0.71 to 0.99. The overall average M/C ratio is 0.90 with an associated standard deviation of 12.6%. In the case of the comparison of the best-estimate and calculated fast neutron exposure parameters for Unit 2, the BE/C comparisons are 0.91 and 0.92 for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, respectively.

Based on these comparisons, it is concluded that the calculated fast neutron exposures provided in Section 2.2 of this report are valid for use in the assessment of the condition of the materials comprising the beltline region and the extended beltline region of the Watts Bar Units 1 & 2 reactor pressure vessels.

Table A-1: Monthly Thermal Generation for the Watts Bar Unit 1 Reactor

Cycle 1		Cycle 2		Cycle 3	
Month	Generation (MWt-h)	Month	Generation (MWt-h)	Month	Generation (MWt-h)
Jan-96	9519	Oct-97	709914	Apr-99	1068631
Feb-96	49773	Nov-97	2449654	May-99	2454763
Mar-96	475248	Dec-97	2527971	Jun-99	2454685
Apr-96	999029	Jan-98	2523492	Jul-99	2536887
May-96	1713718	Feb-98	2142012	Aug-99	2526645
Jun-96	2348718	Mar-98	2180599	Sep-99	2455157
Jul-96	2523691	Apr-98	2370444	Oct-99	2540350
Aug-96	2525629	May-98	2535488	Nov-99	2454874
Sep-96	2184725	Jun-98	2445551	Dec-99	2536956
Oct-96	1114619	Jul-98	2531951	Jan-00	2536601
Nov-96	2202224	Aug-98	2486237	Feb-00	2373327
Dec-96	2523130	Sep-98	2429447	Mar-00	2536915
Jan-97	1956638	Oct-98	2470835	Apr-00	2451520
Feb-97	2147746	Nov-98	2452741	May-00	2536436
Mar-97	1645462	Dec-98	2535907	Jun-00	2455171
Apr-97	2236507	Jan-99	2280206	Jul-00	2534880
May-97	2519097	Feb-99	1519605	Aug-00	2279989
Jun-97	2237128	Mar-99	0	Sep-00	570521
Jul-97	2399891				
Aug-97	1934439				
Sep-97	262258				

Table A-1 (continued): Monthly Thermal Generation for the Watts Bar Unit 1 Reactor

Cycle 4		**Cycle 5**		**Cycle 6**	
Month	Generation (MWt-h)	Month	Generation (MWt-h)	Month	Generation (MWt-h)
Oct-00	1906668	Mar-02	759955	Oct-03	797571
Nov-00	2455222	Apr-02	2486032	Nov-03	2489411
Dec-00	2536907	May-02	2041632	Dec-03	2572114
Jan-01	2539043	Jun-02	2489640	Jan-04	2356325
Feb-01	2323719	Jul-02	2440231	Feb-04	2406294
Mar-01	2571100	Aug-02	2572406	Mar-04	2572166
Apr-01	2486145	Sep-02	2489113	Apr-04	2483973
May-01	2572676	Oct-02	2573865	May-04	2568656
Jun-01	2384162	Nov-02	2489493	Jun-04	2489087
Jul-01	1723109	Dec-02	2547822	Jul-04	2571263
Aug-01	2572727	Jan-03	2572408	Aug-04	2571859
Sep-01	2298210	Feb-03	2322599	Sep-04	2198177
Oct-01	2576028	Mar-03	2052502	Oct-04	2575547
Nov-01	2489629	Apr-03	2485939	Nov-04	2489216
Dec-01	2423020	May-03	2572277	Dec-04	2571088
Jan-02	2572218	Jun-03	2487655	Jan-05	2572113
Feb-02	1970262	Jul-03	2570050	Feb-05	1731861
		Aug-03	2409124		
		Sep-03	485359		

Cycle 7	
Mar-05	3388
Apr-05	2375517
May-05	2571958
Jun-05	2488183
Jul-05	2572181
Aug-05	2572139
Sep-05	2361712
Oct-05	2575678
Nov-05	2489136
Dec-05	2572013
Jan-06	2570408
Feb-06	2147690
Mar-06	2517604
Apr-06	2485651
May-06	2465114
Jun-06	394229
Jul-06	2508244
Aug-06	2184917
Sep-06	809562

Table A-2: Monthly Thermal Generation for the Watts Bar Unit 2 Reactor

Cycle 1		Cycle 2	
Month	Generation (MWt-h)	Month	Generation (MWt-h)
May-16	30290	Dec-17	1491562
Jun-16	347922	Jan-18	2530416
Jul-16	1056046	Feb-18	2287280
Aug-16	664736	Mar-18	2532054
Sep-16	314358	Apr-18	2210328
Oct-16	1745340	May-18	2440366
Nov-16	2369963	Jun-18	1738791
Dec-16	2526323	Jul-18	2432179
Jan-17	2530416	Aug-18	2020404
Feb-17	2284824	Sep-18	2451827
Mar-17	1588980	Oct-18	2534509
Apr-17	0	Nov-18	2455101
May-17	0	Dec-18	2536965
Jun-17	819	Jan-19	2535328
Jul-17	18829	Feb-19	1829660
Aug-17	2011398	Mar-19	2534509
Sep-17	2446915	Apr-19	970907
Oct-17	2184950		
Nov-17	0		

Table A-3: Measured Sensor Reaction Rates for Unit 1 In-Vessel Surveillance Capsule U – Dual Capsule Holder; 34° Azimuthal, 207.32 cm Radial, Core Midplane Location; Cycle 1 Irradiation

Reaction	Measured Activity (dps/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	5.05E+04	3.68E+05	5.62E-17		
^{63}Cu (n,α) ^{60}Co	5.34E+04	3.90E+05	5.94E-17		
^{63}Cu (n,α) ^{60}Co	5.25E+04	3.83E+05	5.84E-17	5.80E-17	5.80E-17
^{54}Fe (n,p) ^{54}Mn	1.62E+06	3.86E+06	6.12E-15		
^{54}Fe (n,p) ^{54}Mn	1.71E+06	4.07E+06	6.46E-15		
^{54}Fe (n,p) ^{54}Mn	1.71E+06	4.07E+06	6.46E-15	6.34E-15	6.34E-15
^{58}Ni (n,p) ^{58}Co	1.20E+07	5.97E+07	8.54E-15		
^{58}Ni (n,p) ^{58}Co	1.25E+07	6.21E+07	8.90E-15		
^{58}Ni (n,p) ^{58}Co	1.26E+07	6.26E+07	8.97E-15	8.80E-15	8.80E-15
^{238}U (n,f) ^{137}Cs	2.34E+05	8.64E+06	5.69E-14	5.69E-14	4.76E-14
^{237}Np (n,f) ^{137}Cs	1.94E+06	7.16E+07	4.54E-13	4.54E-13	4.50E-13
^{59}Co (n,γ) ^{60}Co	1.55E+07	1.13E+08	7.38E-12		
^{59}Co (n,γ) ^{60}Co	1.28E+07	9.34E+07	6.09E-12		
^{59}Co (n,γ) ^{60}Co	1.43E+07	1.04E+08	6.80E-12		
^{59}Co (n,γ) ^{60}Co	1.19E+07	8.68E+07	5.66E-12		
^{59}Co (n,γ) ^{60}Co	1.41E+07	1.03E+08	6.71E-12		
^{59}Co (n,γ) ^{60}Co	1.22E+07	8.90E+07	5.81E-12	6.41E-12	6.41E-12
(Cd)^{59}Co (n,γ) ^{60}Co	7.59E+06	5.54E+07	3.61E-12		
(Cd)^{59}Co (n,γ) ^{60}Co	7.06E+06	5.15E+07	3.36E-12		
(Cd)^{59}Co (n,γ) ^{60}Co	7.10E+06	5.18E+07	3.38E-12	3.45E-12	3.45E-12
Note: Measured activity decay corrected to January 30, 1998. (Cd) denotes cadmium shielded.					

Table A-4: Measured Sensor Reaction Rates for Unit 1 In-Vessel Surveillance Capsule W – Single Capsule Holder; 34° Azimuthal, 207.32 cm Radial, ore Midplane Location; Cycles 1 through 3 Irradiation

Reaction	Measured Activity (dps/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	1.84E+05	4.87E+05	5.14E-17		
^{63}Cu (n,α) ^{60}Co	1.95E+05	5.17E+05	5.45E-17		
^{63}Cu (n,α) ^{60}Co	1.94E+05	5.14E+05	5.42E-17	5.34E-17	5.34E-17
^{54}Fe (n,p) ^{54}Mn	4.63E+07	5.95E+07	5.52E-15		
^{54}Fe (n,p) ^{54}Mn	4.84E+07	6.22E+07	5.77E-15		
^{54}Fe (n,p) ^{54}Mn	4.70E+07	6.04E+07	5.61E-15	5.63E-15	5.63E-15
^{58}Ni (n,p) ^{58}Co	6.16E+07	7.79E+07	7.59E-15		
^{58}Ni (n,p) ^{58}Co	6.34E+07	8.02E+07	7.81E-15		
^{58}Ni (n,p) ^{58}Co	6.16E+07	7.79E+07	7.59E-15	7.66E-15	7.66E-15
^{238}U (n,f) ^{137}Cs	3.97E+05	4.69E+06	3.09E-14	3.09E-14	2.52E-14
^{237}Np (n,f) ^{137}Cs	4.62E+06	5.46E+07	3.46E-13	3.46E-13	3.43E-13
^{59}Co (n,γ) ^{60}Co	2.03E+10	5.38E+10	5.26E-12		
^{59}Co (n,γ) ^{60}Co	1.78E+10	4.72E+10	4.61E-12		
^{59}Co (n,γ) ^{60}Co	1.88E+10	4.98E+10	4.87E-12		
^{59}Co (n,γ) ^{60}Co	1.55E+10	4.11E+10	4.02E-12		
^{59}Co (n,γ) ^{60}Co	1.94E+10	5.14E+10	5.03E-12		
^{59}Co (n,γ) ^{60}Co	1.64E+10	4.34E+10	4.25E-12	4.67E-12	4.67E-12
(Cd)^{59}Co (n,γ) ^{60}Co	1.01E+10	2.68E+10	2.62E-12		
(Cd)^{59}Co (n,γ) ^{60}Co	9.68E+09	2.56E+10	2.51E-12		
(Cd)^{59}Co (n,γ) ^{60}Co	9.95E+09	2.64E+10	2.58E-12	2.57E-12	2.57E-12

Note: Measured activity decay corrected to September 10, 2000. (Cd) denotes cadmium shielded.

Table A-5: Measured Sensor Reaction Rates for Unit 1 Reaction Rates for In-Vessel Surveillance Capsule X – Dual Capsule Holder; 34° Azimuthal, 207.32 cm Radial, Core Midplane Location; Cycles 1 through 5 Irradiation

Reaction	Measured Activity (dps/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.52E+05	2.89E+05	4.42E-17		
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.55E+05	2.95E+05	4.50E-17		
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.56E+05	2.97E+05	4.53E-17	4.48E-17	4.48E-17
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.84E+06	2.75E+06	4.37E-15		
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.96E+06	2.93E+06	4.65E-15		
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.90E+06	2.84E+06	4.51E-15	4.51E-15	4.51E-15
$^{58}Ni\ (n,p)\ ^{58}Co$	1.59E+07	4.56E+07	6.53E-15		
$^{58}Ni\ (n,p)\ ^{58}Co$	1.68E+07	4.82E+07	6.90E-15		
$^{58}Ni\ (n,p)\ ^{58}Co$	1.69E+07	4.85E+07	6.94E-15	6.79E-15	6.79E-15
$^{238}U\ (n,f)\ ^{137}Cs$	6.61E+05	4.76E+06	3.13E-14	3.13E-14	2.47E-14
$^{237}Np\ (n,f)\ ^{137}Cs$	5.82E+06	4.19E+07	2.65E-13	2.65E-13	2.63E-13
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.13E+07	5.96E+0787	3.89E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.54E+07	6.74E+07	4.40E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.55E+07	6.76E+07	4.41E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.75E+07	5.24E+07	3.42E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.93E+07	5.58E+07	3.64E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.41E+07	6.49E+07	4.24E-12	4.00E-12	4.00E-12
$(Cd)^{59}Co\ (n,\gamma)\ ^{60}Co$	1.87E+07	3.56E+07	2.32E-12		
$(Cd)^{59}Co\ (n,\gamma)\ ^{60}Co$	1.79E+07	3.41E+07	2.22E-12	2.27E-12	2.27E-12
Note: Measured activity decay corrected to November 26, 2003. (Cd) denotes cadmium shielded.					

Table A-6: Measured Sensor Reaction Rates for Unit 1 In-Vessel Surveillance Capsule Z – Single Capsule Holder; 34° Azimuthal, 207.32 cm Radial, Core Midplane Location; Cycles 1 through 7 Irradiation

Reaction	Measured Activity (dps/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.62E+05	2.73E+05	4.16E-17		
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.67E+05	2.81E+05	4.29E-17		
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	1.61E+05	2.71E+05	4.14E-17	4.20E-17	4.20E-17
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.35E+06	2.68E+06	4.26E-15		
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.38E+06	2.74E+06	4.35E-15		
$^{54}Fe\ (n,p)\ ^{54}Mn$	1.35E+06	2.68E+06	4.26E-15	4.29E-15	4.29E-15
$^{58}Ni\ (n,p)\ ^{58}Co$	3.14E+06	4.15E+07	5.94E-15		
$^{58}Ni\ (n,p)\ ^{58}Co$	3.17E+06	4.19E+07	6.00E-15		
$^{58}Ni\ (n,p)\ ^{58}Co$	3.09E+06	4.08E+07	5.84E-15	5.93E-15	5.93E-15
$^{238}U\ (n,f)\ ^{137}Cs$	9.49E+05	5.09E+06	3.35E-14	3.35E-14	2.58E-14
$^{237}Np\ (n,f)\ ^{137}Cs$	7.14E+06	3.83E+07	2.43E-13	2.43E-13	2.40E-13
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.60E+07	6.06E+07	3.95E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.99E+07	5.03E+07	3.28E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	2.92E+07	4.92E+07	3.21E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.28E+07	5.52E+07	3.60E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.02E+07	5.09E+07	3.32E-12		
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.47E+07	5.84E+07	3.81E-12	3.53E-12	3.53E-12
$(Cd)^{59}Co\ (n,\gamma)\ ^{60}Co$	1.84E+07	3.10E+07	2.02E-12		
$(Cd)^{59}Co\ (n,\gamma)\ ^{60}Co$	1.75E+07	2.95E+07	1.92E-12		
$(Cd)^{59}Co\ (n,\gamma)\ ^{60}Co$	1.78E+07	3.00E+07	1.96E-12	1.97E-12	1.97E-12

Note: Measured activity decay corrected to May 16, 2007. (Cd) denotes cadmium shielded.

Table A-7: Measured Sensor Reaction Rates for Unit 2 In-Vessel Surveillance Capsule U – Dual Capsule Holder; 34° Azimuthal, 207.32 cm Radial, Core Midplane Location; Cycles 1 through 2 Irradiation

Reaction	Measured Activity (dps/g)	Radially Corrected Saturated Activity (dps/g)	Reaction Rate (rps/atom)	Average Reaction Rate (rps/atom)	Corrected Average Reaction Rate (rps/atom)
^{63}Cu (n,α) ^{60}Co	5.71E+04	2.63E+05	4.02E-17		
^{63}Cu (n,α) ^{60}Co	6.04E+04	2.79E+05	4.25E-17		
^{63}Cu (n,α) ^{60}Co	5.80E+04	2.67E+05	4.08E-17	4.11E-17	4.11E-17
^{54}Fe (n,p) ^{54}Mn	1.96E+06	3.31E+06	5.25E-15		
^{54}Fe (n,p) ^{54}Mn	2.09E+06	3.53E+06	5.60E-15		
^{54}Fe (n,p) ^{54}Mn	1.83E+06	3.09E+06	4.91E-15	5.25E-15	5.25E-15
^{58}Ni (n,p) ^{58}Co	2.08E+07	4.98E+07	7.13E-15		
^{58}Ni (n,p) ^{58}Co	2.14E+07	5.12E+07	7.33E-15		
^{58}Ni (n,p) ^{58}Co	2.13E+07	5.10E+07	7.30E-15	7.25E-15	7.25E-15
^{238}U (n,f) ^{137}Cs	1.75E+05	3.94E+06	2.58E-14	2.58E-14	2.15E-14
^{237}Np (n,f) ^{137}Cs	2.21E+06	4.97E+07	3.12E-13	3.12E-13	3.09E-13
^{59}Co (n,γ) ^{60}Co	1.73E+07	7.98E+07	5.20E-12		
^{59}Co (n,γ) ^{60}Co	1.59E+07	7.33E+07	4.78E-12		
^{59}Co (n,γ) ^{60}Co	1.62E+07	7.47E+07	4.87E-12		
^{59}Co (n,γ) ^{60}Co	1.39E+07	6.41E+07	4.18E-12		
^{59}Co (n,γ) ^{60}Co	1.70E+07	7.84E+07	5.11E-12		
^{59}Co (n,γ) ^{60}Co	1.49E+07	6.87E+07	4.48E-12	4.77E-12	4.77E-12
(Cd)^{59}Co (n,γ) ^{60}Co	8.73E+06	4.03E+07	2.63E-12		
(Cd)^{59}Co (n,γ) ^{60}Co	9.16E+06	4.22E+07	2.76E-12		
(Cd)^{59}Co (n,γ) ^{60}Co	9.11E+06	4.20E+07	2.74E-12	2.71E-12	2.71E-12

Note: Measured activity decay corrected to June 25, 2019. (Cd) denotes cadmium shielded.

Table A-8: Least-Squares Evaluation of Dosimetry in Watts Bar Unit 1 Surveillance Capsule U (Dual Capsule Holder, 34° Azimuth, Cycle 1 Irradiation)

Reaction	Reaction Rate (rps/atom)			M/C	M/BE	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
^{63}Cu (n,α) ^{60}Co	5.80E-17	5.56E-17	5.63E-17	1.04	1.03	1.01
^{54}Fe (n,p) ^{54}Mn	6.34E-15	6.61E-15	6.57E-15	0.96	0.96	0.99
^{58}Ni (n,p) ^{58}Co	8.80E-15	9.37E-15	9.28E-15	0.94	0.95	0.99
^{238}U (n,f) ^{137}Cs	4.75E-14	3.75E-14	3.90E-14	1.27	1.22	1.04
^{237}Np (n,f) ^{137}Cs	4.49E-13	3.88E-13	4.33E-13	1.16	1.04	1.11
^{59}Co (n,γ) ^{60}Co	6.41E-12	6.27E-12	6.36E-12	1.02	1.01	1.01
^{59}Co(Cd) (n,γ) ^{60}Co	3.45E-12	4.13E-12	3.49E-12	0.84	0.99	0.85
Fast Reaction Threshold Foil Average				1.07	1.04	1.03
Standard deviation (%)				13.0	10.4	4.9

	C	BE	%Unc	BE/C		
Fluence Rate E>1.0 MeV (n/cm^2-s)	1.22E+11	1.29E+11	6	1.06		χ^2/DOF
DPA/s	2.43E-10	2.61E-10	8	1.08		0.988

Table A-9: Least-Squares Evaluation of Dosimetry in Watts Bar Unit 1 Surveillance Capsule W (Single Capsule Holder, 34° Azimuth, Cycles 1 Through 3 Irradiation)

Reaction	Reaction Rate (rps/atom)			M/C	M/BE	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
^{63}Cu (n,α) ^{60}Co	5.34E-17	4.29E-17	5.18E-17	1.25	1.03	1.21
^{54}Fe (n,p) ^{54}Mn	5.63E-15	4.94E-15	5.57E-15	1.14	1.01	1.13
^{58}Ni (n,p) ^{58}Co	7.66E-15	6.98E-15	7.75E-15	1.10	0.99	1.11
^{238}U (n,f) ^{137}Cs	2.52E-14	2.75E-14	2.97E-14	0.91	0.85	1.08
^{237}Np (n,f) ^{137}Cs	3.43E-13	2.83E-13	3.20E-13	1.21	1.08	1.13
^{59}Co (n,γ) ^{60}Co	4.67E-12	4.17E-12	4.63E-12	1.12	1.01	1.11
^{59}Co(Cd) (n,γ) ^{60}Co	2.57E-12	2.80E-12	2.60E-12	0.92	0.99	0.93
Fast Reaction Threshold Foil Average				1.12	0.99	1.13
Standard deviation (%)				11.8	8.7	4.3

	C	BE	%Unc	BE/C		
Fluence Rate E>1.0 MeV (n/cm^2-s)	8.91E+10	9.48E+10	6	1.06		χ^2/DOF
DPA/s	1.77E-10	1.89E-10	8	1.07		0.490

Table A-10: Least-Squares Evaluation of Dosimetry in Watts Bar Unit 1 Surveillance Capsule X (Dual Capsule Holder, 34° Azimuth, Cycles 1 Through 5 Irradiation)

Reaction	Reaction Rate (rps/atom)			M/C	M/BE	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
^{63}Cu (n,α) ^{60}Co	4.48E-17	4.13E-17	4.36E-17	1.09	1.03	1.06
^{54}Fe (n,p) ^{54}Mn	4.51E-15	4.71E-15	4.67E-15	0.96	0.96	0.99
^{58}Ni (n,p) ^{58}Co	6.79E-15	6.65E-15	6.68E-15	1.02	1.02	1.00
^{238}U (n,f) ^{137}Cs	2.47E-14	2.61E-14	2.55E-14	0.95	0.97	0.98
^{237}Np (n,f) ^{137}Cs	2.63E-13	2.66E-13	2.59E-13	0.99	1.01	0.98
^{59}Co (n,γ) ^{60}Co	4.00E-12	4.23E-12	3.98E-12	0.95	1.01	0.94
^{59}Co(Cd) (n,γ) ^{60}Co	2.27E-12	2.78E-12	2.30E-12	0.82	0.99	0.83
Fast Reaction Threshold Foil Average				1.00	1.00	1.00
Standard deviation (%)				5.6	3.1	3.3
	C	BE	%Unc	BE/C		
Fluence Rate E>1.0 MeV (n/cm^2-s)	8.41E+10	8.14E+10	6	0.97		χ^2/DOF
DPA/s	1.66E-10	1.61E-10	8	0.97		0.156

Table A-11: Least-Squares Evaluation of Dosimetry in Watts Bar Unit 1 Surveillance Capsule Z (Single Capsule Holder, 34° Azimuth, Cycles 1 Through 7 Irradiation)

Reaction	Reaction Rate (rps/atom)			M/C	M/BE	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	4.19E-17	4.03E-17	4.07E-17	1.04	1.03	1.01
$^{54}Fe\ (n,p)\ ^{54}Mn$	4.29E-15	4.59E-15	4.37E-15	0.93	0.98	0.95
$^{58}Ni\ (n,p)\ ^{58}Co$	5.92E-15	6.48E-15	6.10E-15	0.91	0.97	0.94
$^{238}U\ (n,f)\ ^{137}Cs$	2.58E-14	2.55E-14	2.40E-14	1.02	1.08	0.94
$^{237}Np\ (n,f)\ ^{137}Cs$	2.40E-13	2.61E-13	2.42E-13	0.92	0.99	0.93
$^{59}Co\ (n,\gamma)\ ^{60}Co$	3.53E-12	3.82E-12	3.51E-12	0.92	1.01	0.92
$^{59}Co(Cd)\ (n,\gamma)\ ^{60}Co$	1.97E-12	2.57E-12	1.99E-12	0.77	0.99	0.78
Fast Reaction Threshold Foil Average				0.96	1.01	0.95
Standard deviation (%)				6.3	4.5	3.4

	C	BE	%Unc	BE/C		
Fluence Rate E>1.0 MeV (n/cm^2-s)	8.22E+10	7.68E+10	6	0.94		χ^2/DOF
DPA/s	1.63E-10	1.53E-10	8	0.94		0.229

Table A-12: Least-Squares Evaluation of Dosimetry in Watts Bar Unit 2 Surveillance Capsule U (Dual Capsule Holder, 34° Azimuth, Cycles 1 and 2 Irradiation)

Reaction	Reaction Rate (rps/atom)			M/C	M/BE	BE/C
	Measured (M)	Calculated (C)	Best-Estimate (BE)			
^{63}Cu (n,α) ^{60}Co	4.11E-17	4.61E-17	4.21E-17	0.89	0.98	0.91
^{54}Fe (n,p) ^{54}Mn	5.25E-15	5.39E-15	5.04E-15	0.97	1.04	0.94
^{58}Ni (n,p) ^{58}Co	7.25E-15	7.63E-15	7.10E-15	0.95	1.02	0.93
^{238}U (n,f) ^{137}Cs	2.15E-14	3.03E-14	2.77E-14	0.71	0.78	0.91
^{237}Np (n,f) ^{137}Cs	3.09E-13	3.11E-13	2.96E-13	0.99	1.04	0.95
^{59}Co (n,γ) ^{60}Co	4.77E-12	4.99E-12	4.74E-12	0.96	1.01	0.95
^{59}Co(Cd) (n,γ) ^{60}Co	2.71E-12	3.29E-12	2.73E-12	0.82	0.99	0.83
Fast Reaction Threshold Foil Average				0.90	0.97	0.93
Std deviation (%)				12.6	11.3	1.9

	C	BE	%Unc	BE/C	
Fluence Rate E>1.0 MeV (n/cm^2-s)	9.79E+10	8.94E+10	6	0.91	χ^2/DOF
DPA/s	1.94E-10	1.79E-10	8	0.92	0.910

Table A-13: Comparison of Measured and Calculated Threshold Reaction Rates for Unit 1

Capsule	M/C Ratio					Average M/C	Std dev
	^{63}Cu (n,α) ^{60}Co	^{54}Fe (n,p) ^{54}Mn	^{58}Ni (n,p) ^{58}Co	^{238}U (n,f) ^{137}Cs	^{237}Np (n,f) ^{137}Cs		
U	1.04	0.96	0.94	1.27	1.16	1.07	13.0%
W	1.25	1.14	1.10	0.91	1.21	1.12	11.8%
X	1.09	0.96	1.02	0.95	0.99	1.00	5.6%
Z	1.04	0.93	0.91	1.02	0.92	0.96	6.3%
Average	1.11	1.00	0.99	1.04	1.07	**1.04**	**11.0%**
Std dev	9.0%	9.6%	8.6%	15.6%	13.7%		

Table A-14: Comparison of Best-Estimate and Calculated Exposure Rates for Unit 1

Capsule	Average Fast Neutron (E > 1.0 MeV) Fluence Rate	Average Iron Displacement Rate
	BE/C	BE/C
U	1.06	1.08
W	1.06	1.07
X	0.97	0.97
Z	0.94	0.94
Average	**1.01**	**1.02**
Std dev	**6.1%**	**6.9%**

Table A-15: Comparison of Measured and Calculated Threshold Reaction Rates for Unit 2

Capsule	M/C Ratio					Average M/C	Std dev
	^{63}Cu (n,α) ^{60}Co	^{54}Fe (n,p) ^{54}Mn	^{58}Ni (n,p) ^{58}Co	^{238}U (n,f) ^{137}Cs	^{237}Np (n,f) ^{137}Cs		
U	0.89	0.97	0.95	0.71	0.99	0.90	12.6%
Average	0.89	0.97	0.95	0.71	0.99	**0.90**	**12.6%**
Std dev	--	--	--	--	--		

Table A-16: Comparison of Best-Estimate and Calculated Exposure Rates for Unit 2

Capsule	Average Fast Neutron (E > 1.0 MeV) Fluence Rate	Average Iron Displacement Rate
	BE/C	BE/C
U	0.91	0.92
Average	**0.91**	**0.92**
Std dev	**--**	**--**

A.2 REFERENCES

A-1 U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

A-2 A. Schmittroth, *FERRET Data Analysis Core*, HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

A-3 RSICC Data Library Collection DLC-178, *SNLRML Recommended Dosimetry Cross-Section Compendium*, July 1994.

A-4 ASTM Designation E944, 2019, “Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance,” ASTM International, West Conshohocken, PA, 2019, DOI: 10.1520/E0944-19, www.astm.org.

A-5 ASTM Designation E844, 2018, “Standard Guide for Sensor Set Design and Irradiation for Reactor Surveillance,” ASTM International, West Conshohocken, PA, 2018, DOI: 10.1520/E0944-08, www.astm.org.

APPENDIX B : CREDIBILITY EVALUATION OF THE WATTS BAR UNIT 1 SURVEILLANCE PROGRAM

Regulatory Guide 1.99, Revision 2 (Reference 3) describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Positions 2.1 and 2.2 of Regulatory Guide 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there have been four surveillance capsules removed and tested from the Watts Bar Unit 1 reactor vessel. To use these surveillance data sets, they must be shown to be credible. In accordance with the discussion of Regulatory Guide 1.99, Revision 2, there are five requirements that must be met for the surveillance data to be judged credible.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Watts Bar Unit 1 reactor vessel surveillance data and determine if that surveillance data is credible. Table B-1 reviews the five criteria in Regulatory Guide 1.99, Revision 2.

Table B-1
Regulatory Guide 1.99, Revision 2, Credibility Criteria

Criterion No.	Description
1	Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.
2	Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.
3	When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.
4	The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.
5	The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," (Reference 5) as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Watts Bar Unit 1 reactor vessel consists of the following beltline and extended beltline region materials:

- Upper Shell Forging 06, Heat # 411595
- Intermediate Shell Forging 05, Heat # 527536
- Lower Shell Forging 04, Heat # 528522
- Bottom Head Ring 03, Heat # 528170
- Upper Shell Forging to Intermediate Shell Forging Circumferential Weld Seam W06, Heat # 899680
- Intermediate Shell Forging 05 to Lower Shell Forging 04 Circumferential Weld Seam W05, Heat # 895075
- Lower Shell Forging 04 to Bottom Head Ring 03 Circumferential Weld Seam W04, Heat # 899680

The vessel forging material selected for inclusion in the surveillance program was Intermediate Shell Forging 05, which had the highest initial RT_{NDT} (Initial RT_{NDT} = 47°F per Table 3-1, and lowest initial USE (which was below the 75 ft-lbs limit from 10CFR50 Appendix G). Thus, it was selected as the surveillance base metal.

The weld material in the Watts Bar Unit 1 surveillance program was made of the same wire as the reactor vessel beltline circumferential welds, thus it was chosen as the surveillance weld material.

<u>Hence, Criterion 1 is met for the Watts Bar Unit 1 reactor vessel.</u>

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper shelf energy unambiguously.

Based on engineering judgment, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Watts Bar Unit 1 surveillance materials unambiguously.

<u>Hence, Criterion 2 is met for the Watts Bar Unit 1 reactor vessel.</u>

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for plates or forgings.

The Watts Bar Unit 1 surveillance weld will be evaluated for credibility. This weld is made from weld wire Heat # 895075. This weld metal is also contained in the Watts Bar Unit 2, Catawba Unit 1, and McGuire Unit 2 surveillance programs. Since the welds in question utilize data from other surveillance programs, the recommended NRC methods for determining credibility will be followed. The NRC methods for credibility determination were presented to industry at a meeting held by the NRC on February 12 and 13, 1998. At these meetings the NRC presented five cases. Of the five cases, Case 4 most closely represents the situation listed above for Watts Bar Unit 1 surveillance weld metal. Case 1 represents the surveillance forging material.

<u>Evaluation of the Watts Bar Unit 1 Data Only (Case 1)</u>

Table B-2 contains the calculation of chemistry factors for the Watts Bar Unit 1 surveillance forging and weld materials using only data from the WB1 surveillance program. These chemistry factors are calculated per Regulatory Guide 1.99, Revision 2, Position 2.1.

Table B-2
Surveillance Forging and Weld Material Interim Chemistry Factors using Watts Bar Unit 1 Data Only

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(a)]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell Forging 05 (Tangential)	U	0.46	0.784	98.30	77.05	0.614
	W	1.08	1.022	111.40	113.80	1.044
	X	1.75	1.154	94.70	109.27	1.331
	Z	2.40	1.236	144.50	178.60	1.528
Intermediate Shell Forging 05 (Axial)	U	0.46	0.784	28.70	22.49	0.614
	W	1.08	1.022	79.00	80.70	1.044
	X	1.75	1.154	115.90	133.73	1.331
	Z	2.40	1.236	104.90	129.65	1.528
				SUM:	845.28	9.034
	CF = Σ(FF * ΔRT_{NDT}) ÷ Σ(FF)^2 = (845.28) ÷ (9.034) = 93.6°F					
Surv. Weld Heat # 895075	U	0.46	0.784	0.00	0.00	0.614
	W	1.08	1.022	30.50	31.16	1.044
	X	1.75	1.154	25.80	29.77	1.331
	Z	2.40	1.236	13.90	17.18	1.528
				SUM:	78.11	4.517
	CF = Σ(FF * ΔRT_{NDT}) ÷ Σ(FF)^2 = (78.11) ÷ (4.517) = 17.3°F					

Notes:

(a) Information taken from Tables 4-1 and 4-2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

The scatter of ΔRT_{NDT} values for the surveillance capsule data about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table B-3.

Table B-3
Watts Bar Unit 1 Surveillance Capsule Data Scatter about the Best-Fit Line

Material	Capsule	CF[(a)] (Slope$_{best-fit}$) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(c)]	Measured ΔRT_{NDT}[(d)] (°F)	Predicted ΔRT_{NDT}[(e)] (°F)	Scatter ΔRT_{NDT}[(f)] (°F)	<17°F (Base Metal)
Intermediate Shell Forging 05 (Tangential)	U	93.6	0.46	0.784	98.3	73.4	24.9	No
	W	93.6	1.08	1.022	111.4	95.6	15.8	Yes
	X	93.6	1.75	1.154	94.7	108.0	13.3	Yes
	Z	93.6	2.40	1.236	144.5	115.7	28.8	No
Intermediate Shell Forging 05 (Axial)	V	93.6	0.46	0.784	28.7	73.4	44.7	No
	X	93.6	1.08	1.022	79.0	95.6	16.6	Yes
	U	93.6	1.75	1.154	115.9	108.0	7.9	Yes
	W	93.6	2.40	1.236	104.9	115.7	10.8	Yes
Surv. Weld Heat # 895075	U	17.3	0.46	0.784	0	13.6	13.6	Yes
	W	17.3	1.08	1.022	30.5	17.7	12.8	Yes
	X	17.3	1.75	1.154	25.8	20.0	5.8	Yes
	Z	17.3	2.40	1.236	13.9	21.4	7.5	Yes

Notes:

(a) CF calculated in Table B-2.
(b) Information taken from Table B-2.
(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(d) Measured ΔRT_{NDT} taken from Table B-2.
(e) Predicted ΔRT_{NDT} = CF x FF.
(f) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Measured ΔRT_{NDT}].

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 17°F for base metal and 28°F for weld metal. From a statistical point of view, +/- 1σ would be expected to encompass 68% of the data. Table B-3 indicates that five of the eight surveillance data points fall inside the +/- 1σ of 17°F scatter band for surveillance forging materials, which is 62.5% of the data (5/8 x 100). Therefore, the forging data is deemed "not credible" per the third criterion.

Table B-3 indicates that all 4 surveillance data points fall inside the +/- 1σ of 28°F scatter band for surveillance weld materials when considering only Watts Bar Unit 1 data. 100% of the data is bounded; therefore, the surveillance weld data is deemed "credible" per the third criterion when considering only Watts Bar Unit 1 data. Next, data from all sources is considered in order to evaluate the credibility of the weld metal using the NRC Case 4 guidelines.

Evaluation of Weld Data from All Sources (Case 4)

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. Data applicable to the Watts Bar Unit 1 surveillance weld material is also available from the Catawba Unit 1, Watts Bar Unit 2, and McGuire Unit 2 surveillance programs. Since data are from multiple sources, the data must be adjusted for chemical and irradiation environment differences. The chemistry adjustment ratios are shown in Section 5.

Table B-4 calculates the adjusted ΔRT_{NDT} for weld Heat #895075 in order to calculate the interim CF for the credibility evaluation.

Table B-4
Mean Chemical Composition and Temperature for Weld Heat # 895075

Material	Capsule	Cu[(a)] (Wt. %)	Ni[(a)] (Wt. %)	Chemistry Ratio[(b)]	Inlet Temp.[(c)] (°F)	Temp. Adjust.[(d)] (°F)	Measured ΔRT_{NDT}[(a)] (°F)	Adjusted ΔRT_{NDT}[(e)] (°F)
Watts Bar Unit 2 Surveillance Weld	U	0.033	0.70	1.2	559	-0.1	32.6	39.02
Catawba Unit 1 Surveillance Weld	Z	0.05	0.73	0.79	562	2.9	1.91	3.81
	Y	0.05	0.73	0.79	562	2.9	17.79	16.36
	V	0.05	0.73	0.79	562	2.9	26.5	23.24
Watts Bar Unit 1 Surveillance Weld	U	0.03	0.75	1.32	559	-0.1	0.0	-0.11
	W	0.03	0.75	1.32	559	-0.1	30.5	40.15
	X	0.03	0.75	1.32	559	-0.1	25.8	33.95
	Z	0.03	0.75	1.32	559	-0.1	13.9	18.24
McGuire Unit 2 Surveillance Weld	V	0.04	0.74	1.00	557	-2.1	38.51	36.43
	X	0.04	0.74	1.00	557	-2.1	35.93	33.85
	U	0.04	0.74	1.00	557	-2.1	23.81	21.73
	W	0.04	0.74	1.00	557	-2.1	43.76	41.68
MEAN	---	0.04	0.74	-	559.1	-	-	-

Notes:

(a) Information taken from Table 4-2.

(b) Chemistry Ratio = 54.0°F / (Surv Weld CF). 54.0°F is the Position 1.1 CF based on the average chemistry, Cu = 0.04% and Ni = 0.74%. Since this is equal to the CF for the Watts Bar Units 1 and 2 intermediate to lower circumferential weld, the Chemistry Ratios are taken from Section 5.

(c) Watts Bar Units 1 & 2 temperatures are determined by averaging (time-weighted) the inlet temperatures for all cycles prior to the capsule being removed.

(d) Temperature Adjustment = $T_{capsule} - T_{average}$.

(e) Adjusted ΔRT_{NDT} = ($\Delta RT_{NDT, Measured}$ + Temp. Adjustment) x (Chemistry Ratio)

Table B-5 calculates the interim CF for weld Heat # 895075 considering all available data adjusted to account for chemical and irradiation environment differences.

Table B-5
Heat # 895075 Interim Chemistry Factor Using All Available Surveillance Data

Material	Capsule	Capsule Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	Adjusted ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Watts Bar Unit 2 Surveillance Weld	U	0.614	0.863	39.02	33.69	0.745
Catawba Unit 1 Surveillance Weld	Z	0.286	0.658	3.81	2.51	0.433
	Y	1.29	1.071	16.36	17.52	1.147
	V	2.27	1.222	23.24	28.39	1.493
Watts Bar Unit 1 Surveillance Weld	U	0.46	0.784	-0.11	-0.09	0.614
	W	1.08	1.022	40.15	41.01	1.044
	X	1.75	1.154	33.95	39.17	1.331
	Z	2.40	1.236	18.24	22.54	1.528
McGuire Unit 2 Surveillance Weld	V	0.302	0.672	36.43	24.48	0.452
	X	1.38	1.089	33.85	36.87	1.187
	U	1.90	1.176	21.73	25.54	1.382
	W	2.82	1.276	41.68	53.17	1.628
				SUM:	324.82	12.983
	CF $_{Surv.\ Weld}$ = $\Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = (324.82) ÷ (12.983) = **25.0°F**					

Notes:

(a) Information taken from Table 4-2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) Adjusted ΔRT_{NDT} taken from Table B-4.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table B-6.

Table B-6
Heat # 895075 Surveillance Capsule Data Scatter about the Best-Fit Line Using All Available Surveillance Data

Material	Capsule	CF[(a)] (Slope$_{best-fit}$) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(c)]	Adjusted ΔRT_{NDT}[(d)] (°F)	Predicted ΔRT_{NDT}[(e)] (°F)	Scatter ΔRT_{NDT}[(f)] (°F)	<28°F (Weld)
Watts Bar Unit 2 Surveillance Weld	U	25.0	0.614	0.863	39.0	21.6	17.4	Yes
Catawba Unit 1 Surveillance Weld	Z	25.0	0.286	0.658	3.8	16.5	12.6	Yes
	Y	25.0	1.29	1.071	16.4	26.8	10.4	Yes
	V	25.0	2.27	1.222	23.2	30.6	7.3	Yes
Watts Bar Unit 1 Surveillance Weld	U	25.0	0.46	0.784	-0.1	19.6	19.7	Yes
	W	25.0	1.08	1.022	40.1	25.6	14.6	Yes
	X	25.0	1.75	1.154	33.9	28.9	5.1	Yes
	Z	25.0	2.4	1.236	18.2	30.9	12.7	Yes
McGuire Unit 2 Surveillance Weld	V	25.0	0.302	0.672	36.4	16.8	19.6	Yes
	X	25.0	1.38	1.089	33.8	27.3	6.6	Yes
	U	25.0	1.9	1.176	21.7	29.4	7.7	Yes
	W	25.0	2.82	1.276	41.7	31.9	9.8	Yes

Notes:

(a) CF calculated in Table B-5.
(b) Information taken from Table 4-2.
(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(d) Adjusted ΔRT_{NDT} taken from Table B-4.
(e) Predicted ΔRT_{NDT} = CF x FF.
(f) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Adjusted ΔRT_{NDT}].

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 17°F for base metal and 28°F for weld metal. From a statistical point of view, +/- 1σ would be expected to encompass 68% of the data. Table B-6 indicates that five of the eight surveillance data points fall inside the +/- 1σ of 17°F scatter band for surveillance forging materials, which is 62.5% of the data (5/8 x 100). Therefore, the forging data is deemed "not credible" per the third criterion.

Table 5-2 contains the calculation of chemistry factors for the Watts Bar Unit 1 intermediate shell forging 05 material contained in the surveillance program. These chemistry factors are calculated per Regulatory Guide 1.99, Revision 2, Position 2.1.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the neutron pad and the vessel wall and are positioned opposite the center of the core. The test capsules are located in brackets attached to the neutron pad. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Watts Bar Unit 1 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the Watts Bar Unit 1 surveillance program.

CONCLUSION:

Based on the preceding responses to the 5 criteria of Regulatory Guide 1.99, Revision 2, Section B, the Watts Bar Unit 1 surveillance weld data for Heat # 895075 are deemed credible and the data for the intermediate shell forging 05 are deemed non-credible.

APPENDIX C : CREDIBILITY EVALUATION OF THE WATTS BAR UNIT 2 SURVEILLANCE PROGRAM

Regulatory Guide 1.99, Revision 2 (Reference 3) describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Positions 2.1 and 2.2 of Regulatory Guide 1.99, Revision 2, describe the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Positions 2.1 and 2.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

Capsule U is the first surveillance capsule to be removed and tested from the Watts Bar Unit 2 reactor vessel. In accordance with Regulatory Guide 1.99, Revision 2, the credibility of the surveillance data will be judged based on five criteria. However, criterion 3 requires at least two data sets in order to determine the credibility. Since this is the first capsule withdrawn from Watts Bar Unit 2, this criterion cannot be applied to the surveillance forging. For this reason, the credibility of the surveillance forging cannot be determined due to the limited data available. The surveillance weld Heat # 895075 was utilized in the surveillance programs of sister-plants; therefore, criterion 3 can be applied to the surveillance weld with consideration of all available sister-plant data.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Watts Bar Unit 2 reactor vessel surveillance data and determine if that surveillance data is credible. Table C-1 reviews the five criteria in Regulatory Guide 1.99, Revision 2.

Table C-1
Regulatory Guide 1.99, Revision 2, Credibility Criteria

Criterion No.	Description
1	Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.
2	Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously.
3	When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.
4	The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.
5	The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," (Reference 5) as follows:

> *"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

The Watts Bar Unit 2 reactor vessel consists of the following beltline and extended beltline region materials:

- Upper Shell Forging 06, Heat # 411572
- Intermediate Shell Forging 05, Heat # 527828
- Lower Shell Forging 04, Heat # 528658
- Bottom Head Ring 03, Heat # 5329
- Upper Shell Forging to Intermediate Shell Forging Circumferential Weld Seam W06, Heat # 899680
- Intermediate Shell Forging 05 to Lower Shell Forging 04 Circumferential Weld Seam W05, Heat # 895075
- Lower Shell Forging 04 to Bottom Head Ring 03 Circumferential Weld Seam W04, Heat # 899680

The Watts Bar Unit 2 surveillance program utilizes tangential and axial test specimens from the Intermediate Shell Forging 05, Heat # 527828. The surveillance weldment is identical to the closing girth seam weldment between Forgings 04 and 05. The closing seam used weld wire Heat # 895075 with flux type Grau L.O. (LW320), lot P46, except for the 1-inch root pass at the I.D. of the vessel. This root pass used weld wire Heat # 899680 with type Grau L.O. (LW320) flux, lot P23, with an as-deposited copper and phosphorous content of 0.03 and 0.009, respectively. However, the surveillance weldment specimens were not removed from this root area.

Per WCAP-9455 (Reference 13), the Watts Bar Unit 2 surveillance program was developed to the requirements of ASTM E185-73. At the time of the surveillance program development, the Upper Shell Forging 06 and Bottom Head Ring 03 were not considered a "beltline" material. Of the other beltline forgings, Intermediate Shell Forging 05 was foreseen to be the most limiting forging. Intermediate Shell Forging 05 has the highest estimated initial and end of life RT_{NDT} and the lowest initial upper-shelf energy value of the Watts Bar Unit 2 beltline forgings. The chemistry values (Cu and Ni weight percent) for the beltline forgings are relatively consistent and no forging is clearly differentiated from the rest by its high copper or nickel content. Therefore, Intermediate Shell Forging 05 was appropriately selected as the base metal material for the surveillance program.

Intermediate Shell Forging 05 to Lower Shell Forging 04 Circumferential Weld Seam W05 was considered the only weld in the beltline region and therefore, was representative of all the beltline welds. Hence, the surveillance program weld was fabricated with the same weld wire heat (# 895075), the same flux type (Grau L.O., LW320), and the same flux lot (# P46) as the Intermediate to Lower Shell Forging Circumferential Weld Seam W05.

Therefore, the materials selected for use in the Watts Bar Unit 2 surveillance program were those judged to be most likely limiting with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed.

Hence, Criterion 1 is met for the Watts Bar Unit 2 reactor vessel.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30ft-lb temperature and upper shelf energy unambiguously.

Based on engineering judgment, the scatter in the data presented in these plots, as documented in Appendix D of WCAP-18518-NP (Reference 1), is small enough to permit the determination of the 30 ft-lb temperature and the upper-shelf energy of the Watts Bar Unit 2 surveillance materials unambiguously.

Hence, Criterion 2 is met for the Watts Bar Unit 2 reactor vessel.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

This criterion requires at least two data sets in order to determine the credibility. Since this is the first capsule withdrawn from Watts Bar Unit 2, this criterion cannot be applied to the surveillance forging. However, since the surveillance weld Heat # was utilized in the surveillance programs of sister-plants, this criterion can be applied to the surveillance weld.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 28°F for welds and less than 17°F for plates or forgings.

Following is the calculation of the best-fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2. In addition, the recommended NRC methods for determining credibility will be followed. The NRC methods were presented to the industry at a meeting held by the NRC on February 12 and 13, 1998. At this meeting the NRC presented five cases. Of the five cases, Case 4 ("Evaluation of Weld Data from All Sources") most closely represents the situation for the Watts Bar Unit 2 (Heat # 895075) material. Since only one capsule has been tested, an evaluation of the Watts Bar Unit 2 surveillance data alone cannot be completed.

Evaluation of Weld Data from All Sources (Case 4)

In accordance with the NRC Case 4 guidelines, the data from all sources should be adjusted to the mean chemical composition of all the data. Data applicable to the Watts Bar Unit 2 surveillance weld material is also available from the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance programs. Since data are from multiple sources, the data must be adjusted for chemical and irradiation environment differences.

Table C-2 calculates the adjusted ΔRT_{NDT} for weld Heat #895075 in order to calculate the interim CF for the credibility evaluation.

Table C-2
Mean Chemical Composition and Temperature for Weld Heat # 895075

Material	Capsule	Cu[a] (Wt. %)	Ni[a] (Wt. %)	Chemistry Ratio[b]	Inlet Temp.[c] (°F)	Temp. Adjust.[d] (°F)	Measured ΔRT_{NDT}[a] (°F)	Adjusted ΔRT_{NDT}[e] (°F)
Watts Bar Unit 2 Surveillance Weld	U	0.033	0.70	1.2	559	-0.1	32.6	39.02
Catawba Unit 1 Surveillance Weld	Z	0.05	0.73	0.79	562	2.9	1.91	3.81
	Y	0.05	0.73	0.79	562	2.9	17.79	16.36
	V	0.05	0.73	0.79	562	2.9	26.5	23.24
Watts Bar Unit 1 Surveillance Weld	U	0.03	0.75	1.32	559	-0.1	0.0	-0.11
	W	0.03	0.75	1.32	559	-0.1	30.5	40.15
	X	0.03	0.75	1.32	559	-0.1	25.8	33.95
	Z	0.03	0.75	1.32	559	-0.1	13.9	18.24
McGuire Unit 2 Surveillance Weld	V	0.04	0.74	1.00	557	-2.1	38.51	36.43
	X	0.04	0.74	1.00	557	-2.1	35.93	33.85
	U	0.04	0.74	1.00	557	-2.1	23.81	21.73
	W	0.04	0.74	1.00	557	-2.1	43.76	41.68
MEAN	**---**	0.04	0.74	-	559.1		-	-

Notes:

(a) Information taken from Table 4-2.

(b) Chemistry Ratio = 54.0°F / (Surv Weld CF). 54.0°F is the Position 1.1 CF based on the average chemistry, Cu = 0.04% and Ni = 0.74%. Since this is equal to the CF for the Watts Bar Units 1 and 2 intermediate to lower circumferential weld, the Chemistry Ratio are taken from Section 5.

(c) Watts Bar Units 1 & 2 temperatures are determined by averaging (time-weighted) the inlet temperatures for all cycles prior to the capsule being removed.

(d) Temperature Adjustment = $T_{capsule} - T_{average}$.

(e) Adjusted $\Delta RT_{NDT} = (\Delta RT_{NDT,\ Measured}$ + Temp. Adjustment) x (Chemistry Ratio)

Table C-3 calculates the interim CF for weld Heat # 895075 considering all available data adjusted to account for chemical and irradiation environment differences.

Table C-3
Heat # 895075 Interim Chemistry Factor Using All Available Surveillance Data

Material	Capsule	Capsule Fluence(a) (x 10^{19} n/cm^2, E > 1.0 MeV)	FF(b)	Adjusted ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Watts Bar Unit 2 Surveillance Weld	U	0.614	0.863	39.02	33.69	0.745
Catawba Unit 1 Surveillance Weld	Z	0.286	0.658	3.81	2.51	0.433
	Y	1.29	1.071	16.36	17.52	1.147
	V	2.27	1.222	23.24	28.39	1.493
Watts Bar Unit 1 Surveillance Weld	U	0.46	0.784	-0.11	-0.09	0.614
	W	1.08	1.022	40.15	41.01	1.044
	X	1.75	1.154	33.95	39.17	1.331
	Z	2.40	1.236	18.24	22.54	1.528
McGuire Unit 2 Surveillance Weld	V	0.302	0.672	36.43	24.48	0.452
	X	1.38	1.089	33.85	36.87	1.187
	U	1.90	1.176	21.73	25.54	1.382
	W	2.82	1.276	41.68	53.17	1.628
				SUM:	324.82	12.983
	$CF_{\text{Surv. Weld}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (324.82) \div (12.983) =$ **25.0°F**					

Notes:

(a) Information taken from Table 4-2.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.

(c) Adjusted ΔRT_{NDT} taken from Table C-2.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table C-4.

Table C-4
Heat # 895075 Surveillance Capsule Data Scatter about the Best-Fit Line Using All Available Surveillance Data

Material	Capsule	CF[(a)] ($Slope_{best-fit}$) (°F)	Capsule Fluence[(b)] (x 10^{19} n/cm², E > 1.0 MeV)	FF[(c)]	Adjusted ΔRT_{NDT}[(d)] (°F)	Predicted ΔRT_{NDT}[(e)] (°F)	Scatter ΔRT_{NDT}[(f)] (°F)	<28°F (Weld)
Watts Bar Unit 2 Surveillance Weld	U	25.0	0.614	0.863	39.0	21.6	17.4	Yes
Catawba Unit 1 Surveillance Weld	Z	25.0	0.286	0.658	3.8	16.5	12.6	Yes
	Y	25.0	1.29	1.071	16.4	26.8	10.4	Yes
	V	25.0	2.27	1.222	23.2	30.6	7.3	Yes
Watts Bar Unit 1 Surveillance Weld	U	25.0	0.46	0.784	-0.1	19.6	19.7	Yes
	W	25.0	1.08	1.022	40.1	25.6	14.6	Yes
	X	25.0	1.75	1.154	33.9	28.9	5.1	Yes
	Z	25.0	2.40	1.236	18.2	30.9	12.7	Yes
McGuire Unit 2 Surveillance Weld	V	25.0	0.302	0.672	36.4	16.8	19.6	Yes
	X	25.0	1.38	1.089	33.8	27.3	6.6	Yes
	U	25.0	1.90	1.176	21.7	29.4	7.7	Yes
	W	25.0	2.82	1.276	41.7	31.9	9.8	Yes

Notes:

(a) CF calculated in Table C-3.
(b) Information taken from Table 4-2.
(c) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$.
(d) Adjusted ΔRT_{NDT} taken from Table C-2.
(e) Predicted ΔRT_{NDT} = CF x FF.
(f) Scatter ΔRT_{NDT} = Absolute Value [Predicted ΔRT_{NDT} – Adjusted ΔRT_{NDT}].

The scatter of ΔRT_{NDT} values about the best-fit line, drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1, should be less than 17°F for base metal and 28°F for weld metal. From a statistical point of view, +/- 1σ would be expected to encompass 68% of the data. Table C-4 indicates that all twelve of the surveillance data points fall inside the +/- 1σ of 28°F scatter band for surveillance weld materials, which is 100% of the data (12/12 x 100). Therefore, the surveillance weld data is deemed "credible" per the third criterion.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the neutron shield pads and the vessel wall and are positioned opposite the center of the core. The test capsules are located in guide tubes attached to the neutron shielding pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criterion is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

The Watts Bar Unit 2 surveillance program does not contain correlation monitor material; therefore, this criterion is not applicable to the Watts Bar Unit 2 surveillance program.

CONCLUSION:

Based on the preceding responses to the 5 criteria of Regulatory Guide 1.99, Revision 2, Section B, the Watts Bar Unit 2 surveillance weld data for Heat # 895075 are deemed credible. Since only one capsule has been withdrawn and tested containing the Watts Bar Unit 2 surveillance forging material, insufficient data exists to determine the credibility of the surveillance forging material.