~~Proprietary Information Withhold Under 10 CFR 2.390(d)(1)~~
This letter is decontrolled when separated from Enclosure 3

1101 Market Street, Chattanooga, Tennessee 37402

CNL-17-144

December 20, 2017

10 CFR 50.90

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

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

Subject: **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)**

References:

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," dated September 23, 2002 (ML022540925)

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)

In accordance with the provisions of 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.

The proposed change revises WBN Unit 2 Technical Specification (TS) 4.2.1, "Fuel Assemblies," to add a limit on the number of Tritium Producing Burnable Absorber Rods (TPBARs) that can be irradiated. This change is analogous to the changes approved by NRC in WBN Unit 1 License Amendments 40 and 107 (References 1 and 2), which authorized the irradiation of 2,304 TPBAR and 1,792 TPBARs, respectively.

This license amendment also provides proposed changes related to the new criticality analyses performed for the spent fuel storage racks. The proposed changes revise WBN Units 1 and 2 TS 3.7.15, “Spent Fuel Assembly Storage,” to simplify the fuel storage limitations on fuel assemblies by eliminating the burnup–related criteria. The proposed changes add WBN Units 1 and 2 TS 3.7.18, “Fuel Storage Pool Boron Concentration,” to specify the minimum fuel storage pool boron concentration when fuel is stored in the pool. The proposed changes revise WBN Units 1 and 2 TS 3.9.9, “Spent Fuel Pool Boron Concentration,” to modify the minimum fuel storage pool boron concentration during refueling operations when fuel is stored in the pool. The proposed changes revise WBN Units 1 and 2 TS 4.3, “Fuel Storage,” to replace the storage limitations on fuel assembly burnup and storage with a single requirement to maintain a specified boron concentration in the spent fuel pool. The proposed changes add WBN Units 1 and 2 TS 5.7.2.21, “Spent Fuel Storage Rack Neutron Absorber Monitoring Program.”

Enclosure 1 to this letter provides a description of the proposed changes, technical evaluation of the proposed changes, regulatory evaluation, and a discussion of 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 the existing WBN Unit 1 TS Bases pages marked-up to show the proposed changes. Attachment 6 to Enclosure 1 provides the existing WBN Unit 2 TS Bases pages marked-up to show the proposed changes.

Enclosure 2 to this letter provides a copy of WCAP-18191-NP, Revision 0, “Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations.”

Enclosure 3 to this letter provides a copy of Holtec Report No: 2177876, “Licensing Report for the Criticality Safety Analysis of the Watts Bar Nuclear Plant Spent Fuel Pool.” This report contains information that Holtec International considers to be proprietary in nature and subsequently, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4), it is requested that such information be withheld from public disclosure. Enclosure 4 contains a non-proprietary version of the report as Holtec Report No: 2177950. Enclosure 5 provides the affidavit supporting this request for withholding.

TVA requests approval of the proposed License Amendment by May 31, 2019, to support a plan to irradiate TPBARs after the WBN Unit 2 Cycle 4 refueling outage in the fall of 2020 to support national security needs. The License Amendment will be implemented prior to startup from the outage where any number of TPBARs is inserted in the WBN Unit 2 reactor core.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

The WBN Plant Operations Review Committee and the TVA Nuclear Safety Review Board have reviewed this proposed change and determined that operation of WBN in accordance with the proposed change will not endanger the health and safety of the public.

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosures to the Tennessee Department of Environment and Conservation.

There is one new regulatory commitment associated with this submittal that is contained in Enclosure 6. Please address any questions regarding this request to Mr. Edward D. Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 20th day of December 2017.

Respectfully,

J. W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosures:

1. Evaluation of Proposed Change
2. WCAP-18191-NP, Revision 0, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations"
3. Proprietary Holtec Report No. 2177876, "Licensing Report for the Criticality Safety Analysis of the Watts Bar Nuclear Plant Spent Fuel Pool"
4. Holtec Report No. 2177950, "Non-Proprietary Licensing Report for the Criticality Safety Analysis of the Watts Bar Nuclear Plant Spent Fuel Pool"
5. Holtec International Application for Withholding Proprietary Information From Public Disclosure
6. List of Commitments

cc (Enclosures):

NRC Regional Administrator - Region II
NRC 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

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

EVALUATION OF PROPOSED CHANGE

Subject: Application to Revise Technical Specification 4.2.1, "Fuel Assemblies" (WBN-TS-17-028)

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 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.

The proposed change revises WBN Unit 2 Technical Specification (TS) 4.2.1, "Fuel Assemblies," to add a limit on the number of Tritium Producing Burnable Absorber Rods (TPBARs) that can be irradiated. This change is analogous to the changes implemented in WBN Unit 1 License Amendments 40 and 107 (References 1 and 2), which authorized the irradiation of 2,304 and 1,792 TPBARs, respectively, in WBN Unit 1.

The proposed change is required to support a planned increase of TPBAR inventory in the WBN Unit 2 Cycle 4 refueling outage in the fall of 2020 to support national security needs.

This license amendment also provides proposed changes related to the new criticality analyses performed for the spent fuel storage racks. The proposed changes revise WBN Units 1 and 2 TS 3.7.15, “Spent Fuel Assembly Storage,” to simplify the fuel storage limitations on fuel assemblies by eliminating the burnup–related criteria. The proposed changes add WBN Units 1 and 2 TS 3.7.18, “Fuel Storage Pool Boron Concentration,” to specify the minimum fuel storage pool boron concentration when fuel is stored in the pool. The proposed changes revise WBN Units 1 and 2 TS 3.9.9, “Spent Fuel Pool Boron Concentration,” to modify the minimum fuel storage pool boron concentration during refueling operations when fuel is stored in the pool. The proposed changes revise WBN Units 1 and 2 TS 4.3, “Fuel Storage,” to replace the storage limitations on fuel assembly burnup and storage with a single requirement to maintain a specified boron concentration in the spent fuel pool. The proposed changes add WBN Units 1 and 2 TS 5.7.2.21, “Spent Fuel Storage Rack Neutron Absorber Monitoring Program.”

2.0 DETAILED DESCRIPTION

This license amendment request (LAR) revises WBN Unit 2 TS 4.2.1, "Fuel Assemblies," to add a limit of 1,792 on the number of Tritium Producing Burnable Absorber Rods (TPBARs) that can be irradiated.

This license amendment also provides proposed changes related to the new criticality analyses performed for the spent fuel storage racks. The proposed changes revise WBN Units 1 and 2 TS 3.7.15, “Spent Fuel Assembly Storage,” to simplify the fuel storage limitations on fuel assemblies by eliminating the burnup–related criteria. The proposed changes add WBN Units 1 and 2 TS 3.7.18, “Fuel Storage Pool Boron Concentration,” to specify the minimum fuel storage pool boron concentration when fuel is stored in the pool. The proposed changes revise WBN Units 1 and 2 TS 3.9.9, “Spent Fuel Pool Boron Concentration,” to modify the minimum fuel storage pool boron concentration during refueling operations when fuel is stored in the pool. The proposed changes revise WBN Units 1 and 2 TS 4.3, “Fuel Storage,” to replace the storage limitations on fuel assembly burnup and storage location with a single requirement to maintain a specified boron concentration in the

spent fuel pool. The proposed changes add WBN Units 1 and 2 TS 5.7.2.21, “Spent Fuel Storage Rack Neutron Absorber Monitoring Program.” The proposed technical specification changes ensure continued compliance with 10 CFR 50.36, “Technical specifications.”

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 the existing WBN Unit 1 TS Bases pages marked-up to show the proposed changes and is provided for information only. Attachment 6 to Enclosure 1 provides the existing WBN Unit 2 TS Bases pages marked-up to show the proposed changes. The changes to the TS Bases are controlled by WBN Units 1 and 2 TS 5.6, "Technical Specification (TS) Bases Control Program."

Approval of this LAR will authorize the irradiation of up to a maximum of 1,792 TPBARs in WBN Unit 2 and replace the fuel storage limitations for WBN Units 1 and 2 on fuel assembly initial enrichment, burnup, and storage location limitations with a single requirement to maintain a specified boron concentration in the spent fuel pool.

3.0 BACKGROUND

The Department of Energy (DOE) and TVA have agreed to cooperate in a program to produce tritium for the National Security Stockpile by irradiating TPBARs at WBN Unit 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 matter 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 will permeate 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.

The first TPBARs irradiated in WBN Unit 1 were in four lead test assemblies (LTAs), containing a total of 32 TPBARs during WBN Unit 1 Cycle 2. NRC approval of the LTAs was documented in WBN Unit 1 License Amendment 8 (Reference 3).

WBN Unit 1 License Amendment 40 (Reference 1) approved the irradiation of up to 2,304 TPBARs in WBN Unit 1. The exact number of TPBARs to be irradiated is identified in the safety evaluation performed for each reload core and noted in the Core Operating Limits Report (COLR) for each fuel cycle.

Based on issues related to the Reactor Coolant System (RCS) boron concentration, the TVA letter of August 18, 2003, revised the WBN Unit 1 LAR dated May 30, 2003 and limited the maximum number of TPBARs to be irradiated to 240 in WBN Unit 1 Cycle 6. This restriction was approved by the NRC in WBN Unit 1 License Amendment 48 (Reference 4). Design changes made to the TPBARs scheduled for Cycle 9 supported a request to increase the maximum number of TPBARs to be irradiated to 400. This increase was approved with the issuance of WBN Unit 1 License Amendment 67 (Reference 5). The number of TPBARs irradiated in Cycle 9 was 368. TVA reduced the number of TPBARs irradiated in Cycle 10 to 240 after discovering that the design changes deployed in Cycle 9 did not significantly reduce tritium permeation. WBN Unit 1 License Amendment 77 (Reference 6) was issued allowing TVA to increase the maximum number of TPBARs to be irradiated to 704. Because analysis showed consistent tritium releases due to TPBAR permeation in Cycles 6 through 9, the number of TPBARs irradiated in Cycles 11 and 12 were increased to 544. Cycles 13 and 14 irradiated 704 TPBARs. Design changes made to WBN Unit 1 supported a request to increase the maximum number of TPBARs to be irradiated to 1,792. This increase was approved with the issuance of WBN Unit 1 License Amendment 107 (Reference 2).

As described in this LAR, TVA is requesting approval to irradiate up to 1,792 TPBARs in WBN Unit 2. The number of TPBARs to be irradiated in any given operating cycle will be evaluated in the reload safety evaluation and documented in the COLR. The number of TPBARs will not exceed 1,792.

4.0 TECHNICAL EVALUATION

This proposed change is justified based on extensive analysis, testing, and evaluation of the TPBARs as reported previously by the DOE. DOE has previously submitted a classified/proprietary version of the Tritium Production Core (TPC) Topical Report NDP-98-153, Revision 1, and an unclassified/non-proprietary version, NDP-98-181, Revision 1 (Reference 7) for NRC review. NRC reviewed these TPC Topical Reports and issued NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report on the Tritium Production Core" (Reference 8). TVA used both versions of the TPC Topical Report and the NRC Safety Evaluation Report (SER) in the preparation of this LAR and has completed the appropriate plant-specific evaluations for the 17 interface items listed in NUREG-1672, Section 5.1. Copies of the classified documents are available for NRC review at the Pacific Northwest National Laboratory (PNNL) offices. The 17 plant-specific interface items from NUREG-1672 are addressed for WBN Unit 2 in Section 4.1.

Section 4.2 of this enclosure provides the results of the post-LOCA subcriticality analysis for a representative core design with 1,792 TPBARs using the current refueling water storage tank (RWST) and cold leg accumulator boron concentrations, which are not changed by this request. This analysis considers the whole range of break sizes up to and including a double-ended guillotine rupture of the main coolant loop piping. Additionally, the analysis takes credit for the isolation of the potential unborated dilution source that would have entered the containment at a maximum rate of 40 gallons per minute (gpm), and a conservative lithium leaching assumption for TPBARs assumed to fail. The effect of TPBARs on post-LOCA subcriticality was

evaluated using the methodology employed for WBN Unit 1 License Amendment 107 (Reference 2).

4.1 Watts Bar Plant Unit 2 Specific Interface Issues

During the NRC's review of the DOE TPC Topical Report, the NRC determined that there are certain plant specific interface issues for which the licensee must submit additional information and/or analyses. This information would be used to support a plant specific license amendment to the facility's operating license for authorization to operate a tritium production core. Each specific interface issue has been evaluated for WBN Unit 2 and is discussed below.

The following is a listing of the NUREG-1672 interface items along with section number where these items are addressed in this enclosure:

1. Handling of TPBARs (4.1.1)
2. Procurement and Fabrication Issues (4.1.2)
3. Compliance with DNB Criterion (4.1.3)
4. Reactor Vessel Integrity Analysis (4.1.4)
5. Control Room Habitability Systems (4.1.5)
6. Specific Assessment of Hydrogen Source and Timing or Recombiner Operation (4.1.6)
7. Light-Load Handling System (4.1.7)
8. Station Service Water System (4.1.8)
9. Ultimate Heat Sink (4.1.9)
10. New and Spent Fuel Storage (4.1.10)
11. Spent Fuel Pool Cooling and Cleanup System (4.1.11)
12. Component Cooling Water System (4.1.12)
13. Demineralized Water Makeup System (4.1.13)
14. Liquid Waste Management System (4.1.14)
15. Process and Effluent Radiological Monitoring and Sampling System (4.1.15)
16. Use of LOCTAJR Code for LOCA analyses (4.1.16)
17. ATWS Analysis (4.1.17)

Each of the section 4.1.1 through 4.1.17 contains one or more quote from NUREG-1672 followed by a discussion of the plant-specific evaluation of the interface item.

4.1.1 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 TPBAR Interface Issue 1 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, and 10).

TPBAR assemblies will be inserted into new fuel assemblies or specially-designed transport containers prior to shipment to WBN. After inspection, plant operators can place the fuel assemblies containing TPBARs in the new fuel storage area or can place them into the spent fuel pool (SFP). TPBARs shipped in specially-designed transport containers will be removed and placed in fuel assemblies within the SFP. TVA will use the same methods, procedures, and equipment to place fuel assemblies containing TPBARs into the core as it does for fuel assemblies that do not contain TPBARs. Material accountability for TPBAR assemblies is administratively controlled.

After irradiating the TPBARs in the reactor, plant operators will offload the entire core to the spent fuel racks in the SFP. After the core is unloaded to the SFP during refueling, the irradiated TPBAR assemblies will be removed from the fuel and transferred to available storage locations within the spent fuel pool using the burnable poison rod assembly tool.

After irradiation, TPBARs are removed from the fuel assemblies and consolidated into canisters. For the consolidation process, TVA will use a specially designed TPBAR consolidation fixture (TCF), which will be installed in the cask loading pit (see Figures 4.1-1 and 4.1-2). Plant operators will remove the irradiated TPBAR assemblies from the spent fuel assemblies, disassemble all the irradiated TPBARs for consolidation, and place them into consolidation canisters. Operators will return the loaded consolidation canisters to the spent fuel racks, where they will remain until removed from the site. The loaded canisters will be transported to the Tritium Extraction Facility at the Savannah River Site in South Carolina, using a commercially-available cask licensed specifically to transport loaded canisters.

After the core is unloaded to the spent fuel pool during refueling, the irradiated TPBAR assemblies are removed from the fuel assemblies and transferred to available storage locations within the spent fuel pool using the burnable poison rod assembly tool. TVA reuses fuel assemblies that were irradiated for only one fuel

cycle (i.e., once burned). For once-burned fuel assemblies containing TPBARs, plant operators will remove the TPBAR assemblies from the fuel assemblies and temporarily store the TPBAR assemblies in old spent fuel assemblies or TPBAR assembly holding fixtures in the SFP. TVA would then reinsert the once-burned fuel assemblies into the core during the refueling outage.

Burnable poison rods are similar in design to TPBARs, and weigh about the same (i.e., assembly weighs 58 versus 65 pounds, respectively), which allows TPBAR movement using the burnable poison rod assembly handling tool. The hoisting cable on the burnable poison rod assembly tool has a breaking strength of 1,700 pounds, and TVA tested the tool hoist to 900 pounds. Thus, the burnable poison rod assembly tool is capable of safely handling the weight of TPBAR assemblies.

The weight of a fuel assembly with 24 TPBARs and its hold down assembly (62 additional pounds for TPBARs) is less than an assembly with a Rod Control Cluster (94 additional pounds). Therefore, the weight is bounded by the current assumed weight of an assembly for purposes of analyzing fuel handling and storage facilities. The TPBAR equipped fuel assembly has the same external configuration to interface with the fuel handling/storage equipment.

Approximately 30 days after refueling is complete, TPBAR consolidation begins. The time to start consolidating the TPBARs is not limited by any safety issues (e.g., decay heat), but rather is based on scheduling. The 30-day estimate corresponds to when TVA expects to be finished with all outage-related activities, and can begin consolidation efforts.

The time to consolidate two units of maximum load TPBARs (i.e., 3,584 TPBARs) into canisters stored in the SFP and ship are approximated as follows:

#	Activity	Weeks
1	Consolidation mobilization (e.g., fixture assembly, training, refuel floor preparations)	2
2	Consolidation of two units of maximum TPBARS (i.e., 3,584 TPBARs) @ one canister (300) per week	12
3	Consolidation fixture disassembly, storage and clean-up	1
4	Tritium cask shipment (assuming one shipment for one canister per week)	12
5	Post irradiation examination (PIE) Shipment	1
6	Waste shipments (e.g., empty TPBAR baseplates, miscellaneous)	2
	Total	30

The following table represents an approximate 18-month cycle (78 Weeks) for activities involving the SFP:

#	Activity	Weeks
1	U1 New Fuel Receipt	6
2	U1 Refueling Outage	6
3	U2 New Fuel Receipt	6
4	U2 Refueling Outage	6
5	Dry Cask Campaign (eight casks @ one week per cask + two weeks preparation/clean-up)	10
6	Consolidation of two units of maximum TPBARS	30
7	Remaining duration for crane maintenance, miscellaneous, contingency, etc.	14
	Total	78

The weight of the completed TCF is 9,000 pounds (i.e., 3,900 Upper TCF plus 5,100 Lower TCF). The TCF has been successfully utilized to consolidate TPBARs since 2005.

The TCF is quality related in accordance with TVA's NRC-accepted Quality Assurance (QA) Program (TVA-NQA-PLN89-A). The TCF is normally stored in the cask lay-down area when not in use. The TCF includes video monitoring, lighting, and tools designed to remove TPBARs from the assembly baseplate.

The canisters that will receive the irradiated TPBARs are transferred into the SFP and placed into the TCF when required. A TPBAR assembly is then withdrawn from its storage location and moved from the SFP to the TCF using the TPBAR assembly handling tool suspended from the SFP Bridge Crane. A TPBAR removal tool is then utilized by personnel on the platform to detach individual TPBARs from the baseplate.

The TPBAR slides along frame guides, through a funnel and into a roller brake to limit its velocity, and then into a consolidation canister. The funnel, roller brake assembly, and canister are angled at approximately 15 degrees to enable the TPBARs to stack efficiently into the canister to maximize the loading. A canister exciter tool is also utilized to enhance uniform TPBAR stacking into the canister. Up to 300 TPBARs are deposited into a consolidation canister (see Figure 4.1-3).

Each TPBAR is equipped with a top end plug that is threaded into the baseplate. Extending above the baseplate is a hexagonal region on the top end plug to which the crimp sleeve is secured to prevent inadvertent TPBAR loosening during reactor operations. The hex stud facilitates installation and removal and serves as the feature to which the sleeve is crimped. The top end plug threads are left-hand such that when the TPBAR is removed, left hand torque is used by the tool operated from the TCF above the baseplate.

During the consolidating of TPBARs, the TPBARs are unscrewed from the baseplate and removed. A hex socket tool is used to remove the TPBAR using the hex stud. The hex tool is mounted to a pole for manual disassembly. The hex tool is lowered into position from the consolidation platform. Once the tool is engaged on the hex stud, sufficient torque is applied until the resistance of the crimp is exceeded. The TPBAR is turned until it falls from baseplate and drops into the canister.

Activities take place underwater at a safe shielding water depth. The TPBAR handling and consolidation equipment is designed and configured such that minimum water shielding in the SFP and Cask Loading Pit is maintained to keep dose rates as low as reasonably achievable (ALARA). Tool design/features prevent inadvertently raising the TPBAR assemblies, loaded canisters, or post consolidation baseplates above safe shielding depths. The spent fuel bridge crane is used to handle the canisters, and a review of this operation is included with the review of the light-load handling system (see Section 4.1.7). Personnel will work on a platform 24 inches above SFP normal water level over the deep end of the Cask Loading Pit. The platform is designed to accommodate lead shielding, if required, for personnel protection.

TVA has in place the following contingency plans in the event a rod becomes stuck in the fixture before placing it in the canister. If the threaded engagement of the TPBAR to the baseplate becomes galled or is incapable of being removed by the hex tool, a backup method of TPBAR removal is required. To enable TPBAR removal in this case, a small hydraulic cutter would be used to sever the upper end plug shank (hex stud) of the TPBAR from the baseplate. This method would require that all TPBARs that could be de-torqued be removed by the conventional method. Then, the cutter would be applied to the TPBAR just below the baseplate. The cutter would sever the upper end plug shank (hex stud) of the TPBAR at the smallest diameter. Severing the upper end plug shank (hex stud) in this region would not affect the integrity of the rod itself. This method has been successfully tested and has also been utilized in other SFP applications.

After TPBARs have been removed from a baseplate, the baseplate and any attached thimble plugs will be removed from the fixture, and placed in storage. The process is repeated until the canister is filled with up to 300 TPBARs.

Several features are available if a TPBAR fails to drop into the canister:

- The roller-brake gear-motor can be reversed to raise the TPBAR up into the "funnel" section of the TCF. The torque, speed, and direction of the roller-brake motor are programmable.
- The compression of the roller-brake rollers on a TPBAR can be released by pulling up on the roller-brake assembly pivot arm with a shepherd's hook, allowing the TPBAR to drop or be pulled out.
- A TPBAR can be manipulated manually with an underwater clamping device and placed into a canister.
- The canister exciter tool can be utilized to "settle" TPBARs and allow a partially inserted TPBAR to drop completely.

Disposal or storage of the baseplates and thimble plugs will be in accordance with accepted radwaste program following consolidation.

The loaded canister is removed and transported to a designated storage position in the SFP storage rack using the canister handling tool suspended from the SFP Bridge Crane. Damage to TPBARs during canister handling is precluded by a robust canister design, a canister top insert to protect the TPBARs top 12 inches, and a handling lanyard to prevent the canister from tipping past horizontal and spilling TBPARs. The next empty consolidation canister is placed into the consolidation fixture and the process is repeated until all TPBARs irradiated during the fuel cycle have been consolidated. The consolidation fixture is then removed from the cask load pit, disassembled, and stored in the cask lay-down area.

The TCF is designed to remain in place in both its use and storage positions during all credible postulated accidents and natural phenomena, precluding damage to other safety related SSCs. A seismic category I(L) design precludes damage to the SFP liner in the cask loading pit and consolidated TPBARs while in the fixture.

Subsequently, a shipping cask is placed into the cask loading pit. The cask is handled by the Auxiliary Building crane in accordance with NUREG-0612 program requirements. The canisters are transferred into the submerged cask. The cask is removed from the cask loading pit, drained of water and decontaminated, packaged and certified for shipment. This shipping process is repeated until all TPBARs irradiated during the past operating cycle have been shipped. The consolidation process is based upon accepted industry practices. The evolutions are performed with sufficient shielding to minimize exposure, and specialized tooling has been developed to streamline the process. The consequences of a breached TPBAR because of mishandling in the SFP are addressed in Interface Issue 5.

The spent fuel bridge crane is used to handle the canisters, and a review of this operation is included with the review of the light-load handling system (see Interface Issue 7).

Operators will use the 125-ton auxiliary building crane to handle the TCF. The crane has both a 10-ton hoist and a single-failure-proof 125-ton main hoist. The TCF is normally handled by the auxiliary 10-ton hoist. The TCF weighs less than one-half of the hook capacity of the 10-ton hoist. Therefore, the auxiliary building crane has adequate capacity to handle the TCF to meet NUREG-0612 requirements.

The maximum TPBAR Cask weight is 52,000 pounds. It is handled by the main 125-ton single-failure-proof hoist and therefore meets NUREG-0612 requirements.

The consolidation fixture will normally be stored in the cask lay-down area when not in use. NUREG-0612, Section 5.1.1 discusses general guidelines to minimize the potential impact of heavy loads on spent fuel stored in the pool. TVA takes precautions when handling the fixture and cask due to the proximity to the fuel. The special precautions are that the lifting devices, slings, and crane will meet equivalent single failure proof criteria, mainly by doubling the normal safety factors. The cask laydown section of the SFP area is separated from the irradiated fuel storage section by a wall, providing several feet separation. It is not deemed necessary nor

warranted to relocate resident discharged fuel while storing and/or handling of the consolidation fixture in the cask laydown area.

Accordingly, the handling of the Consolidation Platform is normally performed with the 10-Ton hoist of the 125/10-Ton Auxiliary Building Crane and is considered equivalent single-failure-proof for this lift due to the following considerations:

- The Consolidation Platform (or platform sections) weighs less than one-half of the hook capacity of the auxiliary 10-ton hoist. (Note: The platform is handled as a single unit and in two sections during assembly). Along with other design and administrative features, this crane is equivalent single failure-proof consistent with the requirements of NUREG 0612 and NUREG-0554 for this lift.
- The lifting devices are designed to the requirements of ANSI N14.6 for Critical Loads with increased safety factors and load test weights, in addition to the design, fabrication, inspection, and testing contained in Sections 1 through 6 of ANSI N14.6, thereby rendering it equivalent single-failure-proof.

The Auxiliary Building Crane auxiliary hoist, while possessing many of the attributes required for a single-failure-proof crane, has not been evaluated to comply with NUREG-0554. TVA complies with all seven points of NUREG-0612 Section 5.1.1, as is delineated in WBN's response to Generic Letter 81-07 (Reference 11) and accepted by NRC in Section 9.1.4 of NUREG-0847, Supplement 13 (Reference 12).

For TPBAR associated heavy load lifts, the auxiliary hoist of the Auxiliary Building 125/10-Ton crane meets NUREG-0612, Section 5.1.2, option number one by complying with Section 5.1.6, specifically Appendix C of NUREG-0612 for existing cranes, except for the load hang-up protection and associated testing. Lifts are controlled by site procedures, that require pre-lift briefings, trained operators, etc. Therefore lifts will be adequately monitored to help preclude load hang-ups. Certain items of compliance are contingent upon the fact that the loads for TPBAR-associated lifts are less than half of the hook capacity, thereby yielding increased safety factors for the structural/wear- related requirements.

Lifting devices and interfacing lift points for TPBAR-related heavy loads are required to meet the requirements of NUREG-0612 Section 5.1.6, either by redundant paths or increased safety factors, as delineated in ANSI/ASME N14.6.

The use of the Auxiliary Building Crane was previously evaluated in Section 2.1 of the safety evaluation for the WBN Unit 1 license amendment to irradiate 2,304 TPBARs (Reference 1) and in Section 9.1.4 of NUREG-0847, Supplement 22 (Reference 13).

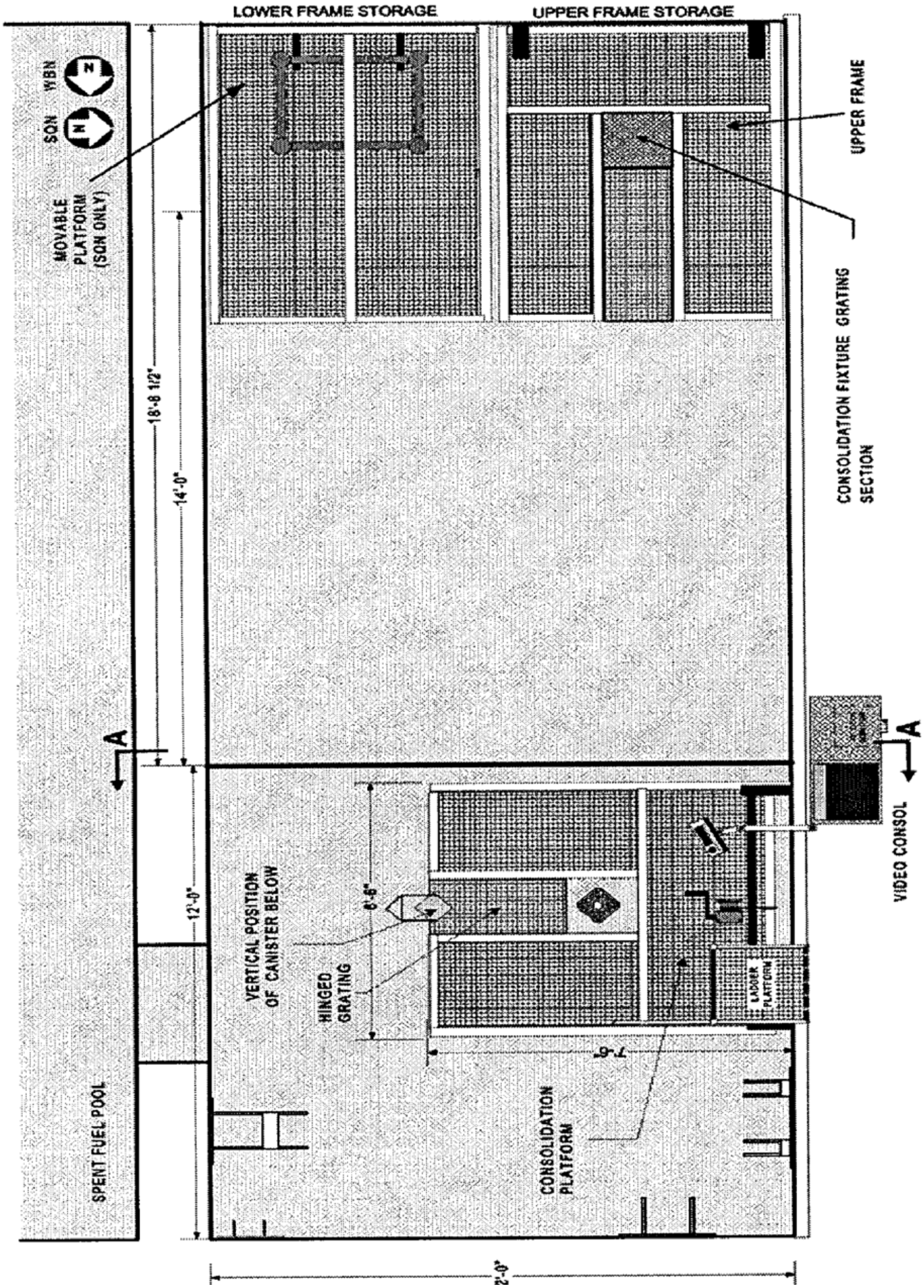

Figure 4.1-1: Consolidation Plan View

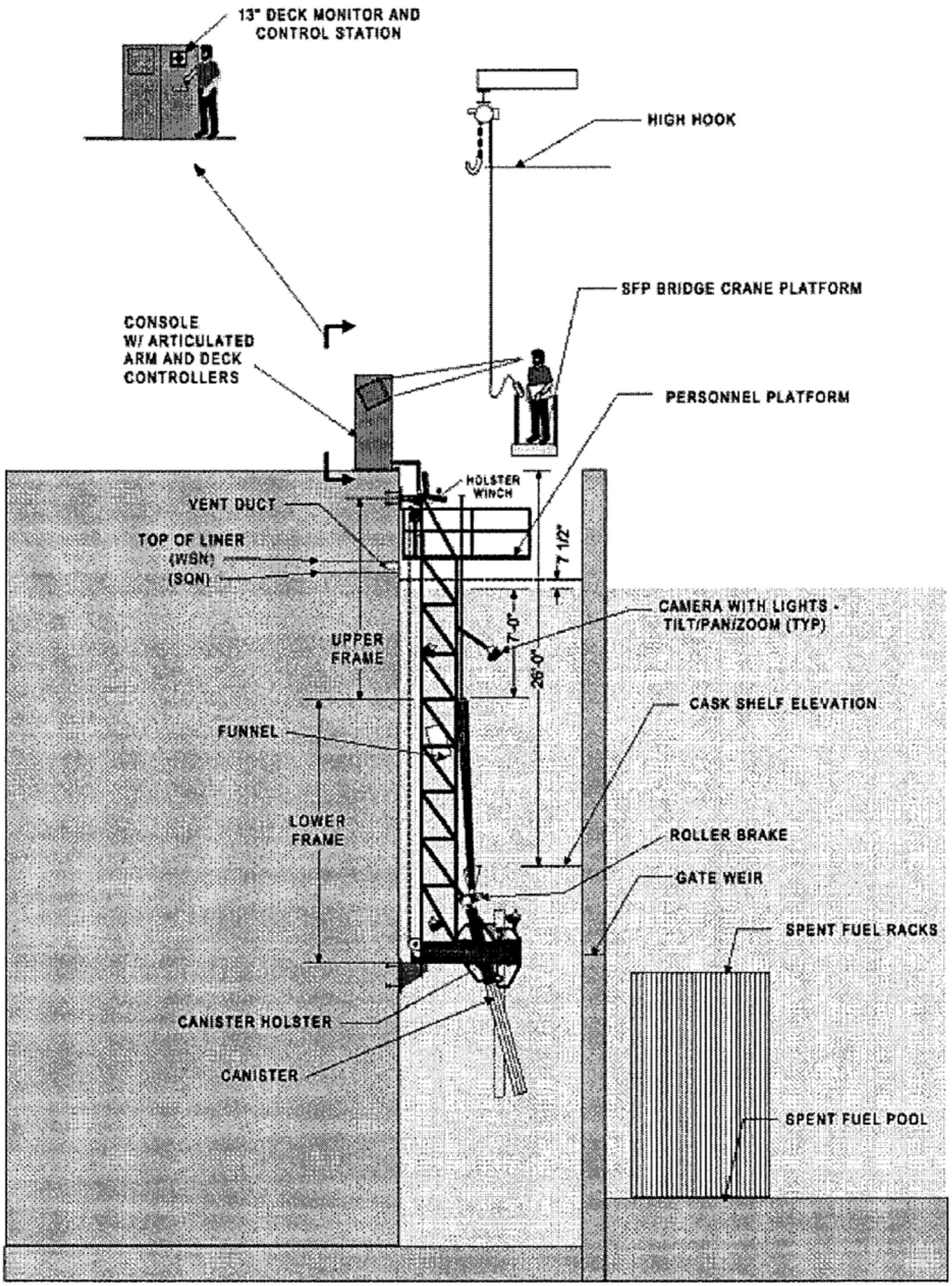

Figure 4.1-2: Consolidation Layout

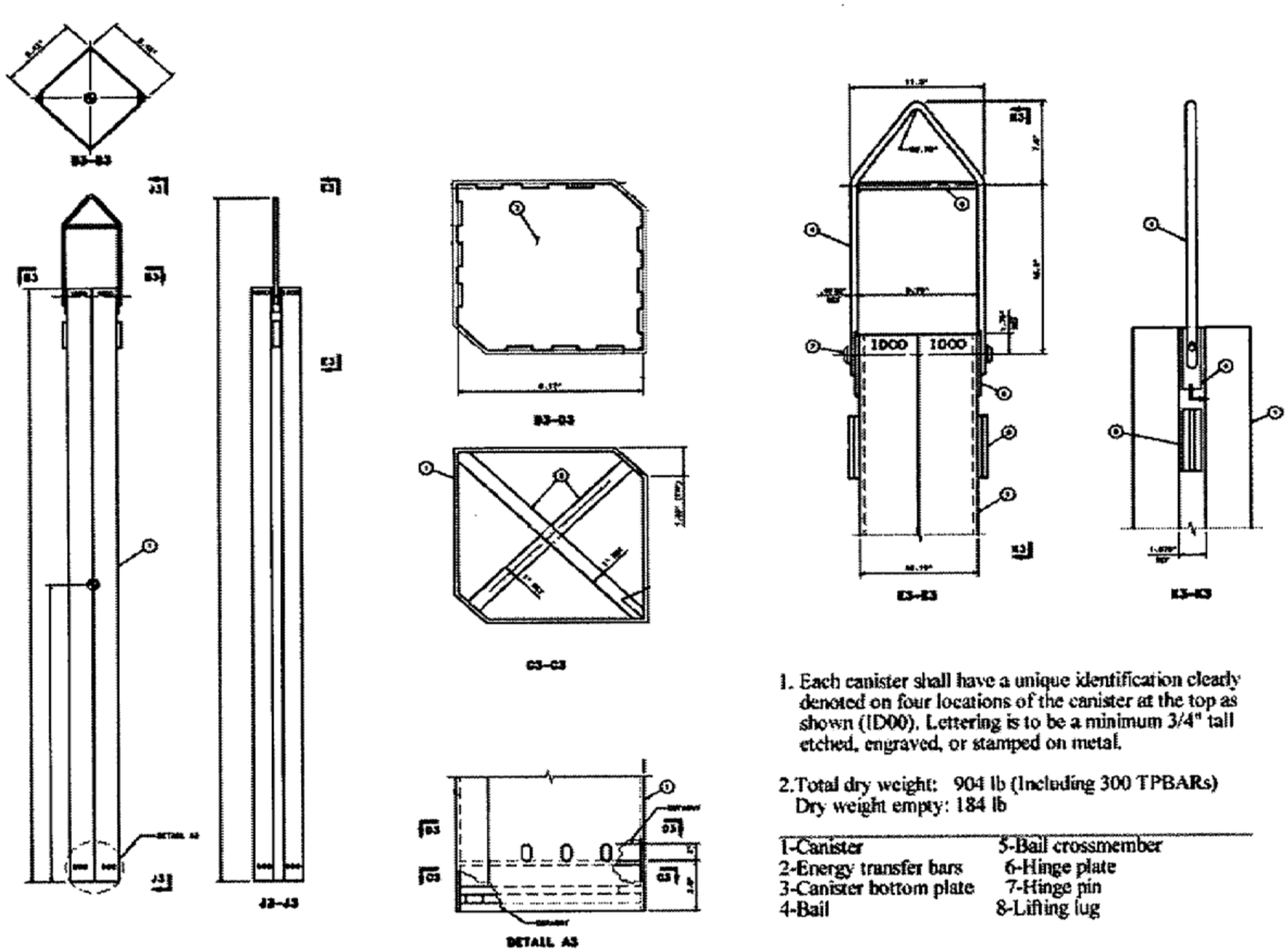

Figure 4.1-3: Consolidation Canister

4.1.2 TPBAR Interface Issue 2: Procurement and Fabrication Issues

NUREG-1672, Section 1.3 "Independent of its review of the DOE TPC topical report, the staff is conducting vendor-related activities with respect to quality assurance (QA) plans and fabrication inspections in order to determine compliance with the requirements of Appendix B to 10 CFR Part 50 and with 10 CFR Part 21. 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 2.17.1 "DOE has not yet selected the supplier for the fabrication of the production core TPBARs, and NRC review and inspection of supplier/vendor QA programs is not within the scope of this evaluation. Procurement processes performed on behalf of DOE for production core TPBAR components by contractors other than the production core TPBAR fabricator will also be subject to NRC review and inspection. 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 2 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, and 14).

TPBARs are supplied to TVA as “Government Furnished Property” per Interagency Agreement No. DE-AI02-00DP00315 between TVA and the National Nuclear Security Agency (NNSA). TVA has no direct procurement document with any of the material, service, or component suppliers of TPBARs. Because TPBARs are classified as safety-related components and procured by NNSA outside of the TVA procurement system, a unique protocol has been established to implement the TVA quality assurance (QA) requirements that are applicable to TPBARs. TVA has an Interagency Agreement with the NNSA that requires NNSA to flow down TVA requirements to suppliers and requires the NNSA direct suppliers to be on the TVA acceptable supplier list (ASL).

The main TVA document establishing these QA requirements is TVA-TPPR-99-01, Revision 4, *Tritium Production Program Requirements: Technical, Functional, & Quality Requirements for TPBARs*, which defines the technical, functional, and quality requirements associated with design, analysis, materials, fabrication, and delivery of TPBARs that will be inserted into host fuel assemblies for irradiation in a TVA nuclear reactor. TVA-TPPR-99-01 requires NNSA to flow down TVA QA requirements to their respective suppliers. It also requires direct suppliers to TVA maintain a TVA-accepted QA program. Other requirements included in TVA-TPPR-99-01 are TVA acceptance of deviation resolution, interface controls, reporting requirements, document submittal requirements, and TPBAR functional requirements.

Activities associated with TPBAR design, material and services procurements, fabrication, and delivery are performed under the auspices of TVA’s NRC-Accepted QA program. TVA-TPPR-99-01 states that although NNSA manages the Tritium Production Program procurement activities, all safety-related materials, items, and services are to be procured from TVA-accepted suppliers and must comply with TVA-specified technical, functional, and quality requirements. TVA reviews applicable NNSA TPBAR procurement documents for acceptance in order to ensure that the NNSA documents used to obtain safety-related materials, items, and services adequately implement the TVA requirements.

There are currently two NNSA direct suppliers involved in the supply of TPBARs. The Pacific Northwest National Laboratory (PNNL), a Department of Energy (DOE) Office of Science site operated by the Battelle Memorial Institute, performs TPBAR design and procurement activities. WesDyne International LLC (WesDyne) performs procurement and fabrication activities. Both of these suppliers are on the TVA ASL for their respective current scopes of work.

NNSA identifies the TVA requirements to project participants in the annual Tritium Sustainment Program Implementation Plan. Evidence of project participant implementation of the TVA requirements is demonstrated through a TPBAR Design Interface Agreement between TVA and PNNL and the WesDyne Project Quality Plan for TPBAR Fabrication. Sub suppliers are required to meet the applicable QA program requirements as determined by the procuring organization (i.e., PNNL or WesDyne). The NNSA direct suppliers have been audited by TVA to ensure compliance with Appendix B requirements. PNNL was initially placed on the TVA ASL under the Lead Test Assembly (LTA) Project and maintained on the ASL through annual evaluations and triennial audits. WesDyne has been on the TVA ASL since 2001.

Figure 4.1-4 depicts a summary view of the flow of QA program requirements.

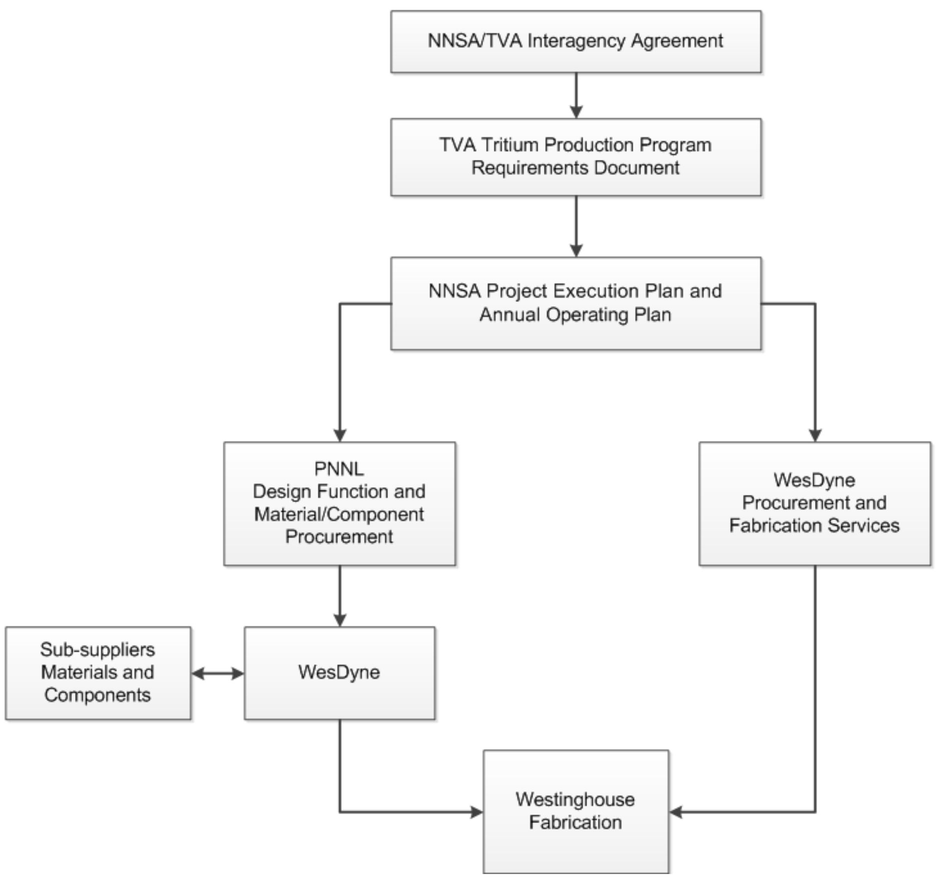

Figure 4.1-4: TPBAR QA Program Requirements Flow

The NNSA procures TPBAR design, fabrication, irradiation, and transportation services for the delivery of irradiated TPBARs to the NNSA Tritium Extraction Facility. The major NNSA suppliers are PNNL, WesDyne, TVA, and a supplier for irradiated TPBAR Transportation Services.

The PNNL in Richland, Washington developed and qualified the design and fabrication processes, fabricated and delivered TPBARs for use as LTAs, obtained LTA irradiation services from TVA, and performed LTA TPBAR post irradiation examinations. PNNL will provide design evolution and fabrication process improvements associated with supporting full-scale TPBAR fabrication and material and subcomponent procurements. TVA-TPPR-99-02 is the TPBAR Design Interface Agreement between TVA and PNNL. TTP-7-065 is the PNNL-controlled document that describes how requirements for TPBARs are met for the interface between PNNL (the TPBAR design authority) and WesDyne (the TPBAR fabricator). It flows down the requirements of TVA-TPPR-99-01 as well as addressing additional interface requirements, test articles, and services that may be exchanged. It identifies the roles and responsibilities of WesDyne, PNNL, and TVA in the evaluation of nonconforming items.

WesDyne is a wholly owned subsidiary of the Westinghouse Electric Company LLC that operates under a separate Board of Directors. WesDyne uses the Westinghouse Quality Management System (QMS). WD-TP-23.1.3 is a WesDyne controlled document that defines the responsibilities and interface requirements between WesDyne (a subsidiary of Westinghouse Electric Co.) and Westinghouse Global Quality for evaluation of TPBAR suppliers. WesDyne is ultimately responsible for TPBAR fabrication and provides management and QA oversight of all TPBAR fabrication activities. WesDyne performs the work for the TPBAR program to a Project Quality Plan (PQP). Per the PQP, the Westinghouse Quality Management System procedures allow WesDyne to utilize the Westinghouse Quality Suppliers List. WesDyne subcontracts a significant portion of the fabrication of its work at the Westinghouse Columbia Fuel Fabrication Facility (CFFF) to Westinghouse Nuclear Fuels. Westinghouse Global Quality annually performs internal audits of Nuclear Fuels activities at CFFF, including implementation of the TPBAR program QA requirements on behalf of WesDyne. Westinghouse Global Quality also performs internal audits of the WesDyne TPBAR Fabrication Program.

The WesDyne TPBAR Fabrication Facility, located at the Westinghouse Fuel Fabrication Plant in Columbia, South Carolina will procure materials and services, assemble, process, and fabricate TPBARs. Westinghouse assembles the TPBARs onto the TPBAR baseplates and delivers certified TPBARs to TVA for use in TVA reactor cores under a separate TVA purchase order. The WesDyne TPBAR Fabrication Project Quality Plan commits to compliance with the QA program requirements using the latest revision of the NRC-approved Westinghouse QMS that meets 10 CFR Part 50, Appendix B requirements. WesDyne requires their subcontractors and suppliers to implement QA programs that meet the applicable 10 CFR Part 50, Appendix B requirements, as well as applicable requirements from the TVA requirements document. WesDyne has subcontracted TPBAR assembly to the Westinghouse Nuclear Fuels in Columbia, SC and has developed an interface agreement with Westinghouse Nuclear Services for support services. TVA has audited WesDyne and placed WesDyne on the TVA ASL with some restrictions. TPBAR assembly has been performed since 2001.

Quality oversight (such as program reviews, source surveillances and audits) of material, service, and subcomponent suppliers are the responsibility of the procuring organization (i.e., PNNL or WesDyne) with periodic participation by a TVA observer. The responsibilities for procurement of materials are split between WesDyne and PNNL. The PNNL QA program meets 10 CFR Part 50 Appendix B requirements. PNNL requires their material, component, and service suppliers to establish and implement QA programs that meet the requirements of 10 CFR Part 50, Appendix B,

and 10 CFR Part 21. These programs are audited by PNNL with periodic participation by a TVA observer. A listing of major suppliers for production TPBARs is contained in Table 4.1-1.

Table 4.1-1: Major TPBAR Component Suppliers

Supplier	Product/Service	QA Program Status	Supplier Status
Veridium	SS Cladding Tubes	Established and Implemented	Approved
Millennitek	$LiALO_2$ Pellets	Established and Implemented	Approved
Superior Tube Company	Getter, Spacer and Liner Tubes (Zirconium)	Established and Implemented	Approved
Hohman Plating and Manufacturing	Plating of getter tubes and spacers	Established and Implemented	Approved
Hitemco	Coating of Cladding Tubes	Established and implemented	Approved

Upon receipt of certified TPBARs, TVA's fuel vendor will install TPBARs onto baseplates in accordance with their respective NRC accepted QA Program.

TVA will irradiate the NNSA furnished TPBARs. After irradiation, TVA will consolidate TPBARs and prepare them for NNSA shipments to the Tritium Extraction Facility.

The activities associated with TPBAR design, material and service procurements, fabrication, and delivery are being performed under the auspices of TVA's NRC Accepted QA program (TVA-NQA-PLN89A).

TVA is responsible for obtaining safety-related components and services from TVA accepted suppliers. NNSA is managing the overall Tritium Sustainment Program including issuance of major procurements. TVA requires that all safety-related materials, items, and services be procured from TVA accepted suppliers and comply with TVA specified technical, functional, and quality requirements. In order to ensure that the NNSA documents used to obtain safety-related materials, items and services adequately address the TVA requirements, TVA reviews applicable NNSA documents for acceptance.

TVA evaluates PNNL and WesDyne for TPBAR design, material and service procurements, fabrication and assembly, and delivery and places them on TVA's ASL. TVA maintains a list of acceptable suppliers in accordance with TVA's NRC accepted QA program. Maintenance of suppliers on TVA's ASL includes annual evaluations, audits, and surveillance of selected supplier activities.

In the area of transportation of radioactive materials, NNSA will furnish a certified transportation package for TVA's use in preparing irradiated TPBARs for transportation. NNSA will be the shipper of record. TVA's scope includes preparing the irradiated TPBARs for transportation by loading irradiated TPBAR consolidation containers into a certified transportation package, loading the package onto the transport vehicle, and preparing shipping papers for NNSA. TVA will implement the

applicable portions of TVA's NRC-approved Radioactive Material Package QA Plan associated with use of licensed/certified transportation packages, including that the package supplier is a TVA accepted supplier.

4.1.3 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 TPBAR Interface Issue 3 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 2, 9, 15, 16, 17, and 18).

In NUREG-1672 (Reference 8), the NRC staff identified compliance with the Departure from Nucleate Boiling (DNB) criterion as an interface issue for which plant-specific information would be required in the licensee's submittal to support an amendment to the facility operating license for authorization to operate a tritium production core. This criterion requires the demonstration that DNB would not occur on the most limiting fuel rod on at least a 95 percent probability at a 95 percent confidence level. For the WBN Unit 1 1,792 TPBAR Tritium Production equilibrium cycle, the normal Thermal-Hydraulic DNB related reload analyses were performed using VIPRE-01 (Reference 19) and are described in more detail below. Due to the power level, fuel type, and other plant parameters used in the Thermal-Hydraulic DNB analysis, the analysis for WBN Unit 1 applies to WBN Unit 2. The following detailed thermal-hydraulic evaluations were performed for WBN Unit 1.

1. An axial power shape study was performed to assure that the power distributions used in design would still be valid in the presence of the TPBAR. This study compares power shapes resulting from depletion during operation of the cycles to reference shapes used as the basis for Thermal-hydraulic design analyses.
2. The Steamline Break with Rod Withdrawal at Power transient was analyzed to demonstrate the continued acceptability of the Departure from Nucleate Boiling Ratio (DNBR) design basis for this transient.
3. The Zero Power Hypothetical Steamline Break was analyzed to demonstrate that the DNBR design basis was met.

The axial power shape comparison showed that, with the assumption of the current operation strategy, the reference power shapes assumed in the current safety analysis for Watts Bar would remain bounding. The TPBARs do not present any excessive power distribution changes beyond those which are already bounded within the thermal-hydraulic design bases. The results of the DNB analyses showed that the DNBR design basis was met. In addition, the core bypass flow limit was shown to be met with the presence of the TPBARs, and there was no bulk boiling in the thimble or surface boiling in the dashpot.

Therefore, the presence of TPBARs in the reload core design does not challenge the DNB criterion. An explicit check of the DNB criterion is included in the cycle-specific

reload safety evaluation performed for each Watts Bar Unit 2 reload core. Continued performance of this check will validate the acceptability of each reload core for operation within the DNB design limits.

4.1.4 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 for the WBN Unit 2 Tritium Production Program LAR in WCAP-18191-NP, Revision 0, “Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations.” This report, which is included as Enclosure 2 to this letter, describes the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the WBN Unit 2 reactor vessel. The analyses consider implementation of TPBARs at the beginning of Cycle 4. The heatup and cooldown P-T limit curves were generated using the limiting Adjusted Reference Temperature (ART) values for WBN Unit 2. The limiting ART values were those of Intermediate Shell Forging 05 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations.

The P-T limit curves were generated for 32 effective full-power years (EFPY) using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the ASME Code, Section XI, Appendix G. The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4. Heatup rates of 60 and 100°F/hr, and cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr were used to generate the P-T limit curves, with the flange requirements and without margins for instrumentation errors. The WBN Unit 2 End of License (EOL) corresponding to 40 years of operation is 32 EFPY. The EOL P-T limit curves can be found in Figures 8-1 and 8-2 in the Enclosure 2 report.

Appendix A to WCAP-18191-NP contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 32 EFPY.

Appendix B to WCAP-18191-NP contains a P-T limit evaluation of the reactor vessel inlet and outlet nozzles based on a 1/4T flaw postulated at the inside surface of the reactor vessel nozzle corner, where T is the thickness of the nozzle corner region. As discussed in Appendix B, the P-T limit curves generated based on the limiting cylindrical beltline material (Intermediate Shell Forging 05) bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for WBN Unit 2 at 32 EFPY.

Appendix C to WCAP-18191-NP contains discussion of the other ferritic Reactor Coolant Pressure Boundary (RCPB) components relative to P-T limits. As discussed in Appendix C, the other ferritic RCPB components meet the applicable requirements of Section III of the ASME Code.

Appendix D to WCAP-18191-NP contains an upper-shelf energy (USE) evaluation for the WBN Unit 2 reactor vessel beltline and extended beltline materials. Per Appendix D, all beltline and extended beltline materials are projected to maintain USE values above the 50 ft-lb screening criterion per 10 CFR 50 Appendix G at 32 EFPY.

Appendix E to WCAP-18191-NP contains a pressurized thermal shock (PTS) evaluation for the WBN Unit 2 reactor vessel beltline and extended beltline materials. Per Appendix E, all beltline and extended beltline materials have projected reference temperature for pressurized thermal shock (RT_{PTS}) values below the screening criteria set forth in 10 CFR 50.61. Additionally, WBN Unit 2 will remain in Category I of the Emergency Response Guidelines through 32 EFPY.

Appendix F to WCAP-18191-NP contains an updated surveillance capsule withdrawal schedule. Per Appendix F, three surveillance capsules are recommended to be withdrawn from the WBN Unit 2 reactor before end of license.

4.1.5 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 Watts Bar Unit 2 Tritium Production Program LAR is based on the applicable Watts Bar Unit 1 and Unit 2 precedent documents (References 1, 2, 9, 17, 20, 21, 22, 23, and 24).

<u>Tritium 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. Therefore, no TPBAR failures are predicted to occur during design-basis accidents except for a large break loss of cooling accident (LBLOCA) or a fuel handling accident (FHA) involving TPBARs. The source terms associated with the LBLOCA and

FHA include fuel inventory. The source terms associated with the MSLB, SGTR, LOOP, and WGDT are all based on the primary and secondary coolant concentrations. The effect of a TPC on each of these source terms is discussed below.

The current licensing basis regulatory limits for WBN Unit 2 are established in terms of whole body dose, beta dose, and thyroid dose, except for the Fuel Handling Accident, 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 Tritium Production Core (TPC), TVA included calculated TEDE for the Fuel Handling Accident, and, for informational purposes, the remaining accidents.

Core Inventory

The core inventory used in the current WBN U2 licensing basis has been replaced to account for a TPC. Changes made to the WBN U2 core inventory used to determine the radiological consequences of the LBLOCA and FHA are the same as those used for WBN U1 and approved in WBN U1 License Amendment 40. This consisted of calculating the core inventory utilizing ORIGEN2.1 and assuming that all tritium in the TPBARs is released to the environment. The WBN U2 retained the conservative assumption of 2,304 TPBARs as utilized in the WBN U1 License Amendment 40. This analysis included a tritium inventory of 1.2 grams of tritium/TPBAR, which results in a total of 2.68E+07 curies (Ci) of tritium in the core. The rest of the core inventories were determined based on a 96 feed equilibrium cycle which consisted of 96 once burned assemblies, 96 twice burned assemblies and 1 thrice burned assembly. The inventory of each set of fuel assemblies was then summed together to determine the total core inventory. The current inventory was determined based on a core average of 1020 EFPD. There is also a difference in the power level assumed. The current inventory was based on 104.5% of the licensed power (3411 MWt). The TPC inventory was based on 102% of the licensed power which is consistent with that allowed by the WBN U2 operating license and assumed ECCS uncertainty. Table 4.1-2 provides information used in the determination of the TPC inventory. Table 4.1-3 provides the core inventory used.

Table 4.1-2: Parameters used to determine the TPC core inventory

Parameter	Current Value	TPC Value
Power (MWt)	3565	3480
Cycle Energy objective (EFPD)	1020 average	510/cycle
Enrichment	5.0	4.95

Table 4.1-3: Core Inventory for TPC and Conventional Core

Nuclide	Inventory	1X Burned	2X Burned	3X Burned	Current LOCA	Current FHA
	Ci	Ci/assembly	Ci/assembly	Ci/assembly	Ci	Ci/assembly
Kr-83m	1.23E+07	7.63E+04	5.15E+04	6.13E+04	1.15E+07	5.20E+04
Kr-85m	2.69E+07	1.69E+05	1.10E+05	1.25E+05	2.39E+07	1.04E+05
Kr-85	8.81E+05	3.56E+03	5.54E+03	6.84E+03	1.03E+06	7.02E+03
Kr-87	5.23E+07	3.31E+05	2.11E+05	2.36E+05	4.81E+07	2.06E+05
Kr-88	7.38E+07	4.68E+05	2.97E+05	3.31E+05	6.66E+07	2.82E+05
Kr-89	9.10E+07	5.81E+05	3.63E+05	3.97E+05	8.28E+07	3.44E+05
Xe-131m	9.54E+05	5.31E+03	4.56E+03	6.18E+03	1.05E+06	5.64E+03
Xe-133m	5.80E+06	3.41E+04	2.60E+04	3.45E+04	6.16E+06	3.22E+04
Xe-133	1.88E+08	1.11E+06	8.36E+05	1.09E+06	1.19E+08	9.63E+05
Xe-135m	3.59E+07	2.08E+05	1.63E+05	2.19E+05	4.05E+07	2.16E+05
Xe-135	4.96E+07	2.84E+05	2.30E+05	2.19E+05	6.43E+07	2.90E+05
Xe-138	1.59E+08	9.55E+05	6.93E+05	8.79E+05	1.67E+08	8.31E+05
I-131	9.01E+07	5.24E+05	4.09E+05	5.49E+05	9.46E+07	4.94E+05
I-132	1.31E+08	7.63E+05	5.89E+05	7.87E+05	1.39E+08	7.21E+05
I-133	1.88E+08	1.11E+06	8.35E+05	1.09E+06	1.95E+08	1.00E+06
I-134	2.08E+08	1.23E+06	9.18E+05	1.19E+06	2.16E+08	1.10E+06
I-135	1.76E+08	1.04E+06	7.81E+05	1.02E+06	1.86E+08	9.60E+05

Primary and Secondary Coolant Concentrations

The current licensing basis primary and secondary coolant concentrations for WBN Unit 2 are based on ANSI/ANS-18.1-1984. The concentrations used for the TPC were also based on this standard, but the input values have been changed to correct errors found during the review process. Therefore, the concentrations have changed since the original review of the WBN Unit 2 operating license. The following are the errors that were corrected for WBN Unit 2:

1. The RCS volume was corrected from 11375 cubic feet to 12708.4 cubic feet.
2. The specific volume previously used to determine the RCS weight was based on a temperature outside the normal operating range.
3. The weight of RCS water previously included the volume of vapor space in the pressurizer.
4. The weight of water in the SG previously included the weight of water in the primary side of the SG instead of just the secondary side.
5. The condensate demineralizer was previously assumed to be in operation but is not typically used and should not have been credited.

Along with correction of these errors, the analysis was also updated to make the power level consistent with the licensing basis. Currently the assumed power level for this analysis is 3565 MWt which is 104.5% of the licensed power level. However, the ECCS uncertainty is 2%, so 102% of the licensed power was assumed (3480 MWt). Table 4.1-4 provides the parameters used and Table 4.1-5 provides the resulting radionuclide concentrations.

The concentration of tritium for a TPC was calculated using the same methodology as used for WBN U1, which involved 1,792 TPBARs with a permeation rate of 5 Ci/TPBAR/year and two TPBAR failures. 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 H-3 concentration should be assumed to be 1.0 μCi/gm in the primary coolant. The total annual tritium expected for a non-TPC is 914 Ci/year. The total annual tritium expected from TPBARs is based on 1792 TPBARs and a permeation rate of 5 Ci/TPBAR/year (8960 Ci/year). This results in an average tritium concentration of 11.4 μCi/gm. The concentration with 2 TPBAR failures was determined by adding the inventory of 2 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 of 11,600 Ci at the end of a cycle. This resulted in an expected tritium concentration in the primary coolant of approximately 120 μCi/gm for 2 TPBAR failures.

Table 4.1-4: Parameters used for the Primary and Secondary Coolant Concentrations

Parameter	Current Value	New Value
Thermal Power (MWt)	3582	3480
Steam Flow Rate (lb/hr)	1.5E+07	1.5E+07
Weight of Water in RCS (lb)	5.4E+05	4.71E+05
Weight of water in all SGs (lb)	3.48E+05	3.80E+05
Reactor coolant letdown flow rate (purification)(lb/hr)	3.7E+04	3.7E+04
Reactor coolant letdown flow rate (yearly average for boron control)(lb/hr)	845	845
Steam Generator Blowdown Flow (lb/hr)	3.00E+04	3.00E+04
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	1.0	1.0
Flow through the purification system cation demineralizer	3.7E+03	3.7E+03
Ratio of condensate demineralizer flow rate to the total steam flow rate	0.55	0.0
Fraction of the noble gas activity in the letdown stream which is not returned to the RCS	0.0	0.0

Table 4.1-5: Primary and Secondary Coolant Concentrations

Nuclide	Reactor Coolant WBN	Secondary Water WBN	Secondary Steam WBN
	uCi/gm	uCi/gm	uCi/gm
<u>Class 1</u>			
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	1.14E+01	1.14E-02	1.14E-02
2 TPBAR failure	1.20E+02	1.20E-01	1.20E-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
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

Table 4.1-6 and 4.1-7 contain a tabulation of common control room parameters and the atmospheric dispersion factors, respectively, used in each of the design basis analyses. These are the same parameters that supported the review for NUREG-0847 Supplement 25 except for the control room isolation time, which has been corrected to account for an error. This is the same error noted in the WBN U1 License Amendment 107. The control room radiation monitor loops utilize the RP-30AM analog rate meter. A time constant of 7.17E-3 minutes was previously used to determine the rate meter response time, which would be appropriate for a count rate between 1E4 and 1E5 counts per minute (cpm). However, the setpoint for these monitors is 400 cpm; thus a time constant of 4.34E-1 minutes should have been used. This resulted in an increase in the rate meter response time from 0.86 seconds to 52.08 seconds. Combined with the response times determined for the remainder of the loop, the total loop response time increased from 6.6 seconds to 57.8 seconds. The analyses rounded this to 60 seconds. The isolation damper response time remains unchanged.

Table 4.1-6: Control Room Parameters

Parameter	Value
Volume	257,198 ft^3
Makeup/pressurization flow	711 cfm
Recirculation flow	2889 cfm
Unfiltered intake	51 cfm
Filter efficiency	
First pass	95%
Second pass	70%
RM loop response time Isolation damper response time Total Isolation time*	60 sec 14 sec 74 sec
Occupancy factors	
0–24 hours	100%
1–4 days	60%
4–30 days	40%

* not used for the LBLOCA as an SI will automatically initiate control room isolation

Table 4.1-7: Atmospheric Dispersion Factors (sec/m^3)

Time Period (hr)	EAB	LPZ	LOCA/FHA[1]	SGTR/MSLB/LOOP	WGDT/FHA[2]
0-2	6.382E-04	1.784E-04	1.09E-03	2.59E-03	2.56E-03
2-8	-	8.835E-05	9.44E-04	2.12E-03	-
8-24	-	6.217E-05	1.56E-04	-	-
24-96	-	2.900E-05	1.16E-04	-	-
96-720	-	9.811E-06	9.59E-05	-	-

[1] - This is used for the Containment FHA
[2] - This is used for the Containment FHA after containment is isolated and for the Auxiliary Building FHA

Large Break Loss of Cooling Accident

The current LBLOCA analysis of record for WBN U2 was revised to account for a TPC by utilizing the core inventory calculated for a TPC as described above. All other parameters remain the same as documented in Section 15.4.1 of NUREG-0847 Supplement 25 (Reference 22) and are provided in Tables 4.1-8 thru 4.1-11.

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 Unit 2 LBLOCA offsite radiological dose consequences for the Control Room (CR), 2-hour Exclusion Area Boundary (EAB), and 30 day Low Population Zone (LPZ) analysis results shown in Table 4.1-12 are below the 10 CFR Part 100 and 10 CFR Part 50 Appendix A General Design Criterion (GDC) 19 regulatory limits. All results include the contribution from ECCS leakage outside containment.

Table 4.1-8: Parameters used in the LBLOCA Analysis

Parameter	Value
Primary containment free volume	1.27 E+06 ft^3
Shield building annulus free volume	3.75 E+05 ft^3
Primary containment deck (air return) fan flow rate	40,000 cfm
Number of containment deck air return fans operating	1 of 2
Fractions of core inventory available for release	
Noble gases	100%
Iodines	25%
Initial iodine composition in containment	
Elemental	91%
Organic	4%
Particulate	5%
Primary containment leak rates	
0–24 hr	0.25% per day
1–30 days	0.125% per day
Percent of primary containment leakage to auxiliary building	25%
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Delay time of activity in auxiliary building before ABGTS operation	None
Delay time before filtration credit is taken for the ABGTS	4 minutes
Mean holdup time in auxiliary building after initial 4 minutes	0.3 hours
ABGTS flow rate	9000 cfm
Leakage from auxiliary building to ABGTS downstream HVAC (bypass of filters)	27.88 cfm
Leakage from ABGTS HVAC into auxiliary building	8.87 cfm

Leakage from auxiliary building into EGTS downstream HVAC (bypass of filters)	10.7 cfm
Leakage from auxiliary building to environment from single failure of ABGTS (from 30 minutes to 34 minutes post-LOCA)	9900 cfm (for 4 minutes)
Percent of primary containment leakage to annulus	75%
Percent of annulus free volume available for mixing of recirculated activity	50%
Number of emergency gas treatment system air-handling units operating	1 of 2
Emergency gas treatment system filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
EGTS Total Flow	3600 cfm/train
Shield Building Mixing Model	50%
ECCS leakage outside containment	3760 cc/hr
Sump Volume	9.63E+04 ft^3
Core inventory of iodine released to the sump	50%
Iodine partition factor	10

Table 4.1-9: Ice Condenser Elemental and Particulate Removal Efficiency

Time Interval Post-LOCA (Hours)	Removal Efficiency
0.0 to 0.156	0.96
0.156 to 0.267	0.76
0.267 to 0.323	0.73
0.323 to 0.489	0.71
0.489 to 0.615	0.60
0.615 to 0.768	0.58
0.768 to 0.824	0.40
0.824 to 720	0.0

Table 4.1-10: EGTS Flow Rates for failure of a single EGTS train case

Time Interval (sec)	Time Interval (hours)	Recirculation Rate (cfm)	Recirculation Rate (cfh)	Exhaust Rate (cfm)	Exhaust Rate (cfh)
0-30	0.00-0.0083	0.00	0.00E+00	0.00	0.00E+00
30-39	0.0083-0.0108	3600.00	2.16E+05	0.00	0.00E+00
39-40	0.0108-0.0111	3286.62	1.97E+05	313.38	1.88E+04
40-41	0.0111-0.0114	2352.31	1.41E+05	1247.69	7.49E+04
41-42	0.0114-0.0117	1304.79	7.83E+04	2295.21	1.38E+05
42-43	0.0117-0.0119	362.60	2.18E+04	3237.40	1.94E+05
43-190	0.0119-0.0528	0.00	0.00E+00	3600.00	2.16E+05
190-191	0.0528-0.0531	537.28	3.22E+04	3062.72	1.84E+05
191-192	0.0531-0.0533	733.23	4.40E+04	2866.77	1.72E+05
192-193	0.0533-0.0536	735.14	4.41E+04	2864.86	1.72E+05
193-194	0.0536-0.0539	737.51	4.43E+04	2862.49	1.72E+05
194-199	0.0539-0.0553	745.23	4.47E+04	2854.77	1.71E+05
199-207	0.0553-0.0575	764.12	4.58E+04	2835.89	1.70E+05
207-215	0.0575-0.0597	790.80	4.74E+04	2809.20	1.69E+05
215-225	0.0597-0.0625	825.45	4.95E+04	2774.56	1.66E+05
225-245	0.0625-0.0681	892.72	5.36E+04	2707.29	1.62E+05
245-265	0.0681-0.0736	992.80	5.96E+04	2607.20	1.56E+05
265-285	0.0736-0.0792	1102.40	6.61E+04	2497.61	1.50E+05
285-305	0.0792-0.0847	1217.05	7.30E+04	2382.95	1.43E+05
305-446	0.0847-0.1239	1664.05	9.98E+04	1935.96	1.16E+05
446-601	0.1239-0.1669	2356.72	1.41E+05	1243.29	7.46E+04
601-602	0.1669-0.1672	2661.35	1.60E+05	938.65	5.63E+04
602-1700	0.1672-0.4722	3600.00	2.16E+05	0.00	0.00E+00
1700-1701	0.4722-0.4725	3508.13	2.10E+05	91.87	5.51E+03
1701-1702	0.4725-0.4728	3423.44	2.05E+05	176.56	1.06E+04
1702-1703	0.4728-0.4731	3410.73	2.05E+05	189.27	1.14E+04
1703-1704	0.4731-0.4733	3408.66	2.05E+05	191.34	1.15E+04
1704-1705	0.4733-0.4736	3408.17	2.04E+05	191.83	1.15E+04
1705-1706	0.4736-0.4739	3407.91	2.04E+05	192.09	1.15E+04
1706-1855	0.4739-0.5153	3395.23	2.04E+05	204.77	1.23E+04
1855-2100	0.5153-0.5833	3372.37	2.02E+05	227.64	1.37E+04
2100-30days	0.5833-720	3350.00	2.01E+05	250.00	1.50E+04

Table 4.1-11:EGTS Flow Rates for PCO Failure Case

Time Interval		Time Interval		Recirculation Rate		Exhaust Rate	
(sec)	**(sec)**	**(hrs)**	**(hrs)**	**(cfm)**	**(cfh)**	**(cfm)**	**(cfh)**
0	30	0	0.0083	0.00E+00	0.00E+00	0.00E+00	0.00E+00
30	39	0.0083	0.0108	7.20E+03	4.32E+05	0.00E+00	0.00E+00
39	40	0.0108	0.0111	6.57E+03	3.94E+05	6.27E+02	3.76E+04
40	41	0.0111	0.0114	4.70E+03	2.82E+05	2.50E+03	1.50E+05
41	42	0.0114	0.0117	2.61E+03	I.57E+05	4.59E+03	2.75E+05
42	43	0.0117	0.0119	7.25E+02	4.35E+04	6.47E+03	3.88E+05
43	71	0.0119	0.0197	0.00E+00	0.00E+00	7.20E+03	4.32E+05
71	78	0.0197	0.0217	0.00E+00	0.00E+00	7.20E+03	4.32E+05
78	79	0.0217	0.0219	1.06E+03	6.37E+04	6.14E+03	3.68E+05
79	80	0.0219	0.0222	4.78E+03	2.87E+05	2.43E+03	1.46E+05
80	102	0.0222	0.0283	4.34E+03	2.60E+05	2.86E+03	1.72E+05
102	132	0.0283	0.0367	4.19E+03	2.51E+05	3.01E+03	1.81E+05
132	165	0.0367	0.0458	3.92E+03	2.35E+05	3.28E+03	1.97E+05
165	170	0.0458	0.0472	3.76E+03	2.26E+05	3.44E+03	2.06E+05
170	210	0.0472	0.0583	3.72E+03	2.23E+05	3.48E+03	2.09E+05
210	307	0.0583	0.0853	3.76E+03	2.26E+05	3.44E+03	2.06E+05
307	498	0.0853	0.1383	4.05E+03	2.43E+05	3.15E+03	1.89E+05
498	602	0.1383	0.1672	4.80E+03	2.88E+05	2.40E+03	1.44E+05
602	603	0.1672	0.1675	5.23E+03	3.14E+05	1.97E+03	1.18E+05
603	850	0.1675	0.2361	5.14E+03	3.08E+05	1.43E+03	8.59E+04
850	1100	0.2361	0.3056	5.24E+03	3.14E+05	1.33E+03	7.99E+04
1100	1350	0.3056	0.375	5.34E+03	3.20E+05	1.23E+03	7.39E+04
1350	1600	0.375	0.4444	5.44E+03	3.26E+05	1.13E+03	6.79E+04
1600	1850	0.4444	0.5139	5.54E+03	3.32E+05	1.03E+03	6.19E+04
1850	2100	0.5139	0.5833	5.64E+03	3.38E+05	9.32E+02	5.59E+04
2100	3600	0.5833	1	5.74E+03	3.44E+05	8.32E+02	4.99E+04
3600	30 days	1	30 days	3.46E+03	2.07E+05	6.04E+02	3.62E+04

Table 4.1-12: Dose Consequences from an LBLOCA

Watts Bar Unit 2 Dose Consequences from an LBLOCA					
Single Train EGTS Case					
Dose (rem)	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	8.87E-01	5	2.07E+00	1.89E+00	25
Beta	7.49E+00	30	1.14E+00	2.26E+00	300
Thyroid	3.62E+00	30	3.87E+01	1.38E+01	300
TEDE	2.24E+00	5	3.59E+00	2.30E+00	25
PCO Control Failure Case					
Dose (rem)	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	1.07E+00	5	2.42E+00	2.30E+00	25
Beta	9.10E+00	30	1.38E+00	2.61E+00	300
Thyroid	3.09E+00	30	3.03E+01	1.19E+01	300
TEDE	2.51E+00	5	3.38E+00	2.48+00	25

Fuel Handling Accident (FHA)

The WBN U2 FHA current analysis of record was revised to account for a TPC by utilizing the activity per fuel assembly determined for a TPC as described above. The tritium activity assumed to release to the environment (21,122.5 Ci) is the same that was assumed for WBN Unit 1 as approved in Amendment 107 (Reference 2). The analysis was also updated to correct an error in the control room isolation time as described above. All other parameters remain the same as documented in Section 15.4.5 of NUREG-0847 Supplement 25 (Reference 22) and are provided in Table 4.1-13. The Watts Bar Unit 2 FHA is the same as the Watts Bar Unit 1 FHA; input parameters that bounded U1 and U2 were utilized.

As described in Section 15.4.5 of NUREG-0847 Supplement 25 (Reference 22), three cases for the FHA were analyzed. The first case considered a FHA inside containment with the containment penetrations closed to the auxiliary building and the reactor building purge ventilation system (RBPVS) operating. This case was evaluated using the assumptions from Regulatory Guide (RG) 1.25, issued March 1972. This case is bounded by the third case and is no longer analyzed.

The second case considered is a FHA in the spent fuel pool (SFP) area located in the auxiliary building. This case was evaluated using the assumptions from RG 1.183, issued July 2000. In this case, no credit is taken in the analysis for the auxiliary building gas treatment system (ABGTS).

The third case is an open containment case for a 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 results of the second and third cases are shown in Table 4.1-14 below.

A TPBAR only accident in the SFP was also evaluated. This postulated event is assumed to result in 21,122.5 Ci of tritium being released over 2 hours. Since tritium is low energy beta decay only, the Spent Fuel Pit monitors and the Control Room Intake monitors will not respond to the tritium; therefore, the Auxiliary Building Exhaust will not be isolated, resulting in all releases being discharged out the Auxiliary Building vent.

Table 4.1-13: Parameters used in the FHA

Number of fuel assemblies damaged	1 (all rods ruptured)
Minimum postshutdown fuel-handling time (decay time)	100 hours
Minimum pool water depth	23 feet
Fuel clad damage gap release fractions	
I-131	8%
Remainder of halogens	5%
Kr-85	10%
Remainder of noble gases	5%
Pool DF	

Noble gases and organic iodine	1
Overall iodine (23 ft of water cover)	200 (effective DF)
Chemical form of iodine released	
Elemental	99.85%
Organic	0.15%
Filter efficiencies	None
Duration of release to the environment	2-hour release

Table 4.1-14: Dose Consequences from an FHA

Watts Bar Unit 2 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	2.39E+00	5	2.83E+00	7.92E-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	2.33E+00	5	2.83E+00	7.92E-01	6.25

The Watts Bar Unit 2 FHA radiological dose consequences analysis results shown in Table 4.1-14 are below the 10 CFR 50.67 regulatory limits.

Main Steam Line Break (MSLB) and Steam Generator Tube Rupture (SGTR)

Analyses for the WBN Unit 2 MSLB and SGTR were revised to account for a TPC by utilizing the average tritium concentration in the primary and secondary coolant with two TPBAR failures. The WBN Unit 2 analyses were updated to utilize the corrected primary and secondary coolant concentrations and to correct an error in the control room isolation time as described above. All other parameters remain the same as documented in Sections 15.4.2 and 15.4.3 of NUREG-0847 Supplement 25 (Reference 22) and are provided in Tables 4.1-15 and 4.1-16.

The Watts Bar Unit 2 calculated radiological consequences for the MSLB and SGTR with a 1,792 TPBAR core, as shown in Table 4.1-17 and Table 4.1-18, remain well within 10 CFR Part 100 and 10 CFR Part 50 Appendix A GDC 19 dose limits.

Table 4.1-15: Parameters used for the MSLB

Initial maximum RCS equilibrium activity	0.265 µCi/g
Accident-initiated iodine spike appearance rate	500 times equilibrium rate
Maximum preaccident spike iodine concentration	14.0 µCi/gm
Secondary coolant iodine activity	0.1 µCi/gm DEI
Primary-to-secondary leak rate	
Faulted steam generator	1.0 gpm
Per intact steam generator	150 gpd
Steam generator secondary-side iodine partition coefficients	
Faulted steam generator	1 (none)
Intact steam generator	100
RCS letdown flow rate	124.39 gpm
Steam releases	
Faulted steam generator (0–30 minutes)	96,100 lbm
Three intact steam generators (0–2 hr)	433,079 lbm
Three intact steam generators (2–8 hr)	870,754 lbm
Primary to secondary side leakage for iodine production	11 gpm
Noble Gas Activity	100/Ebar
Mass of Reactor coolant	2.316E8 g
Mass of water in all SGs	1.724E8 g
Iodine Dose Conversion Factors	RG 1.109

Table 4.1-16: Parameters used for the SGTR

Initial maximum RCS equilibrium activity	0.265 µCi/g
Accident-initiated iodine spike appearance rate	500 times equilibrium rate
Maximum pre-accident spike iodine concentration	14.0 µCi/gm
Secondary coolant iodine activity	0.1 µCi/gm DEI
Primary-to-secondary leak rate	
Faulted steam generator	1.0 gpm
Per intact steam generator	150 gpd
Steam generator secondary-side iodine partition coefficients	
Faulted steam generator	1 (none)
Intact steam generator	100
Secondary-side mass release (ruptured steam generator)	
0–2 hours	103,300 lbm
2–8 hours	32,800 lbm
Secondary-side mass release (intact steam generator)	
0–2 hours	492,100 lbm
2–8 hours	900,200 lbm
Primary coolant mass release	
Total	191,400 lbm
Flashed	10,077.2 lbm
Primary to secondary side leakage for iodine production	11 gpm
Noble Gas Activity	100/Ebar
Mass of Reactor coolant	2.316E8 g
Mass of water in all SGs	1.724E8 g
Iodine Dose Conversion Factors	RG 1.109

Table 4.1-17: Dose Consequences from MSLB Accident

Watts Bar Unit 2 Dose Consequences from MSLB Accident					
Parameter	**Dose (rem)**				
Pre-Accident Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	3.68E-03	5	2.50E-02	1.05E-02	25
Beta	3.60E-02	30	8.15E-03	4.03E-03	300
Thyroid	7.51E+00	30	2.41E+00	1.21E+00	300
TEDE	2.64E-01	5	1.71E-01	8.18E-02	25
Accident Initiated Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	7.85E-03	5	1.12E-01	1.39E-01	2.5
Beta	6.45E-02	30	2.70E-02	3.34E-02	30
Thyroid	1.09E+01	30	3.34E+00	5.32E+00	30
TEDE	4.01E-01	5	3.67E-01	5.38E-01	2.5

Table 4.1-18: Dose Consequences from SGTR Accident

Watts Bar Unit 2 Dose Consequences from SGTR Accident					
Parameter	**Dose (rem)**				
Pre-Accident Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	6.47E-02	5	4.11E-01	1.21E-01	25
Beta	7.23E-01	30	2.37E-01	7.26E-02	300
Thyroid	1.31E+01	30	1.44E+01	4.13E+00	300
TEDE	8.27E-01	5	1.37E+00	3.95E-01	25
Accident Initiated Spike	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	6.27E-02	5	6.39E-01	1.88E-01	2.5
Beta	7.28E-01	30	2.85E-01	8.75E-02	30
Thyroid	2.45E+00	30	8.51E+00	2.52E+00	30
TEDE	4.76E-01	5	1.39E+00	4.06E-01	2.5

Loss of Offsite Power (LOOP)

The LOOP transient dose consequence analysis was revised to account for a TPC by utilizing the average tritium concentration in the primary and secondary coolant with 2 TPBAR failures. It was also updated to utilize the corrected primary and secondary coolant concentrations and to correct an error in the control room isolation time as described above. All other parameters remain the same as documented in Section 15.4.7 of NUREG-0847 Supplement 25 (Reference 22) and are provided in Table 4.1-19.

It should be noted that there is no Standard Review Plan or Regulatory Guide for this accident. The Technical Specification limiting case is calculated utilizing a factor of 13880 as a multiplier to the realistic case. This is the scaling factor determined to scale the realistic inventory to the Technical Specification 3.7.14 limit of 0.1 µCi/gm of Iodine-131 Dose Equivalent.

Table 4.1-19: Parameters use in the LOOP Analysis

Steam generator tube leak rate	1 gpm
Fuel defects (clad damage)	
Realistic analysis	ANSI/ANS 18.1-1984
Conservative analysis	0.1 µCi/gm DEI
Iodine partition factor	0.01
Blowdown rate	25 gpm per steam generator
Duration of plant cooldown	8 hours
Steam release (total)	
0–2 hours	444,875 lbm
2–8 hours	903,530 lbm

Table 4.1-20: Dose Consequences from LOOP

Watts Bar Unit 2 Dose Consequences from LOOP					
Parameter	**Dose (rem)**				
Realistic Case	CR	CR Regulatory Limit	2 Hour EAB	30 Day LPZ	EAB and LPZ Regulatory Limit
Whole Body	9.00E-09	5	2.70E-08	1.54E-08	2.5
Beta	3.46E-04	30	2.07E-05	1.18E-05	30
Thyroid	2.58E-06	30	3.42E-06	1.96E-06	30
TEDE	5.66E-03	5	3.39E-04	1.94E-04	2.5
Technical Specification Limiting Case	CR	CR Regulatory Limit	2 Hour EAB	30 Day LPZ	EAB and LPZ Regulatory Limit
Whole Body	1.25E-04	5	3.74E-04	2.14E-04	2.5
Beta	1.72E-03	30	2.11E-04	1.21E-04	30
Thyroid	3.58E-02	30	4.74E-02	2.71E-02	30
TEDE	7.07E-03	5	3.26E-03	1.87E-03	2.5

The Watts Bar Unit 2 calculated radiological consequences for the LOOP with a 1,792 TPBAR core shown in Table 4.1-20 remain substantially below the 10 CFR Part 100 and 10 CFR Part 50 Appendix A GDC 19 dose limits.

<u>Waste Gas Decay Tank (WGDT) Rupture</u>

The WGDT dose consequence analysis was revised to account for a TPC by utilizing a tritium source term based on 2500 TPBARs with a permeation rate of 10 Ci/TPBAR/year and two TPBAR failures. It was also updated to utilize a corrected realistic source term based on the corrected primary coolant concentrations and to correct an error in the control room isolation time as described above. All other parameters remain the same as documented in Section 15.4.8 of NUREG-0847 Supplement 25 (Reference 22), and are provided in Table 4.1-21.

The Watts Bar Unit 2 calculated radiological consequences for the WGDT rupture shown in Table 4.1-22, remain substantially below 10 CFR Part 100 and 10 CFR Part 50 Appendix A GDC 19 dose limits.

Table 4.1-21: Parameter used in the WGDT Analysis

Core thermal power level	3565 MWt
Steam generator tube leak rate	1 gpm
Fuel defects (clad damage)	
Realistic analysis	ANSI/ANS-18.1-1984
Conservative analysis RG 1.24	1%
Time of accident	
Realistic analysis	After tank fill
Conservative analysis RG 1.24	End of equilibrium core cycle
Activity for Realistic Case (Ci)	
Xe-131m	5.60E+00
Xe-133	2.10E+01
Xe-133m	5.20E-01
Xe-135	3.60E+00
Xe-135m	2.80E-02
Xe-137	1.90E-03
Xe-138	2.80E-02
Kr-83m	-----
Kr-85	3.40E+00
Kr-85m	4.70E-01
Kr-87	1.50E-01
Kr-88	5.80E-01
Kr-89	----
I-131	3.30E-04
I-132	4.20E-04
I-133	8.80E-04
I-134	3.00E-04
I-135	1.10E-03
H-3	3.05+03
Activity for RG 1.24 Case (Ci)	
Xe-131m	8.9E+02
Xe-133	6.8E+04
Xe-133m	1.0E+03
Xe-135	9.4E+02
Xe-135m	4.8E+01
Xe-137	2.7E-01
Xe-138	3.2E+00
Kr-83m	1.7E+01
Kr-85	4.2E+03
Kr-85m	1.3E+02
Kr-87	2.9E+01
Kr-88	1.6E+02
Kr-89	1.0E-01
I-131	4.8E-02
I-132	----
I-133	3.3E-02
I-134	---
I-135	1.2E-2
H-3	3.05E+03

Table 4.1-22: Dose Consequences from WGDT Rupture

Watts Bar Unit 2 Dose Consequences from WGDT Rupture					
Parameter	**Dose (rem)**				
RG 1.24 Analysis	**CR**	**CR Regulatory Limit**	**2 Hour EAB**	**30 Day LPZ**	**EAB and LPZ Regulatory Limit**
Whole Body	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	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**
Whole Body	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	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 current 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 25) and subsequent Supplement 25 (Reference 22), that the FSAR 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 (Reference 22), the source term for a rod ejection accident is considerably less than for a LOCA. Because the dose consequence results for the WBN Unit 2 LOCA 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.

4.1.6 TPBAR Interface Issue 6: Specific Assessment of Hydrogen Source and Timing of Recombiner Operation

NUREG-1672, Section 2.6.2, "The staff agrees with the DOE conclusions, based on the conservative assessment of the TPBARs on the combustible gas concentrations in containment following a LOCA, that the combustible gas control systems are not expected to be affected by the TPC. However, the staff concludes that a plant-specific assessment is required to quantify the sources and to determine the time at which initiation of recombiner operation should commence to limit the hydrogen concentration to acceptable levels. 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 6 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, and 10).

Updated guidance in Regulatory Guide 1.7, Revision 3, removes operation of the recombiners from the design basis. This updated guidance has been implemented at WBN Unit 2. Whereas Section 15.4.1.2 of the previous version of the WBN UFSAR contained an evaluation of post-LOCA hydrogen generation and recombiner initiation timing, Section 15.4.1.2 of the current WBN UFSAR (applicable to Units 1 and 2) states:

> Pursuant to NRC final rule as defined in 10 CFR 50.44 and Regulatory Guide 1.7, the new definition of design-basis LOCA hydrogen release eliminates requirements for hydrogen control systems for mitigation of releases. “All PWRs with ice condenser type containments must have the capability to control combustible gas generated from metal-water reaction involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity. The deliberate ignition systems provided to meet this existing combustible gas source term are capable of safely accommodating even greater amounts of combustible gas associated with even more severe core melt sequences that fail the reactor vessel and involve molten core-concrete interaction. Deliberate ignition systems, if available, generally consume the combustible gas before it reaches concentrations that can be detrimental to containment integrity.” On the basis of this definition, no further analysis is required to support events considered to be outside the design basis. Deliberate ignition systems are described in FSAR Section 6.2.5.

Combustible gas control in containment was previously considered for WBN Unit 2, as documented in Section 6.2.5 of NUREG-0847, Supplement 22 (Reference 13).

4.1.7 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 TPBAR Interface Issue 7 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, 10, 26, and 27).

The TPBAR consolidation and shipping phase of the program has been evaluated with respect to the light load handling system.

The handling of items during TPBAR consolidation will be performed by using the SFP Bridge Crane, which utilizes a specialized fixture and tooling to transport the TPBAR assemblies, consolidate individual rods into consolidation canisters, dispose of empty baseplates, transport the canisters for storage in the SFP, and load canisters into shipping casks for transport off-site.

The weight of a fuel assembly containing 24 TPBARs (including the hold-down assembly) is less than a fuel assembly with a Rod Cluster Control Assembly (RCCA) and therefore is bounded by the current assumed weight of the assembly for purposes of analyzing fuel handling and storage facilities. The fuel assembly with TPBARs has the same external configuration as a fuel assembly without TPBARs allowing for interface with existing fuel handling/storage equipment. Additionally, this weight is conservative for purposes of defining a NUREG-0612 "Heavy Load."

During consolidation of TPBARs from a baseplate, rods are released from the baseplate one at a time. (For a description of the consolidation process, see Interface Item 1) Additionally, the consolidation fixture is designed to seismic category I(L) to preclude damage to consolidated TPBARs while in the fixture and to the SFP liner. After approximately 300 rods are released into a canister, the loaded canister is transported to a designated SFP cell location using a canister-handling tool suspended from the SFP Bridge Crane. Handling of the loaded canister with the following analysis/design features will limit, to an acceptable level, the possibility of damage to more than 24 TPBARs during handling:

1. In accordance with NUREG-0612, NUREG-0554, and ANSI N 14.6, the SFP Bridge Crane and canister-lifting device will contain sufficient aspects of the single failure proof criteria to preclude a drop of the loaded canister as delineated below.

    - The SFP Bridge Crane is equivalent single failure proof with respect to structural integrity in accordance with NUREG-0612 and NUREG-0554 due to the following:

        a. Because the SFP Bridge Crane has a capacity of 4,000 pounds and the weight of the submerged loaded canister is approximately 700 pounds, the crane has safety factors twice the normally required values.
        b. The crane is equipped with redundant high hook limit switches of different designs to preclude structural failure.
        c. The single failure proof criteria pertain to the structural integrity aspects of the crane, and are satisfied, to an acceptable extent, because the loaded canister

weight (<700 pounds in water, < 1,000 pounds dry) is less than half of the rated capacity (4,000 pounds) for the crane, yielding greater than 10:1 safety factors. Together with the other design features as described in this section, provide sufficient aspects of the single failure proof criteria, for this lift, to preclude a handling event from damaging more than 24 TPBARs.

- The lifting tool is provided with a safety lanyard to limit canister descent in the SFP to such an extent that spilling of the TPBARs out of the open-topped canister is prevented, if the canister bottom were to hit an obstruction and cause the canister to tip. The lanyard is sized to stop the canister from a maximum hook speed of 40 feet per minute. Administrative requirements require that the safety lanyard be attached to the lifting tool when the canister is not engaged in an SFP rack cell, the consolidation fixture holster, or cask by at least 12 inches.

- In accordance with ANSI N14.6 sections for Critical Loads, the lifting tool is designed to twice the normal safety factors, tested to twice the normally required loads, and inspected utilizing required non-destructive examination (NDE) methods, thereby rendering it equivalent single failure proof It will also have an air actuated fail-closed safety latch to prevent the tool hook from disengaging from the canister lifting bail.

2. The loaded canister weight and its handling tool is less than that of a fuel assembly and its handling tool. Additionally, due to the design features listed above, the canister descent is limited to an uncontrolled lowering (e.g., a control failure) of a canister at a maximum hoist speed of 40 feet per minute, thereby limiting the kinetic energy to less than that of the fuel assembly. Therefore, fuel assembly drop accidents in the pool remain bounding.

3. An analysis has been performed to demonstrate that damage to more than 24 TPBARs contained in a canister is precluded for all credible impact scenarios during canister handling.

4. The drop of the light-weight, baseplate with TPBARs, within the spent fuel pool/cask load pit area, is bounded by the analysis of a fuel handling accident damaging an irradiated fuel assembly and 24 included TPBARs.

An analysis completed by PNNL has demonstrated that no TPBAR cladding failures are expected to occur during an impact event. The evaluation of TPBAR cladding stress is based on a canister loaded with TPBARs traveling at 40 feet per minute impacting onto a rigid surface. 40 feet per minute is based on a maximum uncontrolled lowering hook speed of the SFP hoist. TPBAR stresses resulting from feasible impact events (e.g. canister impact with a fuel rack, weir gate, pool wall, Consolidation Fixture) are bounded by the rigid surface impact evaluations. The canister and handling system design and configuration limit the impact forces on the canister.

Certain existing fuel storage cells will be designated as consolidation canister storage locations. These consolidation storage locations will be located on the outside row near one corner of the SFP away from any fuel movement path. At WBN, these locations will be restricted from any other use while the TPBARs are contained in the SFP. Fuel storage cells immediately adjacent to the designated canister storage locations may remain empty or may contain new fuel or spent fuel assemblies. In any event, fuel

movement procedure controls put into place because of the tritium program will prevent movement directly over the canisters.

TVA performed an evaluation comparing design features, operational controls, and analyses planned for implementation to those specified in the applicable section of NUREG-0612. This evaluation addressed each specified item separately by describing what is done for implementation and the basis for any difference in scope or depth relative to what is specified in NUREG-0612.

NUREG-0612 provides guidelines to assure that a Heavy Load drop (Heavy Load is defined as a load that weighs more than a single spent fuel assembly and its associated handling tool) would not result in a release of radioactive material that could result in off-site doses exceeding 10 CFR Part 100 limits. A heavy load at WBN is 2,059 pounds. Lifting the TPBAR canister loaded with up to 300 TPBARs is not a heavy load (calculated at approximately 750 pounds buoyant weight); therefore, it is not specifically addressed by NUREG-0612. However, in order to provide added assurance that the crane and lifting device used to lift the TPBAR canisters are safe, they will be evaluated against the requirements of NUREG-0612. The Spent Fuel Bridge Crane will be the only crane designated to lift the TPBAR canister while loaded with TPBARs. The bridge itself is designed specifically by Dwight Foote, Inc. for the provided hoist (4,000-pound capacity hoist).

In Section 9.1.4 of NUREG-0847, Supplement 22, on fuel handling system, NRC concluded (Reference 13):

> Based on the above, the staff concludes that the design and proposed operation of the WBN Unit 2 fuel handling system is acceptable. The descriptions of equipment and operating procedures used for the handling of fuel within the reactor, refueling canal, and shared spent fuel storage facilities included in Section 9.1.4 of Amendment 100 to the WBN Unit 2 FSAR were approved by the NRC staff in the SER. Also, the NRC staff accepted the WBN Unit 1 heavy load handling program based on conformance with the Phase I guidelines of NUREG-0612, as documented in SSER 13 to NUREG-0847, and TVA enhanced the WBN Unit 1 program through implementation of the NEI 08-05 guidelines. Therefore, implementation of a materially equivalent program at WBN Unit 2 and incorporation of the program information in the WBN Unit 2 FSAR is acceptable for fuel and heavy load handling activities associated with the operation of WBN Unit 2.

4.1.8 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 TPBAR Interface Issue 8 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, 10, 28, 29, and 30).

The design basis function of the Station Service Water System, which is called 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 SFP Decay Heat

TVA has prepared a quantitative analysis of expected spent fuel decay heat for both TPCs and non-TPCs. The analysis is based on comparative decay heat data for a base non-tritium core, a TPC with 80 fresh fuel assemblies (80-feed), and a TPC with 96 fresh fuel assemblies (96-feed). The results of the analysis show that the 80-feed case was limiting for decay heat (i.e., freshly offloaded core). The 80-feed TPC core contributes a slightly higher decay heat over the non-TPC and the 96-feed TPC, due to isotopic composition differences between the base and TPC cores, for the same design basis reactor power level. The results of the analysis show that the 96-feed case was limiting for residual SFP heat (i.e., heat coming from the total of previously discharged assemblies). TVA has assumed the worst-case combination of these two heat sources. The TVA analysis quantified the actual TPC impact on core heat loads at approximately 0.3 megawatts-thermal (MWt), which included both the decay heat generated by freshly discharged fuel assemblies during a refueling outage, and the additional residual decay heat from the increased discharge rate (96 per outage) of fuel assemblies into the pool. This value is based on conservative, full pool SFP conditions.

Increased SFP Cooling Heat Rejection on ERCW

The design basis analysis for the ERCW was evaluated for impact from the increased heat load from the CCS. The increased SFPCCS heat load rejection to the CCS will not result in a significant temperature increase in ERCW. The increase in decay heat associated with the TPC is approximately 1 million British thermal units per hour (MBtu/Hr). The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 14 MBtu/Hr. The proposed increase in decay heat above the approximate 1 MBtu/Hr associated with a TPC, is decay heat that is shifted from the RHR System to the SFPCCS. The shifting results from the fact that fuel is either in the core being cooled by the RHR System, or it is in the SFP being cooled by the SFPCCS. Because the decay heat has only shifted between systems, there is no net increase in CCS heat load on the ERCW system for this portion of the increased decay heat.

The design basis thermal analysis of record for the ERCW has sufficient margin to accommodate the increased CCS heat loads resulting from increased SFPCCS allowable decay heat loads. The increase in decay heat load is well within the design basis limiting heat load imposed on the ERCW during other modes of operation. Increased ERCW flows are the same higher flow rates that have been specified during other modes of operation. This small amount of increased decay heat and increased ERCW flow, when compared to the overall flow rates through the ERCW System, produces an insignificant increase in ERCW temperature (< 0.1 °F) leaving the plant site.

The additional heat load rejected to the ERCW from the CCS heat exchanger results in minimally elevated piping temperatures. The downstream dilution effect, however, minimizes the impact of the elevated ERCW temperatures, as all ERCW flows return to a common header prior to being discharged from the plant. The increased thermal loading on the piping analysis and support analysis of the ERCW System is well within existing design temperatures.

The increased decay heat associated with the TPC was previously considered for WBN Unit 2, as documented in Sections 9.1.3, 9.2.1, 9.2.2, and 9.2.5 of NUREG-0847, Supplement 23 (Reference 31).

ERCW Summary

The ERCW System has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. The ERCW system can also accommodate the additional SFP heat loads imposed by a previously approved change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities will not have an adverse impact on the ERCW heat removal capabilities.

For additional information on the SFPCCS, see Sections 4.1.11 and 4.1.12.

4.1.9 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 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, 28, 29, and 30).

The Tennessee River is the UHS for WBN. The purpose of the UHS is to provide a source of cooling water for decay heat removal. The WBN CCS removes heat from the SFP cooling and RHR systems and transfers it to the ERCW system. Heat in the ERCW system is then transferred to the UHS. During plant cooldown, additional heat from irradiated TPBARs in the SFP must be transferred to the UHS.

The design basis function of the UHS is to provide an uninterrupted source of cooling water for decay heat removal. The maximum allowable inlet temperature for the UHS is 85 °F. The ERCW System is utilized to supply water from the UHS 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 via its intermediate cooling loop providing a heat sink to the SFPCCS and RHR System.

<u>Tritium Impact on SFP Decay Heat</u>

See Section 4.1.8.

<u>Increased SFP Cooling Heat Rejection on UHS</u>

The design basis analysis for the UHS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant UHS temperature increase. The increase in decay heat associated with TPC is approximately 1 MBtu/Hr. The increase in allowable decay heat associated with reduced SFP heat exchanger fouling factors and lower CCS temperatures is approximately 14 MBtu/Hr. This total increase in decay heat load is well within the design basis limiting heat load imposed on the ERCW and UHS during other modes of operation. Increased ERCW flows are the same higher flow rates that have been specified during other modes of operation. This small amount of increased decay heat and increased ERCW flow, when compared to the overall flow rates of the UHS through the ERCW System, produces an insignificant increase (< 0.1 °F) in UHS temperature leaving the plant site. Because there is no significant increase in temperature, and because the ERCW has significant margin available, no changes to the ERCW temperature requirements are warranted.

The increased decay heat associated with the TPC was previously considered for WBN Unit 2, as documented in Sections 9.1.3, 9.2.1, 9.2.2, and 9.2.5 of NUREG-0847, Supplement 23 (Reference 31).

<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 tritium production activities. The UHS system can also accommodate the additional SFP heat loads imposed by the previously approved change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities at WBN Unit 2 will not have an adverse impact on the UHS heat removal capabilities. For additional information on the SFPCCS, see Section 4.1.11.

4.1.10 TPBAR Interface Issue 10: New and Spent Fuel Storage

NUREG-1672, Section 2.9.2, "The staff reviewed the effect of storing fuel assemblies with TPBAR assemblies in the new and spent fuel racks for the reference plant in accordance with SRP Section 9.1.1 for the new fuel storage and SRP Section 9.1.2 for the spent fuel storage. An analysis has previously been performed using the weight of 1470 pounds for a standard fuel assembly. The TPBARs, as burnable poisons, are similar inform to the Westinghouse standard burnable poison rod assemblies (BPRAs). Because certain space on the storage racks for fuel assemblies will be replaced by TPBAR assemblies, the combined weight of a fuel assembly with TPBARs was calculated to be less than 1430 pounds. DOE also analyzed the dynamic effects for the TPBAR assembly that rests on the top nozzle adapter plate of the fuel assembly and found that the dynamic effect is insignificant. Because the weight of a fuel assembly with TPBARs is less than the weight of the standard fuel assembly previously analyzed, the staff concludes that the current design of the new and spent fuel pool facilities is still valid for the racks containing TPBAR assemblies. However, because the fuel rack analysis is plant-specific, the staff agrees with DOE's conclusion that the specific storage configuration for a plant participating in DOE's program for the CLWR production of tritium should be analyzed and could require changes to the TS. 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 10 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, 10, 32, 33, 34, 35, 36).

<u>New Fuel Storage Vault</u>

TPBARs are a different type of poison than has been previously used at WBN Unit 2. However, the current New Fuel Storage Vault criticality analysis has shown that unpoisoned fuel assemblies (without either discrete or integral poison) containing nominal enrichments up to 5.0 weight percent ^{235}U can be stored in the fresh fuel rack array utilizing 120 specific cells of the 130 available storage locations. Fresh fuel containing TPBARs stored in the New Fuel Storage Vault will have a lower reactivity than unpoisoned fresh fuel assemblies. With respect to the characteristics modeled in the existing new fuel criticality analysis, the fuel assembly design does not change with the use of TPBARs. Therefore, the existing criticality analysis and New Fuel Storage Vault configuration remains conservative and valid when storing fuel assemblies containing TPBARs. The current New Fuel Storage Vault criticality analysis, approved in Amendment 15 to WBN Unit 1 license NPF-90 (Reference 36), is still bounding for fuel with TPBARs. This information is also applicable to the WBN Unit 2 TPC license amendment, because the New Fuel Storage Vault is common to both units.

<u>Spent Fuel Storage Pool</u>

TVA previously reanalyzed the criticality safety analysis for the installed (region 1) spent fuel storage racks to support TPBAR irradiation. This reanalysis was performed with fuel assemblies of nominal enrichments up to 5.0 weight percent ^{235}U containing TPBARs and addressed other neutron poisons including Wet Annular Burnable Absorbers (WABAs) and Integral Fuel Burnable Absorbers (IFBAs). The fuel was assumed to

operate with TPBARs or WABAs, which are removed at the time the assemblies were placed in storage. Credit was taken for IFBA and fuel burnup, where appropriate. The reanalysis demonstrated that sufficient conservatism was present in the previous analysis of the region 1 storage racks to account adequately for the effects of operating with TPBARs.

In Section 9.1.2 in NUREG-0847, Supplement 22, on spent fuel storage, NRC concluded (Reference 13):

> The spent fuel storage pit is a shared facility for Watts Bar Nuclear Plant (WBN) Units 1 and 2. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed Section 9.1.2 of Final Safety Analysis Report (FSAR) Amendment 95, dated November 24, 2009, and determined that there were no changes from the spent fuel storage pit design described in Section 9.1.2 of FSAR Amendment 92, dated December 18, 2008, which was previously reviewed by the staff. Based on the previous staff evaluation documented in NUREG-0847, "Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," issued June 1982 (hereafter referred to as NUREG-0847 or the SER), and its supplements, the review of the Watts Bar Unit 1 FSAR, and the staff evaluation of the submitted changes, the staff concludes that the spent fuel storage pit meets the relevant requirements of General Design Criterion (GDC) 2 ("Design Bases for Protection against Natural Phenomena"), GDC 4 ("Environmental and Dynamic Effects Design Bases"), GDC 5 ("Sharing of Structures, Systems, and Components"), GDC 61 ("Fuel Storage and Handling and Radioactivity Control"), 62 ("Prevention of Criticality in Fuel Storage and Handling"), and GDC 63 ("Monitoring Fuel and Waste Storage") of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," and the requirements of 10 CFR 50.68, "Criticality Accident Requirements." Therefore, the design of the spent fuel storage pit described in Section 9.1.2 remains acceptable.

NRC also noted the impacts from TPBARs:

> Subsequently, the NRC staff reviewed other issues related to fuel storage. In Supplemental Safety Evaluation Report (SSER) 5 to NUREG-0847, the staff noted that Tennessee Valley Authority (TVA) had contracted for the U.S. Department of Energy to receive spent fuel from WBN. In SSERs 15 and 16, the staff accepted a reduction in allowed spent fuel storage capacity to 484 assemblies because of concerns with the neutron absorber panels and other issues related to the construction of some of the fuel storage racks. By letter dated July 28, 1997 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML020780158), the NRC issued Amendment No. 6 to the WBN Unit 1 operating license, which authorized installation of new spent fuel storage racks and increased the spent fuel storage capacity to 1,610 assemblies. The NRC issued Amendment Nos. 37, 40, 48, 67, and 77 to the WBN Unit 1 operating license on February 21, 2002 (ML020580612), September 23, 2002 (ML022540925), October 8, 2003 (ML032880062),

January 18, 2008 (ML073520546), and May 4, 2009 (ML090920506), respectively.

These amendments authorized irradiation of tritium production burnable absorber rods (TPBARs) within the WBN Unit 1 core and transfer of these irradiated TPBARs through the shared WBN spent fuel pool. In addition, Amendment No. 40 to the WBN Unit 1 operating license authorized removal of smaller racks, which decreased the allowed storage capacity of the spent fuel storage racks to 1,386 assemblies in the remaining fuel storage racks.

The NRC staff reviewed the description of the spent fuel storage pit in Amendment 100 to the WBN Unit 2 FSAR and compared it with the description in Amendment 8 to the WBN Unit 1 FSAR. The staff found the descriptions to be essentially identical. Based on prior staff evaluation documented in NUREG-0847 and its supplements, the staffs review and acceptance of amendments to the WBN Unit 1 operating license, and the staffs comparison of the WBN Unit 1 FSAR with Amendment 100 to the WBN Unit 2 FSAR, the staff concluded that the spent fuel storage pool conforms to the relevant requirements of GDC 2, 4, 5, 61, and 63 for protection against natural phenomena, missiles, pipe break effects, radiation protection, and monitoring provisions. Therefore, the design of the shared spent fuel storage pool described in Section 9.1.2 of the WBN Unit 2 FSAR is acceptable.

Analyses were also performed to determine the limiting amount of water that can be displaced in order to checkerboard non-fissile bearing components with fresh fuel. It was conservatively determined that 75 percent of water can be safely displaced in empty cells by non-fissile bearing components. Because a fully loaded TPBAR storage canister containing 300 TPBARs displaces approximately 51 percent of the water in a storage cell, and the displacing material is a strong neutron poison, no additional restrictions are necessary on the location of a TPBAR canister in the SFP.

The SFP racks have been seismically qualified containing Consolidation Canisters loaded with up to 300 TPBARs. Based on a review of existing SFP rack structural analysis calculations, there are no restrictions regarding how many Tritium Rod Consolidation Canisters can be stored in a rack.

The heat produced by a TPBAR 30 days after reactor shutdown is approximately 3 watts and the total maximum heat load for each canister is approximately 900 watts. This heat load is considered negligible for an open topped thin-walled canister with drain holes at the base Natural circulation, with or without the drain holes, is deemed adequate to dissipate this small heat load. Additionally, drainage holes are on all four sides near the bottom and peripherally in the canister bottom plate. This configuration precludes significant natural circulation/drainage blockage from occurring.

Updated SFP Criticality Analysis

The criticality safety analysis of record for the WBN SFP was developed in 2001 and relied in part on analyses performed in 1996. The purpose of the updated analyses is to provide a complete up-to-date criticality safety evaluation for the WBN SFP based on the

latest methodologies consistent with current NRC guidance. The updated calculations were performed by Holtec International. The results of the analyses are summarized below.

The WBN SFP contains a single type of BORAL^{TM} racks with flux traps designed for storage of Pressurized Water Reactor (PWR) fuel. Criticality control in the WBN BORAL^{TM} storage racks rely on the following:

- Fixed neutron absorbers: BORAL^{TM} panels
- Storage cell spacing, i.e., flux traps between storage cells
- Soluble boron

The criticality calculations qualify the BORAL^{TM} storage racks uniformly loaded with fresh fuel assemblies with an initial enrichment up to 5 weight percent ^{235}U.

- Each design basis analysis calculation considers fresh fuel with a uniform enrichment. The same bounding enrichment is considered along the entire active length for each fuel pin. Lower enriched blankets are neglected. Therefore, there is no axial or radial variation in fuel along the entire active length.
- A bounding fuel density of all types of fuel assemblies is considered. This bounding approach provides analysis simplicity and margin.

<u>Acceptance Criteria</u>

The objective of this analysis is to ensure that the effective neutron multiplication factor (k_{eff}) of the SFP loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with unborated water, and does not exceed 0.95 for the pool flooded with borated water, all for 95 percent probability at a 95 percent confidence level.

<u>Applicable National Codes, Standards, and Regulations</u>

Codes, standards, and regulations or pertinent sections thereof that are applicable to the analysis include the following:

- 10 CFR Part 50, Appendix A, General Design Criterion 62, “Prevention of Criticality in Fuel Storage and Handling”
- 10 CFR 50.68, “Criticality Accident Requirements”
- NRC Standard Review Plan, NUREG-0800, Section 9.1.1, “Criticality Safety of Fresh and Spent Fuel Storage and Handling,” Revision 3
- NRC Memorandum from L. Kopp to T. Collins, August 19, 1998, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants” (ML072710248)
- ANSI ANS-8.17-1984, “Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors”
- NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology, January 2001.
- DSS-ISG-2010-01, Revision 0, “Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools” (ML110620086*)*

- NEI 12-16, Revision 1, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water-Reactor Power Plants” (ADAMS Accession Nos. ML14112A516)

Computer Codes Utilized and Their Benchmarking Status

Holtec International maintains an active list of QA validated computer codes that are approved for use in safety-significant projects.

The computer code MCNP5-1.51 is used for the criticality analyses. MCNP5 is a three-dimensional continuous energy Monte Carlo code developed at Los Alamos National Laboratory. This code offers the capability of performing full three-dimensional calculations for the loaded storage racks; it has a long history of successful use in fuel storage criticality analyses and has all of the necessary features for the analysis to be performed for the WBN SFP. MCNP5-1.51 calculations use continuous energy cross-section data predominantly based on ENDF/B-VII. The default ENDF/B-VII cross sections are adjusted for temperature dependence using the appropriate continuous energy cross-section data processed with the NJOY 99.396 code using the ENDF/B-VII library.

Benchmarking of MCNP5-1.51 for criticality calculations has been performed based on calculations for hundreds of critical experiments with fresh UO_2 fuel, fresh MOX fuel, and fuel with simulated actinide composition of spent fuel. The benchmarking area of applicability and the results of the benchmarking calculations for the hundreds of experiments were determined along with a trending analysis. The maximum bias and bias uncertainty associated with the benchmark subsets are applied to all analysis calculations to determine the maximum k_{eff}.

Approach and Major Assumptions to Ensure Conservative Results

The criticality analyses discussed in this summary were performed using the most recently NRC-accepted Holtec SFP criticality analysis methodologies. The calculations were performed using either the worst case bounding approach or the statistical analysis approach with respect to the various calculation parameters. Therefore, the analysis contains a large reactivity margin, which is also quantified and documented.

Specifically, to address regulatory concerns related to the BORAL™ neutron absorber:

- The minimum BORAL™ ^{10}B areal density, BORAL™ panel width, and BORAL™ panel thickness are used in all calculations because it is a well-known effect that minimum values of these parameters will increase reactivity.
- An uncertainty of 5 percent (a typical value) on the minimum areal density of the ^{10}B content in the BORAL™ is included in the uncertainty analysis to account for the measurement uncertainty associated with the ^{10}B content.
- All design basis calculations are analyzed by assuming a void with thickness of 0.09 inch in the gap between the BORAL™ and the steel sheathing to account for the reactivity effect of BORAL™ blistering. While blisters would only be expected locally, they are conservatively modeled over the entire length and width of all panels in the pool.

Results and Safety Findings

The results of this analysis show that the maximum k_{eff} of the PWR BORAL^{TM} racks of the WBN SFP loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.00 with no credit for soluble boron for normal conditions; and less than 0.95 with credit for soluble boron for both normal and accident conditions, all for 95 percent probability at a 95 percent confidence level.

The summary of the results for normal condition is contained in Table 4.1-23. The results meet the regulatory limits with margin.

Table 4.1-23: SFP Rack Criticality Analysis Results

Parameter	Value
Uncertainties	
Fuel Tolerance Uncertainty	0.0053
Rack Tolerance Uncertainty	0.0167
Eccentric Positioning Uncertainty	0
MCNP5-1.51 Calculation Statistics (95%/95%, 2σ)	0.0008
MCNP5-1.51 Code Bias Uncertainty	0.0078
Statistical Combination of Uncertainties	0.0192
Biases	
Fuel Eccentricity Bias	0
MCNP5-1.51 Code Bias	0.0007
Sum of Biases	0.0007
Total Correction Factor	
Total Correction Factor	0.0199
Determination of k_{eff}, Unborated Water	
Calculated MCNP5-1.51 k_{calc}	0.9759
Maximum k_{eff}	**0.9958**
Regulatory Limit	1.0000
Determination of k_{eff}, 500 parts per million (ppm) Borated Water	
Calculated MCNP5-1.51 k_{calc}	0.9237
Maximum k_{eff}	**0.9436**
Regulatory Limit	0.9500

For accident conditions, a soluble boron level of 500 ppm is sufficient to ensure that the maximum k_{eff} is below the regulatory limit. The analysis reserves 50 ppm of soluble boron to offset the reactivity impact of the fuel assembly grids (as specified in Section 5.1.1 of Reference 61). The boron dilution analysis assumes an initial boron concentration of 2,300 ppm for the limiting evaluation.

The analyses also show that normal fuel movement in the SFP and replacing any cells that contain fuel assemblies with empty water cells or non-fuel hardware are acceptable.

Neutron Absorber Monitoring Program

Neutron absorbing materials installed in SFP storage racks ensure that the effective neutron multiplication factor (k_{eff}) does not exceed the values and assumptions used in the criticality analysis. Degradation or deformation of the credited neutron absorbing materials may reduce safety margin and potentially challenge the subcriticality

requirement. A Neutron Absorber Monitoring Program, following the NRC- approved guidance in NEI 16-03-A, “Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools,” (References 62 and 63) will be implemented at WBN. The Neutron Absorber Monitoring Program ensures compliance with 10 CFR Part 50, Appendix A, GDC 61, “Fuel storage and handling and radioactivity control,” by providing appropriate periodic inspection and testing of spent fuel rack components important to safety.

4.1.11 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 2 Tritium Production Program LAR is based on the applicable WBN Unit 1precedent documents (References 1, 9, 10, 28, 29, and 30).

The SFPCCS for WBN Units 1 and 2 is sized to handle full core offloads. In the 1996-97 timeframe, WBN underwent spent fuel storage rack additions, which included development of a new thermal hydraulic analysis based on standard NRC-approved methodologies that are scenario based. During the rerack design change, TVA recognized the impracticality of following a scenario-based set of limits during plant operation for predicting SFP decay heat load. During the licensing efforts associated with the rerack efforts at WBN, the UFSAR was revised to capture a limiting value of decay heat that could be placed in the SFP, based on outage-specific decay heat analysis performed for each outage. This approach provided a more realistic means (based on quantitative limits instead of a scenario-based limits) of assuring compliance with the maximum allowable design basis decay heat loads that could be placed in the SFP at any time. Compliance with these limiting values provides assurance that maximum analyzed temperatures of the SFP and attendant decay heat removal system piping will not be exceeded should a train of SFPCCS fail.

UFSAR Section 9.1.3 allows outage-specific decay heat values to be used to determine the acceptable point in time that core offloading activities may commence without exceeding the design basis maximum allowable heat load. Prior to each outage, a

core-specific and real time SFP decay heat assessment is prepared, which considers core operating parameters such as average fuel burn-up, interim trips, and coast-downs, to develop pre-outage data for expected core and SFP decay heat. Procedures are in place to assure that at no time during core offloading activities will the design basis limits of the SFPCCS be exceeded. Adherence to the established limiting values of allowable SFPCCS decay heat ensures that the maximum SFP temperature does not exceed the pre-established maximum allowable design temperatures.

Section 9.1.3 in NUREG-0847, Supplement 23 states in part (Reference 31):

> WBN Unit 1 License Amendment No. 37 authorized a revised SFP cooling analysis methodology for WBN that resulted in an increased allowable maximum heat load from 32.6 to 47.4 million British Thermal Units per hour (Btu/hr). This revised methodology credited additional heat removal capability resulting from lower-than-design component cooling water temperature and heat exchanger fouling at the time of the fuel transfer. The improved heat removal capability allowed a decrease in the minimum decay time necessary to maintain the peak SFP temperature below the design temperature of 159.24 °F with one SFPCCS train in operation. The staff's safety evaluation for this amendment concluded that the SFPCCS had adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities.

Operation of WBN Unit 2 results in a further increase in the potential peak heat load above that authorized by WBN Unit 1 License Amendment No. 37 (Reference 30). The increase results from the lower average decay time for past outages when two reactors, rather than only one reactor, discharge to a shared SFP.

TVA also modified the WBN UFSAR to:

- increase the maximum allowed SFP heat load from 47.4 million to 50.21 million Btu/hr for below-design cooling water temperature and heat exchanger fouling conditions
- revise the expected water heat-up rates and boil-off times listed in UFSAR Table 9.1-1 for a total loss of cooling capability accident for the full core discharge, the full core discharge following a normal refueling, and the maximum allowed heat load cases

These changes were accommodated by delaying refueling fuel transfers to the SFP to compensate for the additional heat load resulting from more frequent discharges resulting from the proposed operation of both WBN Unit 1 and Unit 2. These changes were submitted to NRC by TVA letter dated December 21, 2010 (Reference 37).

The NRC review of the updated SFP cooling analysis was documented in Section 9.1.3 in Supplements 23 and 26 to NUREG-0847 (References 31 and 38).

> The staff reviewed the changes proposed by TVA to the WBN Unit 2 FSAR in its letter dated December 21, 2010 [ML103610285], and compared the changes to the SFP cooling acceptance criteria applied to WBN Unit 1 and the FSAR content requirements of 10 CFR 50.34. The

staff found that the design of the SFPCCS is unchanged and remains acceptable, consistent with the conclusions of the staff as documented in the SER and its supplements. Based on its review, the staff concluded that TVA demonstrated that the cooling capability of the existing SFPCCS was adequate for the increased heat load imposed by alternating fuel discharges from WBN Units 1 and 2 under normal operating conditions, as required by GDC 44 and 61. The staff concludes that the proposed description of the design and operation of the spent fuel pool cooling and cleanup system in FSAR Section 9.1.3 adequately supports operation of WBN Unit 2 and is consistent with the requirements of 10 CFR 50.34, and is, therefore, acceptable.

<u>SFPCCS Summary</u>

The SFPCCS has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. The ERCW system can also accommodate the additional SFP heat loads imposed by the previously approved change to allow commencement of core off-loads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities will not have an adverse impact on the SFPCCS heat removal capabilities.

For additional information on the ERCW, UHS, and CCS systems, see Sections 4.1.8, 4.1.9, and 4.1.12.

4.1.12 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 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 9, 10, 28, 29, and 30).

The design basis functions of the CCS include providing an intermediate cooling loop for heat removal from several safety-related radioactive system heat exchangers, as well as several non-safety related components. Two of the highest heat loads placed on the CCS include the SFPCCS and the RHR System. These two decay heat removal

systems are the primary means for cooling the plant and removing residual decay heat during later stages of plant cooldown and during outages.

Tritium Impact on SFP Decay Heat

TVA has prepared a quantitative analysis of expected spent fuel decay heat for both TPCs and non-TPCs. The analysis is based on comparative decay heat data for a base non-tritium core, a TPC with 80 fresh fuel assembles (80-feed), and a TPC with 96 fresh fuel assemblies (96-feed). The results of the analysis show that the 80-feed case was limiting for decay heat. The 80-feed TPC core contributes a slightly higher decay heat over the non-TPC and the 96-feed TPC, due to isotopic composition differences between the base and TPC cores, for the same design basis reactor power level. The results of the analysis show that the 96-feed case was limiting for residual heat. The TVA analysis quantified the actual TPC impact on core heat loads at approximately 0.3 MWt (approximately 1 MBtu/hr), which included both the decay heat generated by freshly discharged fuel assemblies during a refueling outage, and the additional residual decay heat from the increased discharge rate (96 per outage) of fuel assemblies into the pool. This value is based on conservative, full pool SFP conditions for single unit operation.

The production of tritium at WBN results in both higher fuel decay heat loads during the outage as well as higher residual SFP heat loads remaining in the pool, which affect future outages. The purpose of this analysis was to examine the decay heat loads for two tritium assembly feed cases, an 80-feed Tritium Core, and a 96-feed Tritium Core, and compare them to the decay heat loads for a normal (Base Case) Core heat load.

The analysis assumed that the use of a specific outage decay heat curve is acceptable. The technical justification for this assumption is based on the fact that the intent of the analysis was to determine relative impacts of TPC on the plant. Because relative values were being developed, the use of outage specific decay heat data combined with conservative rounding of results provided bounding results.

Heat Generation

The SFP decay heat generation values consist of heat from both the current core and residual (old) fuel that is in the SFP. The core decay heat generation values were taken from nuclear fuels data 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, Regulatory Guide 3.54, and NUREG/CR-2397. All data utilized in the analysis was based on results from DHEAT-generated data sets for a Base (existing), 80-feed, and 96-feed cores.

Projected Core Decay Heat Impact

Per Technical Specification (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 at approximately 10 days after core shutdown. The period from 100 hours, Day 4, to Day 10, represents the period in which 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 4 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 4.1-24 were determined, after averaging the Day 4 and Day 10 results.

Table 4.1-24: 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

Projected SFP Residual Heat Impact

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 was expected because a 96-feed core requires more fuel assemblies to be placed in the SFP each outage.

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 decay heat values for both a typical tritium production 80-feed assembly core (for Unit 2 discharges) and a 96-feed assembly core (for Unit 1 discharges). These values are conservative because WBN Unit 1 is only authorized to place a maximum of 1,792 TPBARs into the reactor in an operating cycle, which corresponds to a less than an 80-feed assembly core. Similarly, in this LAR, TVA is requesting to use a maximum of 1,792 TPBARs in WBN Unit 2.

For the analysis of core offload time for dual unit operations, the fourteenth 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 7 was a full core offload). 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 because the use of a 96-feed assembly core for Unit 1 discharges would cause the SFP to fill up faster, allowing for one less core offload, thus causing the SFP in this analysis to be filled with younger fuel which increases the decay heat.

Note that for single unit operation, assuming a 96-feed core would result in a three-cycle difference before reaching full pool conditions, the effect of tritium on the SFP was determined to be an increase in residual heat of 0.1526 MWt.

Net SFP Decay Heat Impact Related to Tritium Production Activities

The net SFP decay heat impact for single unit operations related to tritium production activities was obtained by adding the tritium impacts on core decay heat (both 80-feed and 96-feed TPBAR Cases) and the limiting 96-feed Case value for the SFP residual decay heat:

TPBAR 96-feed Case	$\rightarrow$	0.1149 MWt + 0.1526 MWt = 0.2675 MWt
TPBAR 80-feed Case	$\rightarrow$	0.1936 MWt + 0.1526 MWt = 0.3462 MWt

Note that the overall residual heat in the SFP is increased for dual unit operation, but the net impact from Tritium production for a second TPC would be less pronounced because there would be no difference in the number of cycles to fill the SFP (when comparing the 80-feed, Base Non-TPBAR Case, with the 80-feed TPBAR Case). Therefore, the net TPC impact on core heat loads of approximately 0.3 MWt remains an acceptable approximation for the impact of Unit 2 operation as a TPC.

Results / Conclusions

Based on the analysis, it was shown that tritium production activities would have an impact on SFP decay heat loads. Due to this impact, the critical path time related to required hold time prior to offloading the core would also be affected. The overall conclusion of the analysis is that Tritium production activities at WBN will have a small but measurable negative impact (i.e., an increase) on SFP decay heat. The increase in SFP decay heat may impact outage critical path time, due to a delay in commencement of offload activities.

Increased SFP Cooling Heat Rejection on CCS

The design basis analysis for the CCS was evaluated for impact by the increased heat load from the SFPCCS. The increased SFPCCS heat load will not result in any significant temperature increase on CCS. The increase in decay heat associated with TPC is approximately 1 MBtu/Hr. This decay heat load increase is approximately 1 percent of the total design heat load on the CCS.

CCS design thermal analyses have been evaluated and determined to be capable of accepting the increased SFPCCS allowable decay heat loads. CCS flows to the SFPCCS heat exchangers will not be increased. The additional heat load rejected to the CCS from the SFPCCS heat exchanger results in slightly elevated CCS temperatures, but is within existing design basis values. Piping analysis and support analysis of the CCS have been previously analyzed at a higher ultimate temperature associated with more bounding operational modes, and are not affected by the increased CCS heat load. The downstream dilution effect also helps to minimize the impact of the elevated CCS temperatures, because as SFPCCS heat loads increase, the RHR System heat loads decrease. With all CCS flows returning to a common header prior to returning to the CCS/ERCW heat exchangers, there is no measurable change to the mixed stream CCS temperature.

Because higher allowable SFP decay heat can be placed in the SFP if CCS temperatures and/or SFP heat exchanger fouling factors are shown to be less than design conditions, maintaining the CCS temperature during outages to as low as

possible is desired. CCS temperatures can be lowered considerably if ERCW flows to the CCS heat exchangers are increased. Increased ERCW flow rates are within existing flow criteria established for other modes of operations.

The CCS has adequate capacity and cooling margin to perform its safety and non-safety functions with the additional heat loads imposed by tritium production activities. The CCS can also accommodate the additional SFP heat loads imposed by the change to allow commencement of core offloads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature. Tritium production activities will not have an adverse impact on the CCS heat removal capabilities. Additional information on SFP decay heat is provided in Section 4.1.11.

For the purpose of this discussion, the UHS is considered synonymous with ERCW. Note that UHS also provides heat removal for other non-safety related systems, including Raw Cooling Water (RCW), and the Supplemental Condenser Circulating Water (SCCW). Safety related system heat loads on the UHS are the only impacts related to operation with TPCs; therefore, the UHS discussion below has been combined with ERCW. (The ERCW discharges into the Cooling Tower Basin; however, the impact on the SCCW system is negligible.) The use of "nominal" values is used, because there are some variations in heat loads between independent trains of cooling in the CCS and ERCW systems.

Operation of WBN Unit 2 as a TPC will increase the SFP heat load by approximately 1 MBtu/hr. This heat is ultimately rejected to the environment by the ERCW system. There is no absolute limit on the ERCW heat load. The TS maximum ERCW (river water) temperature of 85 °F is the maximum cooling water temperature for the components cooled directly by ERCW. Minimum required ERCW flow rates are established to transfer the component design heat loads at the maximum cooling water supply temperature. The ERCW current maximum heat load is on the order of 300 MBtu/hr. Adding 1 MBtu/hr to the current load has a negligible impact on the ERCW discharge temperature.

The SFPCCS heat exchangers are cooled by the CCS. The CCS heat exchangers are cooled by the ERCW and the CCS heat exchanger outlet temperature is limited to less than 110°F by the piping thermal analysis, but typically is maintained less than 95°F. For dual unit operation, the CCS heat exchanger duty is approximately 97 MBtu/hr during Unit 2 refueling and Unit 1 LOCA Recirculation, with a CCS temperature difference of 25°F. Adding 1 MBtu/hr to the current load changes the CCS outlet temperature less than 1°F.

For dual unit operation with TPCs, the SFP heat load under non-refueling conditions is 21 MBtu/hr. The CCS cooling water design temperature rise in the SFP heat exchanger is 8°F for the design duty of 11.94 MBtu/hr. Adding 1 MBtu/hr to the current load without increasing CCS flow increases the CCS outlet temperature less than 1 °F.

The peak heat loads for the proposed change include the additional heat loads that CCS or ERCW/UHS would see based on the combined effect of TPCs and reduced fouling of the SFPCCS heat exchangers. All existing design values and analyses are based on maximum CCS and ERCW temperatures, as these maximum temperatures result in maximum piping temperatures used in piping/support analyses. This approach is

acceptable, because higher heat loads can only be achieved by lower CCS temperatures, which are achieved by lower ERCW temperatures and assuring that the final analyses for piping and support thermal analyses remain bounding because they have been based on maximum temperatures.

As an example, the maximum allowable decay heat load that can be placed in the SFP is 50.2 MBtu/hr. However, the maximum allowable heat load that will be rejected to CCS was determined to be 42 MBtu/hr (with zero fouling and design CCS temperature of 95°F). The difference between 50.2 and 42 MBtu/hr is the additional heat load that can only be allowed based on sub-design CCS temperatures. While the actual heat load rejected from the SFP to CCS and ERCW at 50.2 MBtu/hr and 80 °F CCS will be greater than the heat load at 42 MBtu/hr and 95°F CCS, the resulting piping temperatures and related analyses are maximized and bounding at the 42 MBtu/hr and 95°F CCS temperature design points.

The effects on SFP heat load impacts from the TPC and the methodology using lower CCS temperatures and credit for reduced SFP heat exchanger fouling have not been independently determined. The reason for this approach is that the TPC impacts are very low (nominal 1 MBtu/hr). Because existing UFSAR statements allow the use of analysis-based inputs to determine commencement of offload time, compensation for higher heat load impacts from the TPC alone could have been achieved by delaying the commencement of core offloads by 19 to 24 hours (for single unit operation), which would have resulted in no net impact on CCS or ERCW/UHS heat loads. Note that with two units discharging into the SFP, the delay may be longer for dual unit operation because there will be additional residual heat in the SFP. Therefore, the above maximum values, consistent with supporting analyses, are based on the existing methodology of combining the heat load effects of the TPC with the credit for reduced CCS temperatures and reduced fouling of the SFPCCS heat exchangers.

The increased decay heat associated with the TPC was previously considered for WBN Unit 2, as documented in Sections 9.1.3, 9.2.1, 9.2.2, and 9.2.5 of NUREG-0847, Supplement 23 (Reference 31).

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 tritium production activities for Unit 2. Existing SFPCCS operational parameters can accommodate tritium production operations by delaying the start of offloading the core until design allowable heat loads can accommodate core and residual decay heat. The CCS system can also accommodate the additional SFP heat loads imposed by the change to allow commencement of core offloads as early as 100 hours, consistent with other design guidance regarding SFP heat exchanger fouling and CCS temperature.

Tritium production activities will not have an adverse impact on the CCS heat removal capabilities.

4.1.13 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 ' 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 Unit 2 Tritium Production Program LAR is based on the relevant WBN Unit 1 precedent documents (References 1, 2, 9, 17, 23, 39, 40, and 41).

NUREG-1672 (Reference 8) and the DOE TPC Topical Report (Reference 7) Section 2.9.5 addressed possible impacts on the Demineralized Water Makeup System (DWMS). Section 2.9.5 acknowledged that tritium production activities would result in increased tritium levels in the RCS. To maintain tritium levels within the RCS at current levels, additional feed and bleed operations may be required. Any increase in feed and bleed operation requires additional demineralized water as makeup. NUREG-1672 required the specific impact on DWMS from increased feed and bleed demand be evaluated.

The DOE TPC Topical Report addresses the Potable and Sanitary Water Systems as follows:

> SRP 9.2.4
>
> Potable and Sanitary Water Systems: The acceptance criteria in this section are based on the relevant requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 60. They deal with the design provisions provided to control the release of liquid effluents containing radioactive material from contaminating the Potable and Sanitary Water System (PSWS). The design of the PSWS is not being modified, therefore, the design features which prevented contamination of the PSWS in the reference plant (i.e., no cross-connection between the PSWS and any potentially radioactive system and the use of backflow prevention devices where plumbing fixtures are located in areas susceptible to potential radiological hazard) are still present for the TPC plant. Therefore, there is no impact on this system.

In NUREG 1672, the NRC made the following decision regarding the DOE TPC Topical Report:

> <u>Potable and Sanitary Water Systems (SRP Section 9.2.4)</u>
>
> DOE evaluated the effect of TPBARs on the Potable and sanitary water systems for the reference plant against the guidance of SRP

Section 9.2.4 and concludes that there is no effect on this system. The staff agrees with this evaluation.

WBN UFSAR Section 9.2.4 describes the plant features for the Potable and Sanitary Water Systems. The potable water system is not cross connected with any radioactive system. The NRC review of the Potable and Sanitary Water Systems is documented in NUREG-0847 (Reference 25).

These conclusions are consistent with the tritium program operational experience at WBN Unit 1 and remain valid for the proposed tritium production operation for WBN Unit 2.

For systems that are considered as nonradioactive, but could possibly become radioactive through interfaces with radioactive systems, Watts Bar Chemistry Manual, Chapter 3.01, "System Chemistry Specifications," establishes the routine sampling/analysis or monitoring program summarized in Table 4.1-25.

Table 4.1-25: Tritium Sampling/Analysis and Monitoring Program

System	Frequency Tritium Checked
Primary Water Storage Tank	Quarterly
Component Cooling Water System	Per request (Note 1)
Feedwater	Weekly
Demineralized Water Head Tank	Monthly
Condensate Storage Tanks	Per request (Note 1)
Auxiliary Boiler	Monthly during operation
Raw Cooling Water	Quarterly
Yard Holding Pond	Per request (Note 1)
Low Volume Waste Treatment Pond	Per request (Note 1)
Potable Water System	Monthly

Note 1: Associated with maintenance or troubleshooting

TVA does not intend to make changes to the current feed and bleed operations used to control boron concentration in the RCS. With routine boron control and 1,792 TPBARs at a permeation of 5 Ci/TPBAR/year, the RCS average tritium value will be approximately 12 μCi/gm. This value is due to normal reactor tritium production plus the tritium permeation from TPBARs. Public doses from liquid and airborne effluent release will remain below applicable Offsite Dose Calculation Manual (ODCM) limits, and tritium release concentrations will remain within 10 CFR Part 20 and ODCM release limits.

Within the WBN DWMS, there exists sufficient surge capacity as well as production capacity to meet these projected needs. As tritium levels increase in the RCS, ample planning time will be available to assure adequate surge volume is available and production rates are capable of meeting demand.

WBN uses vendor-supplied equipment to produce high purity water for use in the site DWMS. Following an upgrade to the DWMS in 2015 to support Unit 2 operation, the production rate at the WBN DWMS is in the nominal 400 gallons per minute range. Storage of demineralized water exceeds 500,000 gallons in available tanks.

TVA's review of the DWMS for WBN has determined that the current system's storage and water production capacity, compared to the expected increase in feed and bleed required to mitigate a two TPBAR failure event, is adequate. Public doses from liquid and airborne effluent release will remain below applicable ODCM limits, and tritium release concentrations will remain within 10 CFR Part 20 and ODCM release limits. See TPBAR Interface Item 14 for more information concerning Liquid Waste Management.

The DWMS and storage tanks will not require modification, nor will the water supply contract require changes to support tritium production activities at WBN Unit 2.

4.1.14 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 Watts Bar Unit 2 Tritium Production Program LAR is based on the relevant Watts Bar Unit 1 precedent documents (References 1, 2, 9, 17, 21, 23, 39, 40, 41, and 42):

TVA previously requested authorization to irradiate not more than 1,792 TPBARs in WBN Unit 1 (References 17, 23, 40, 41, and 42). TVA's application for that amendment provided radiological analyses based on 1,792 TPBARs and a functional requirement of 5 Ci/TPBAR/year tritium permeation. NRC approved WBN Unit 1 License Amendment 107, authorizing WBN Unit 1 to irradiate up to a maximum of 1,792 TPBARs (Reference 2).

This section builds on the previously approved WBN Unit 1 analysis to include a TPC for Unit 2 with a maximum of 1,792 TPBARs. The same permeation rate of 5/Ci/TPBAR/year is considered. This section addresses the two items discussed in the interface issue:

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

2) 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

Evaluation of potential leaching of chemical contaminants from TPBARs has determined that the effect of these potential chemical contaminant releases into the RCS or the SFP will not require any changes to the existing sampling frequencies. However, procedures were revised prior to TPBAR irradiation to require liquid sampling in the SFP for tritium while moving and storing irradiated TPBARS. While irradiated TPBARs are stored in the SFP, liquid tritium sampling is conducted on a weekly basis. When moving irradiated TPBARs, the SFP is sampled daily. TVA will also monitor for airborne tritium in the SFP area using a portable tritium air monitor when fuel containing irradiated TPBARs is being moved or while consolidating irradiated TPBARs. TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

The DOE Topical Report (Reference 7) and NUREG-1672 (Reference 8) concluded that there would be a negligible increase in the annual worker radiological exposure due to operation with TPBARs. This conclusion was based on the assumption that the reference plant would maintain the tritium primary coolant activity within the control value. However, TVA did not assume that the tritium primary coolant activity would be maintained at this control value in the LAR that formed the basis for License Amendments 40 (Reference 9) or 107 (Reference 17) and instead determined the effect on occupational dose assuming design basis levels. The WBN Unit 1 License Amendment 40 (Reference 1) RCS tritium fixed action levels of 9 µCi/gm and 15 µCi/gm were based on a cycle inventory of 2,304 TPBARs and breaker-to-breaker runs. The fixed action levels were insensitive to variations in the number of TPBARs and RCS water balance.

TVA has established more effective performance metrics with two tritium-based action levels to continually monitor TPBAR performance. These action levels are cycle specific and are based on the difference between the total calculated tritium released to the RCS (i.e., current RCS inventory plus removed via letdown) from all sources minus the estimated tritium released to the RCS from the traditional non-TPBAR sources (e.g., boron, lithium, fuel rods, control rods, secondary source rods, etc.), which yields the net estimated TPBAR tritium.

Action Level (AL) 1 is triggered when the net cumulative estimated TPBAR tritium exceeds 1.5 times the TPC estimated value. AL 2 is triggered when the net cumulative estimated TPBAR tritium exceeds 3 times the TPC estimated value. The AL 1 value of 1.5 is approximately the 95 percent confidence level of the total uncertainty in the net estimated TPBAR tritium value. That is, if exceeded, there is a 5 percent probability that the estimated value is consistent with expected TPBAR permeation performance. The TPC estimated value is at a specific time in the cycle dependent calculated value. The tritium attributed to TPBARs is determined by subtracting the expected tritium value established by measurements taken in cycles without TPBARs from the total tritium measured in the RCS with TPBARs. The estimated value is established prior to each cycle and is based on the number of TPBARs to be irradiated during the cycle and

observed previous TPBAR permeation performance. For a specific fuel cycle Effective Full Power Day, the AL Trigger is calculated as:

ALTrigger = (Total RCS Inventory – non-TPBAR Sources) / TPC Estimated Value

The use of the cycle-specific TPC estimated value as the permeation performance metric compensates for RCS water balance (i.e., water makeup and letdown) and the cycle-specific reactor power history. The lower AL requires more frequent tritium system sampling to monitor, verify, track, and trend the tritium levels. In the unlikely event that the higher AL is exceeded, TVA will take further action to minimize the onsite and offsite radiological impacts of abnormal RCS tritium levels. These actions may include, but not be limited to, procedural and administrative measures that will serve to:

- ensure that the core is operated consistent with design objectives
- act as a trigger for increased data monitoring, tracking and trending
- provide a catalyst to prompt appropriate state, federal, contractual, and regulatory notifications
- initiate appropriate recovery and restoration actions
- aid in the development of appropriate actions for minimizing the impact of unexpected tritium production increases on:
    - worker dose
    - dose to members of the public
    - the potential uncontrolled release of radioactive material
    - low level waste

Specific actions and evaluations are contained within a WBN Technical Instruction.

TVA will review and modify actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

Monitoring TPBAR Estimated Permeation Performance

When taking measurements of RCS tritium levels, it is not possible to differentiate between tritium from TPBARs and tritium from other core components and RCS sources, therefore the tritium attributed to TPBARs is determined by subtracting the expected tritium value established by measurements taken in cycles without TPBARs from the total tritium estimated in the RCS with TPBARs.

The cumulative TPBAR tritium release at any point in the cycle is calculated as the difference of two larger quantities as described below:

(1) The total (calculated) cumulative tritium to-date, T_{Total}, including that which is currently in RCS (daily measurement) plus the sum of the (estimated) tritium removed from the RCS to-date via letdown to compensate for water/boric acid injections. This total includes the tritium released from the TPBARs plus tritium from non-TPBAR sources in the RCS.

$$T_{Total}(t) = T_{Current\ RCS\ Inventory}(t) + T_{Removed\ Letdown}(t)$$

(2) The projected cumulative tritium that would have accrued to-date, in the absence of TPBARs ($T_{non\text{-}TPBAR}$) from sources including production from soluble boron and lithium, and permeation into the RCS from fuel rods, burnable absorber rods, secondary source rods, and control rods, all of which produce tritium in their internal components. Thus,

T_{TPBAR} (t) = T_{Total} (t) – $T_{non\text{-}TPBAR}$ (t), and

$T_{non\text{-}TPBAR}$ (t) = $T_{Soluble\ Boron}$ (t) + $T_{Lithium}$ (t) + $T_{Fuel\ Rods}$ (t) + $T_{Burnable\ Absorber}$s (t) + $T_{Control\ Rods}$ (t) + $T_{Secondary\ Source\ Rods}$ (t)

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 Production Cycles 6 through 14 with a 90 percent uncertainty is shown in Table 4.1-26.

Table 4.1-26: Estimated TPBAR Permeation for WBN Cycles 6 through 14

Cycle Number	**Cycle Length (EFPD)**	**End of Cycle (Ci/TPBAR)**	**Last 365 Days (Ci/TPBAR/year)**
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

Tritium Bioassay Program

TVA revised procedure RCI-137, "Radiation Protection Tritium Control Program," Table 3.1, "Tritium Action Levels," to incorporate the 0.01 curie per kilogram (Ci/kg) criteria from NRC Regulatory Guide (RG) 8.32, "Criteria for Establishing a Tritium Bioassay Program." The associated 0.01 µCi/ml action level specified by RCI-137, Table 3.1 state, "bioassays should be initiated whenever employees are exposed to the air in a room or area where more than 10 kilograms of water containing this or greater concentration." RCI-137, Table 3.1 provides guidance as follows.

TRITIUM ACTION LEVELS					
Process Tritium Concentration (µCi/ml)	**DAC, DAC-hours**	**Mode of Exposure**	**Tritium Survey Requirements**	**Action**	**Basis for Bioassay (Regulatory guidance and TVA Procedure Requirements)**
≥ 0.01	N/A	direct contact	measurement of process water	Urinalysis following skin contact, ingestion, or absorption through cuts or abrasions. Diving requires routine bioassays as specified in Note 1.	US NRC Regulatory Guide 8.32 RCDP- 7, Bioassay and Internal Dose Program
≥ 10.0		inhalation	measurement of process water and tritium air samples	Urinalysis following exposure to air in a room whenever employees are exposed to greater than 10 kg of water containing 0.01 Ci/kg or when water containing a total of more than 0.1 Ci of tritium is in contact with air (such as a fuel pool).	US NRC Regulatory Guide 8.32
	≥ 0.3 DAC	inhalation	tritium air samples	Urinalysis recommended, see Note 2	RCDP-7
	≥ 4 DAC-hours in 7 consecutive days	inhalation	tritium air samples with DAC-hour tracking	Urinalysis shall be requested for any employee who exceeds this limit	RCDP-7. Basis is 10 mrem/week, which is easily detected and verified by bioassay.

Note 1 For underwater diving operations in tritiated water exceeding 0.01 µCi/ml, RCDP-7 Bioassay and Internal Dose Program specifies that collection and analysis of urine samples is recommended for each diver:
(a) prior to the first on-site dive
(b) within 24 hours following the completion of the initial dive
(c) once each week while diving operations are in progress
(d) upon completion of diving operations
(e) whenever diving suit leakage results in skin contact with tritiated water

Note 2 For work activities where workers are known or may be exposed to tritium atmospheres exceeding 0.3 DAC or other site project criteria, the collection and analysis of urine is recommended as detailed:
(a) pre-job should be performed to establish a baseline value
(b) within 24 hours following the completion of the first exposure
(c) weekly to ten days for the duration of the work involving tritium exposure
(d) upon completion of the work involving tritium exposure

TVA procedure RCI-137 along with TVA procedures 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. TVA's program establishes criteria for performing in vitro bioassay; (1) based on process water concentrations using the guidance in RCI-137, Table 3.1, (2) when an individual worker's exposure is greater than or equal to four DAC-hours in seven consecutive days, and (3) for tritium skin contamination with tritiated water concentrations exceeding 0.01 µCi/ml. Tritium DAC-hour tracking is initiated whenever a worker is in an airborne area, exposed to concentrations of 0.3 DAC or greater. TVA's procedures cover types of bioassay, selection of individuals for bioassay, bioassay collection, sample volume, sample storage, packaging and shipping, detection limits, and internal dose calculation methods. TVA's vendor contract for analysis requires a Minimum Detectable Activity (MDA) that meets the limit of 0.3 µCi/L specified in ANSI N13.14. The frequency of bioassays is determined based on the work, the exposure scenario, and trigger levels as described in RCDP-7. Baseline tritium bioassays are obtained for all divers prior to the first on-site dive. Baseline (pre-job) tritium bioassays are also recommended for work activities where workers are known or may be exposed to tritium atmospheres exceeding the trigger levels of greater than 0.3 DAC. All data (i.e., bioassay, air samples with DAC-hour tracking, and whole body counts) are reviewed and evaluated to arrive at the best estimate of the worker's intake and CEDE for each radionuclide. In some cases, a single bioassay is enough to evaluate the exposure; in other cases, multiple bioassays are obtained for work in tritium airborne conditions that may continue for days or weeks.

Sample collection guidance in RCDP-7 for tritium includes collecting urine samples no sooner than two hours following the tritium exposure event to allow the activity to equilibrate, and no later than 72 hours following an acute exposure if possible. The first voiding of the bladder following the exposure is not used for the urinalysis. Sample collection instructions are provided to the worker with the bioassay container. This guidance in RCDP-7 is consistent with the guidance from RG 8.32.

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

<u>Tritium Source Term Definition and Discussion</u>

Following the review guidance in Chapter 11, "Source Terms," in NUREG-800 Standard Review Plan, TVA uses two source terms for the effluent evaluations: radwaste system design basis source term and realistic source term. The definition of these two source terms is consistent with the description of the source terms found in Section C.I.11 of Regulatory Guide 1.206.

> "*Provide two source terms for (1) the primary coolant and reactor steam for BWRs, and (2) primary and secondary coolants for PWR plants. The first source term is a conservative or Radwaste System Design Basis source term, which assumes a Radwaste System Design Basis fuel defect level. Provide the Radwaste System Design Basis reactor primary and secondary coolant fission, activation, and corrosion product activities. The reactor core fission product inventories are determined based on time-dependent fission product core inventories that are calculated by the ORIGEN code. The first source term serves as a basis for:*

(1) Radwaste system design capability to process radioactive wastes at Radwaste System Design Basis fuel defect level and fission product leakage level,

(2) Confirmation of compliance with radioactive gaseous and liquid effluent release standards and effluent monitoring requirements under routine operations and anticipated operational occurrences, and

(3) Shielding requirements and compliance with occupational radiation exposure limits.

The second source term is a realistic model, which represents the expected average concentrations of radionuclides in the primary and secondary coolant. Provide realistic reactor primary and secondary coolant fission, activation, and corrosion product activities. The supporting information should describe expected liquid and gaseous source terms by plant systems, transport or leakage mechanisms, system flow rates, applicable radionuclide partitioning and decontamination factors, etc., and release pathways. For PWRs, provide these activities in the steam generator secondary side for the liquid and steam phases. These values should be determined using the model in ANSI/ANS 18.1-1999, NUREG-0016 (BWR-GALE code), and NUREG-0017 (PWR-GALE code).

The realistic source term provides the bases for estimating typical concentrations of the principal radionuclides. This source term model reflects the industry experience at a large number of operating reactor plants. The realistic source term is used to calculate the quantity of radioactive materials released annually in liquid and gaseous effluents during normal plant operation, including AOOs to demonstrate compliance with the liquid and gaseous effluent concentration limits in Table 2 of Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20; the dose limits in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public"; the compliance requirements in 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public"; and the ALARA design objectives of Appendix I to 10 CFR Part 50."

The Radwaste System Design Basis source term and the realistic source term both addressed a TPC by adding a tritium source term based on 1,792 TPBARs and a permeation rate of 5 Ci/TPBAR/year. This permeation rate bounds that observed for WBN Unit 1 and is consistent with that approved for WBN Unit 1 in License Amendment 107.

The TPBARs are designed with a stainless steel cladding and an aluminized internal coating. The tritium is produced by neutron irradiation of lithium aluminate pellets contained within the cladding and is gettered (collected and retained) by annular zirconium sleeves (getters) around the pellets. The aluminized coating and stainless steel cladding act as a barrier to tritium release (see Figure 4.1-5).

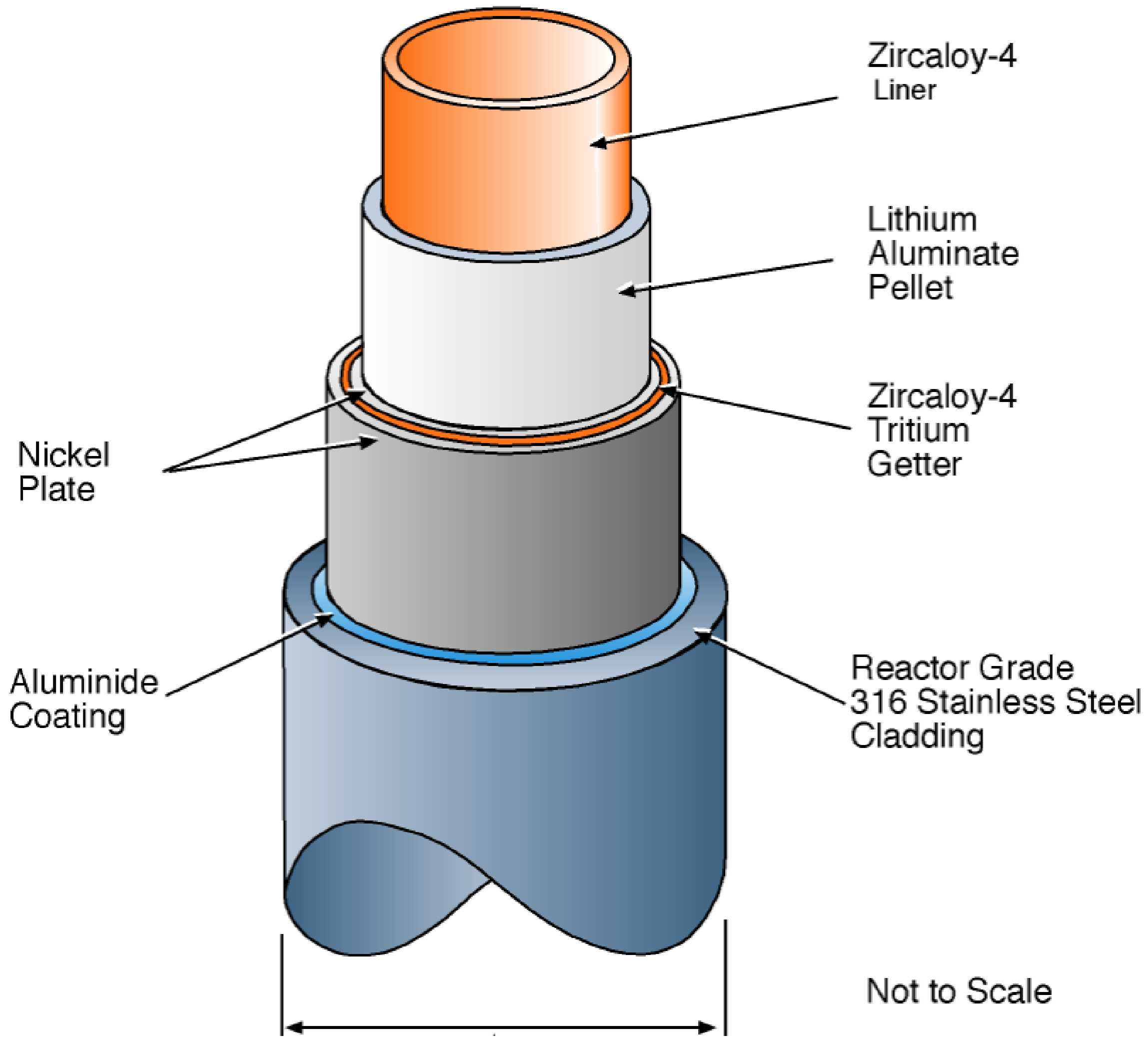

Figure 4.1-5: Concentric, Cylindrical, Internal Components of a TPBAR

TPBARs are designed and fabricated to retain as much tritium as possible within the TPBAR. Because the majority of TPBAR produced tritium is chemically bonded within the TPBAR, only a small percentage of the produced tritium is available in a form that could permeate through the TPBAR cladding.

As with other tritium producing components (fuel rods, control rods, secondary neutron source rods, etc.) 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 as low as reasonably achievable. TPBAR permeation is nonlinear with respect to the core's effective full power days (see Figure 4.1-6). A typical TPBAR's tritium inventory begins at zero at the start of the irradiation cycle and ends with about 9,200 Ci 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 4.1-6 demonstrates this process by using the Cycle 14 estimated tritium production.

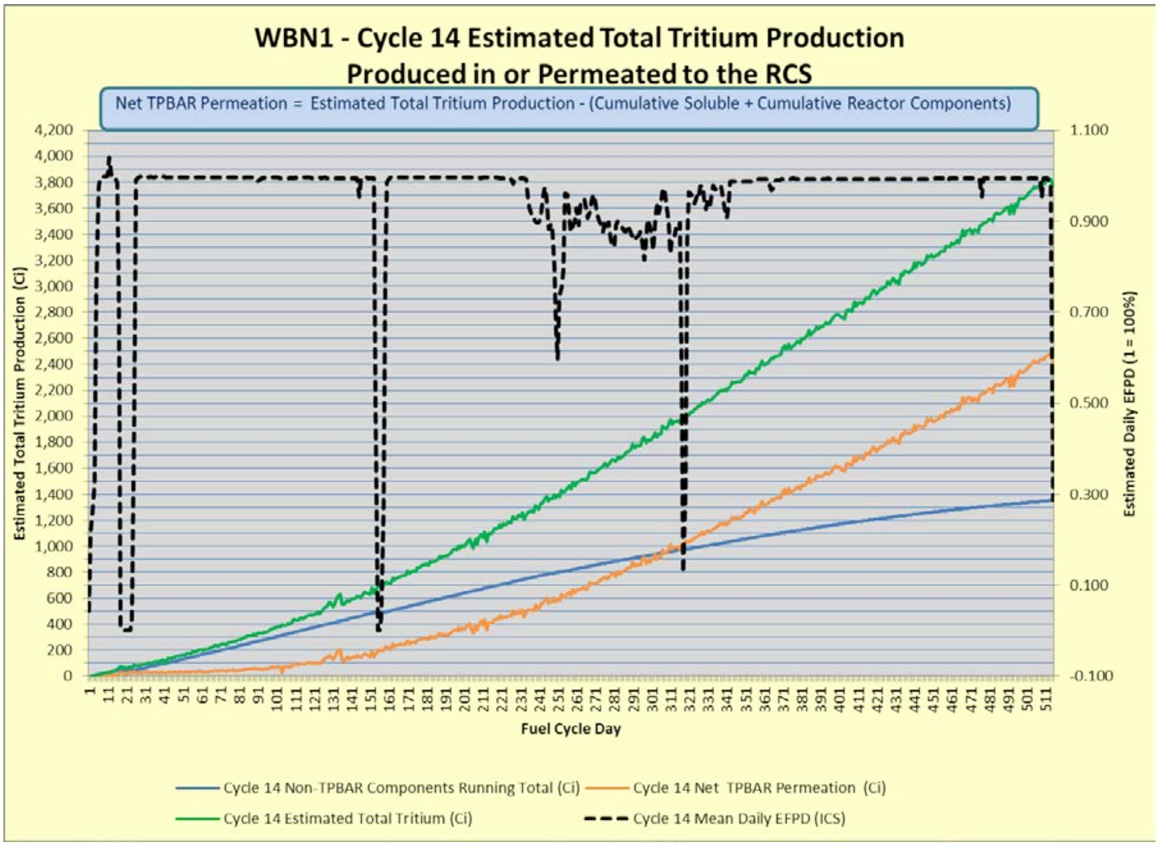

Figure 4.1-6: Cycle 14 Estimated Total Non-TPBAR and Total Tritium Production/Releases to the RCS

The estimated cumulative tritium permeation per cycle time in effective full power days (EFPD) per TPBAR for WBN Unit 1 Tritium Production Cycles 6 through 14 are shown in Figure 4.1-7. The uncertainty bars represent the 90 percent confidence interval.

Estimated Tritium Permeation

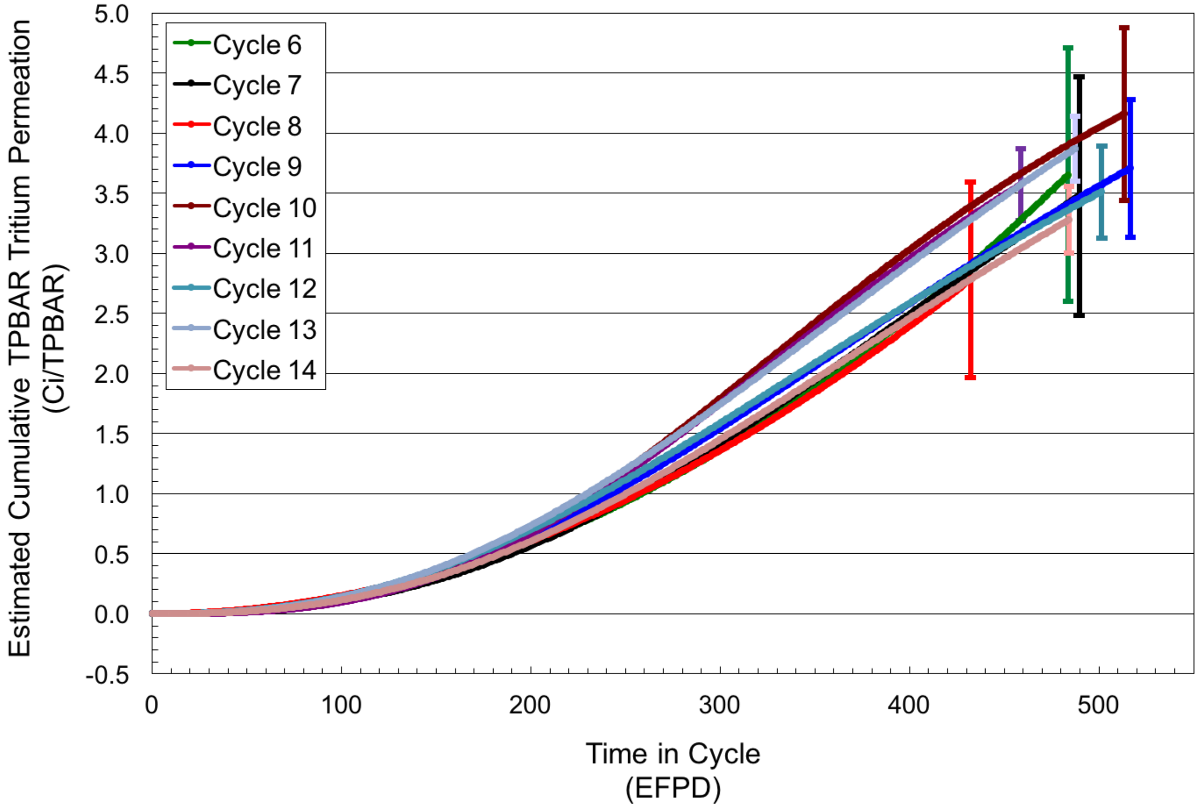

Figure 4.1-7: Estimated TPBAR Permeation for WBN Unit 1 Cycles 6 through 14

The Mark 9.2 TPBAR design included significant design changes from the multi-pencil Production TPBAR design of the prior TPC Cycles (Reference 43). However, the average annual release rate per Mark 9.2 TPBAR (3.4 ± 0.8 Ci/TPBAR/cycle) is similar to that estimated for the multi-pencil Production Design TPBARs of previous WBN Unit 1 cycles.

When the TPBAR permeation estimates are presented in a calendar year format (see Figure 4.1-8), corresponding to the NRC monitoring and reporting requirements, the annualized per TPBAR permeation have consistently remained less than 3 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.

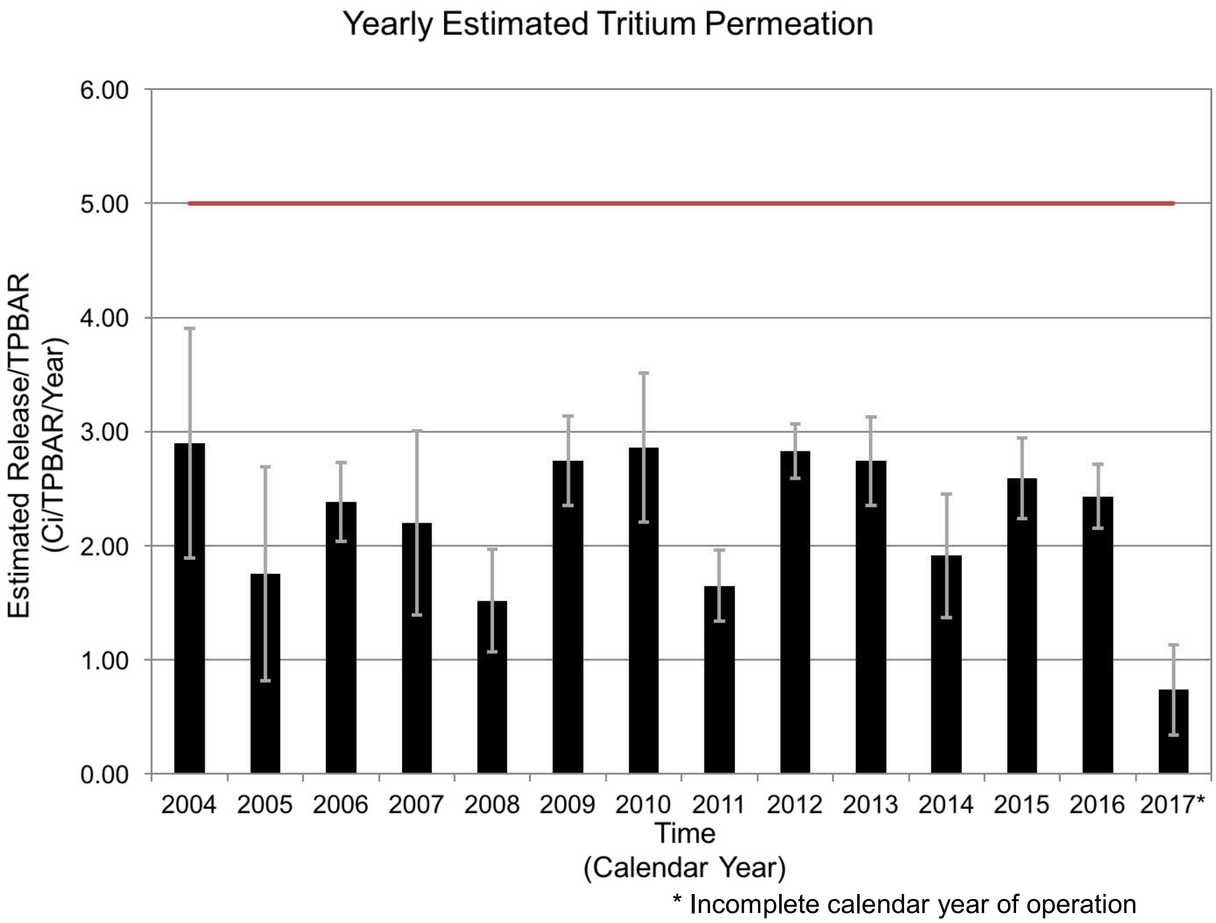

Figure 4.1-8: Estimated Annual TPBAR Permeation for WBN Unit 1 Cycles 6 through 14

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 TPC core is provided in Table 4.1-27. The Iodine and noble gas inventories are less for a TPC, with the exception of Iodine-131. This increase can be attributed to modeling differences and is not considered significant. 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 License Amendment 40. However, as discussed in Interface issue #5, the realistic primary coolant concentrations have been updated to correct errors. Therefore the design basis source terms have been changed since the initial review of the WBN Unit 2 operating license.

Table 4.1-27: ORIGEN2.1 Radioisotope Non-TPC and TPC Comparison

Isotope	Total Core Inventory (Ci)	
	Conventional Core	TPC
Kr 85m	3.95E+07	2.69E+07
Kr 85	9.99E+05	8.81E+05
Kr 87	7.59E+07	5.23E+07
Kr 88	1.08E+08	7.38E+07
Xe 133	2.03E+08	1.88E+08
Xe 135m	5.46E+07	3.59E+07
Xe 135	5.55E+07	4.96E+07
Xe 138	1.79E+08	1.59E+08
I 131	8.80E+07	9.01E+07
I 132	1.34E+08	1.31E+08
I 133	1.97E+08	1.88E+08
I 134	2.31E+08	2.08E+08
I 135	1.79E+08	1.76E+08

Note 1: WBN 96-Feed Equilibrium Core End-of-Cycle Operation at 3480 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 and is calculated to be 1,392 Ci/year. The contribution from TPBARs is calculated to be 8,960 Ci/year (1,792 TPBARs at 5 Ci/TPBAR/year). This results in an assumed total annual tritium release of 10,352 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.

<u>Realistic Source Terms</u>

The NRC's regulatory guidance on WBN's nominal tritium production is found in NUREG-0017 (Reference 44). The calculated realistic WBN Unit 2 average annual tritium value from NUREG-0017 is 1,392 Ci. To account for a TPC, an additional 8,960 Ci/year (1,792 TPBARs at 5 Ci/year) was used. Therefore, a total average annual

10,352 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 the WBN Unit 1 License Amendment 107, 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.

The realistic source term was also updated to correct errors as discussed in Interface Issue #5. The source terms were determined utilizing ANSI/ANS-18.1-1984 with corrected input values.

<u>Radwaste System Design Basis Operation</u>

WBN Unit 2 operation with a TPC containing up to 1,792 TPBARs will have a minimal effect on the Radwaste System Design Basis and realistic fission and corrosion product sources and the treatment of these isotopes in liquid and gaseous waste.

Effluent releases to the environment are controlled to meet 10 CFR Part 20 release limits by WBN Unit 2 Technical Specification (TS) 5.7.2.7, *Radioactive Effluent Controls Program*. The Radioactive Effluent Release Report is submitted to NRC as required by 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 and solid waste released from each unit.

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 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. The minimum CTB dilution flow required for discharge of radioactivity has been revised from 20,000 gpm to 30,000 gpm.

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 TPC in WBN Unit 2 by utilizing the Radwaste System Design Basis source term described above. The analysis was also updated to credit increased dilution flow from the CTB line from 20,000 gpm to 30,000 gpm. All other parameters reviewed and approved as part of NUREG-0847 Supplement 24 remain the same.

Table 4.1-28 shows the result without radwaste system processing. Table 4.1-29 and Table 4.1-30 demonstrate that the liquid releases do not exceed the 10 CFR Part 20

Appendix B Table 2 limits. Table 4.1-30 shows the results when the calculated annual total quantity of radioactive material, except tritium and dissolved gases, is limited to 5 curies for each reactor at the Watts Bar site.

Table 4.1-31 and Table 4.1-32 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 4.1-28: 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
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
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

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
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)	9316.80	1	9316.80	1.56E-04	1.00E-03	1.56E-01	3.12E-01
Total						5.09E+00	1.02E+01
Total (TPC)						5.23E+00	1.05E+01

Table 4.1-29: 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
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

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
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)	9316.80	1	9316.80	1.56E-04	1.00E-03	1.56E-01	3.12E-01
Total						2.83E-01	5.65E-01
Total(TPC)						4.18E-01	8.35E-01

Table 4.1-30: 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
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

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
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)	9316.80		1	9.32E+03	1.55E-04	1.00E-03	1.55E-01	3.10E-01
Total							2.89E-01	5.79E-01
Total(TPC)							4.23E-01	8.47E-01

Table 4.1-31: 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
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)	1.04E+03	1.00E+00	1.04E+03	4.69E-10	1.00E-07	4.69E-03	9.39E-03
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)						4.56E-01	9.12E-01

Table 4.1-32: 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+01	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-13	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
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)	1.04E+03	1.00E+00	1.04E+03	4.69E-10	1.00E-07	4.69E-03	9.39E-03
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)						4.01E-01	8.02E-01

Tritium Impacts on Public Dose During Normal Operation

Using the realistic TPC source terms for 1,792 TPBARs, the annual releases were reanalyzed. All other parameters reviewed and approved as part of NUREG-0847 Supplement 24 remain the same. The liquid annual releases are summarized in Table 4.1-33. The gaseous releases are summarized in Table 4.1-34.

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 4.1-35. This table also lists WBN's regulatory established radioactive effluent guidelines and the estimated non-TPC values.

Table 4.1-33: 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
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

Isotope	LRW (no SGB) (Ci)	SGB with no CD process (Ci)	Total (Ci)
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
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)	9316.8		9316.8
Total w/o H-3			5.0
Total w/ H-3			1257.8
Total w/(TPC)			9321.8

Table 4.1-34: 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)	1.04E+03	0.00E+00	0.00E+00	1.04E+03

Nuclide	Containment Building (Ci)	Aux. Building (Ci)	Turbine Building (Ci)	Total (Ci)
Cr51	9.21E-05	5.00E-04	0.00E+00	5.92E-04
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 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 4.1-35: Annual Projected Impact of TPC (1,792 TPBARs) on Effluent Dose to Maximally Exposed Members of the Public and Population Dose per Unit

	Non-TPC Dose	TPC Dose	Incremental Increase from TPC	NRC Annual Effluent Exposure Guideline
Annual Radioactive Gaseous Emissions				
Maximally Exposed Individual (mrem) Total Body	0.63	0.63	0	5.00 Total Body
Maximally Exposed Individual (mrem) Organ	5.54 (Bone)	6.14 (Bone)	0.60	15.00 Any Organ
50-mile Population Dose (person-rem)	11.3 (Thyroid)	16.9 (Thyroid)	5.6	NA
Annual Radioactive Liquid Emissions				
Maximally Exposed Individual (mrem) Total Body	0.34	0.37	0.03	3.00 Total Body
Maximally Exposed Individual (mrem) Organ	0.47 (Liver)	0.49 (Liver)	0.02	10.00 Any Organ
50-mile Population Dose (person-rem)	6.9 (Thyroid)	11.0 (Thyroid)	4.1	NA

Table 4.1-35 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.

40 CFR 190.10, “Standards for normal operations,” in part (a), sets annual dose equivalent limits for the normal operation at Watts Bar site. Using the revised realistic TPC source terms for 1,792 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 4.1-36. This table also lists the regulatory established dose limits.

Table 4.1-36: Annual Projected Impact of Two TPCs (1,792 TPBARs) on 40 CFR Part 190 Compliance

	Site Dose from Two TPCs	40 CFR 190 Limit
Whole Body (mrem)	7.89	25
Thyroid (mrem)	21.01	75
Other Maximum Organ (mrem)	16.65	25

Table 4.1-36 demonstrates that the resultant environmental releases from tritium production at the site meet the Environmental Protection Agency limits.

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 ≈5.0 percent. 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 1,792 TPBARs at 5 Ci/TPBAR/year, the average RCS tritium concentration is calculated to be approximately 12 µCi/gm.

There is a strong correlation between the RCS tritium concentration and the containment airborne tritium concentration (Station tritium dose). It is understood that containment tritium 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 License Amendment 40, site-specific data collected during extended non-TPC operating cycles (i.e., WBN Unit 1 Cycle 3 and SQN Unit 1 Cycle 10, breaker-to-breaker Non-TPC cycles) have provided useful data to estimate the effect from tritium production on TVA PWR station radiological conditions. The RCS maximum tritium levels noted during the extended operating cycles were ~2.5 µCi/gm with a cycle RCS tritium mean of ~1.0 µCi/gm. The extended cycle tritium peak RCS tritium value of ~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 ~1.0 µCi/gm resulted in a containment average DAC-fraction of about 0.08.

The calculated tritium release to the RCS with a TPC containing 1,792 TPBARs will result in about a factor of seven increase over the Non-TPC tritium production rate, that is,

Ratio = (TPC) 10,352 Ci/year / (Nominal Core) 1,392 Ci/year = 7.4

TVA determined that with no modifications to the current boron-control feed and bleed methodologies (i.e., ~366,000 gallon cycle letdown), the RCS average tritium value will be approximately 12 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 * 12 µCi/gm = 0.96 DAC-Fraction

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

0.96 DAC-fraction * 2.5 mrem/DAC-hour = ~2.4 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 * 12 µCi/gm /1 µCi/gm H3 = 24 person-rem/year

It should be noted that in NUREG-0713 volumes 29 through 34 (2007 through 2012), 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 1,792 TPBAR core, this additional TEDE is approximately 1.3 rem/year (1.9 rem per TPC cycle) for TPBAR handling and consolidation activities.

Therefore, an additional 25.3 rem/year is estimated for the increase in airborne activity and for fuel and TPBAR handling activities. WBN's three year collective TEDE per reactor year 2010 – 2012 was 39.998 rem. An additional annual average 25.3 tritium rem would raise the TEDE total to 65.3 rem; a value that remains within the 149 rem assessment total.

Solid Radioactive Waste

For normal TPC operations, the additional solid waste associated with TPCs that TVA will need to handle will be the base plate and thimble plug assemblies that remain after TPBAR consolidation activities. TVA will consolidate and temporarily store these items on-site. Offsite shipment and ultimate disposal is conducted in accordance with agreements between TVA and DOE. WBN Unit 1 License Amendment 40 estimated activity inventory associated with these additional irradiated components (112 base plates and 384 thimble plugs) as 5,921 Ci per cycle (180 day post irradiation decay) or an average of 3,527 Ci per year when adjusted to reflect measured dose rate for Base Plate with 24 Thimble Plugs following 113 day decay adjusted to 180 days. This represents an increase from the current WBN UFSAR estimated non-TPC value of 1,800 Ci/year to approximately 5,530 Ci/year for a TPC. This increased activity is associated with metal activation products. The estimated disposal volume of this additional solid waste is 50 cubic feet per TPC operating cycle or an average of 33.3 cubic feet per year.

This additional volume is an insignificant increase in the WBN annual estimated non-TPC solid waste (from the UFSAR), from 32,820 cubic feet per year to 32,853 cubic feet per year for a TPC.

WBN Unit 1 License Amendment 40 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 resins was evaluated. As discussed previously, the Radwaste Design Basis does not include consideration of two TPBAR failures, so no additional feed and bleed operations are expected for WBN Unit 2, and therefore, no additional resins are evaluated.

WBN Unit 1 License Amendment 40 assessed the environmental impact from the solid radioactive waste associated with the production of 2,304 TPBARs. The WBN Unit 2 proposed license amendment establishes 1,792 as the maximum number of TPBARs per cycle.

Thus, the WBN Unit 2 tritium production solid radioactive waste environmental impact is consistent with the WBN Unit 1 License Amendment 40 impact assessment and results in an insignificant increase to the WBN Unit 2 non-TPC waste.

Spent Fuel Generation and Storage

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 1,792 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.

Thus, the WBN Unit 2 tritium production additional spent fuel generation environmental impact is consistent with the WBN Unit 1 License Amendment 40 impact assessment.

4.1.15 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 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents (References 1, 2, 9, 17, 23, 39, 40, and 41).

Process and Effluent Monitoring and Sampling

TVA previously performed an evaluation of the production of tritium using TPBARs in WBN Unit 1 and determined that no additional sampling points were needed beyond those presently required by plant technical specifications during the normal plant operating and refueling operations with a TPC. This information is also applicable to the WBN Unit 2 TPC license amendment.

TVA previously reviewed its process and effluent monitoring and sampling equipment program and determined that this program required minor modifications for a TPC. A limited number of changes were made. Prior to initial TPBAR irradiation, TVA modified the Auxiliary Building and Shield Building Exhaust tritium sampling from periodic grab samples to continuous sampling with fixed monitoring systems. These samplers do not initiate any automatic actions. The Auxiliary Building and Shield Building Exhaust are independent of the control room heating, ventilation, and air conditioning (HVAC) system (i.e., no interconnections on the air supply from these two HVAC systems to the supply of the control room normal HVAC system). Plant specific procedures were developed before TPBAR irradiation addressing these actions.

TVA examined the need to perform tritium monitoring at the Condenser Vacuum Exhaust (CVE) air ejector, given the RCS sampling frequency and the allowable Technical Specifications primary to secondary leakage, to determine if it constituted an unmonitored release point, because most air ejector monitoring at commercial plants cannot detect tritium beta activity. The CVE is a continuously monitored release point with a noble gas radiation monitor. This monitor uses increased gamma activity as the indicator of a primary to secondary leak. The monitor has a setpoint that causes an

alarm in the Control Room. The monitor output is also fed to the plant computer. If the monitor shows an increase in count rate, a noble gas sample is obtained to validate the monitor response. If the sample contains radioactivity, then tritium, particulate, and iodine samples are obtained and factored into the release to the environment.

TVA previously reviewed the radioactivity monitoring programs for outdoor liquid storage tanks and has verified that the existing programs provide an appropriate level of assurance for operation with a TPC. The programs ensure that with an uncontrolled release of the tanks' contents the resulting radioactivity would be less than the regulatory limits at the nearest potable water supply or the nearest surface water supply.

TVA provided a summary of the groundwater monitoring program for WBN to NRC (Reference 45). The letter was submitted because of an industry initiative sponsored by the Nuclear Energy Institute (NEI) to compile baseline information about the status of site programs for monitoring and protecting groundwater and to share that information with NRC.

WBN is committed to the industry initiative described in NEI 07-07 (Reference 46) as documented in UFSAR Section 1.1.3, item 30. NEI 07-07 identifies actions to improve utility management and response to instances where the inadvertent release of radioactive substances may result in low but detectible levels of plant-related materials in subsurface soils and water.

The most recent results of the TVA groundwater monitoring program were provided to the NRC in the 2015 Annual Radioactive Effluent Release Report (Reference 47).

TVA will review and modify process and effluent monitoring and sampling actions, action levels, and sample frequencies, as necessary, based on TPC operating experience.

Tritium Monitoring Methods

In this section, the various techniques used to monitor for tritium in gases (primarily air), and in liquids are discussed.

Air Sampling

For tritium air sampling the sampled gas (usually air) must be analyzed for tritium activity by liquid scintillation counting. The technique is to flow the sampled air through either a molecular sieve containing solid desiccant or a column containing silica gel.

The typical lower limit of detection for in-station tritium air samples is 2×10^{-10} microcuries per cubic centimeter (μCi/cc).

TVA uses airborne monitoring for TPBAR consolidation activities using SCINTREX portable tritium air monitors model 309A, or equivalent. These monitors are used in the fuel handling area when moving fuel containing irradiated TPBARs or while consolidating irradiated TPBARs. Procedures are in place for using and calibrating these portable tritium air monitors.

Liquid Monitoring

Liquids will be monitored by liquid scintillation counting. The typical lower limit of detection for in-station tritium liquid samples is 1 x 10^{-6} microcuries per gram (μCi/gm).

Liquid Scintillation Counting

Liquid scintillation counting is a convenient, reliable, and practical way of measuring tritium in the liquid phase. The technique consists of dissolving or dispersing the tritiated compound in a liquid scintillation cocktail, and counting the light pulses emitted from the interaction between the tritium betas and the cocktail. The light pulses are counted by a pair of photomultiplier tubes which, when coupled with a discriminator circuit, can effectively distinguish between tritium betas and those from other sources.

TVA's liquid scintillation counters are periodically calibrated with radioactive sources, which are National Institute of Standards and Technology traceable and are calibrated in an ISO9001 certified and ISO17025 accredited laboratory. The counters are checked periodically with standard radioactive sources in accordance with instrument specific calibration and maintenance procedures.

TVA's current techniques for tritium air sampling, liquid monitoring, and liquid scintillation counting are appropriate and modifications are not warranted.

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

NUREG-1672, Section 2.15.5, "The staff concludes from its review that calculated TPBAR performance under LOCA conditions has demonstrated that TPBARs can be assessed with approved licensing LOCA models and can perform acceptably under LOCA conditions. However, the staff also concludes that, although the LOCTA_JR code was appropriate for use in the demonstration analyses and assessments discussed herein, LOCTA_JR was not reviewed for licensing use and should be reviewed by the staff for licensing applications and for its interface with the specific plant licensing LOCA models before it is used in specific plant licensing applications."

The TPBAR Interface Issue 16 information provided for the Watts Bar Unit 2 Tritium Production Program LAR is based on the applicable Watts Bar Unit 1 precedent documents (References 1, 9, 48, 49, and 50).

The LOCTA_JR computer code was used in preparation of the Tritium Production Core Topical Report NDP-98-181, Revision 1, (Reference 7) to evaluate the thermal response of TPBARs to a LOCA. This code performed one-dimensional radial heat conduction calculations for a fuel rod.

On June 23, 2000, TVA submitted Westinghouse Topical Report WCAP-15409, “Description of the Westinghouse LOCTA_JR 1-D Heat Conduction Code for LOCA Analysis of Fuel Rods,” to the NRC for review and approval (Reference 48). The NRC staff completed its review of WCAP-15409 and concluded in a January 17, 2001, letter to TVA that the report was acceptable for referencing in licensing analyses (Reference 50).

TVA no longer uses the NRC-approved LOCTA_JR code to predict the thermal behavior of TPBARs during a LOCA. The TPBAR temperatures calculated by LOCTA_JR contained in NDP-00-0344 Rev. 1 were obtained using boundary conditions from both Best Estimate (LBLOCA) and Appendix K (SBLOCA) analyses of record for the Watts Bar Plant. The fuel rod temperatures and the core steam and entrained liquid convective heat transfer coefficients temperatures used in the previous LOCTA_JR predictions bound those expected in Watts Bar Unit 2.

4.1.17 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 TP BARs for the production of tritium."

The TPBAR Interface Issue 17 information provided for the WBN Unit 2 Tritium Production Program LAR is based on the relevant WBN Unit 1 precedent documents (References 1, 9, 51, and 52).

An Anticipated Transient without SCRAM (ATWS) is an operational occurrence followed by a common fault that prevents the reactor trip portion of the reactor protection system. The scenario that resulted in the highest consequences is an unmitigated ATWS which results in RCS over-pressurization which may result in a compromise of the RCS boundary and lead to core damage.

To limit the probability and consequences of such an occurrence, the NRC issued the 10 CFR 50.62, which required all operators to install additional hardware and protection features including auxiliary feedwater and turbine trip actuation. Like Watts Bar Unit 1, Unit 2 currently has ATWS Mitigation System Actuation Circuity (AMSAC) installed (see FSAR Section 7.7.1.12) to satisfy compliance with the rule.

The Westinghouse licensing implementation strategy that addresses 10 CFR 50.62 was documented in WCAP-15831-NP-A, Revision 2 (Reference 53). In this topical report, plants were placed into three separate groups sorted according to their ATWS Configuration Management Program. Both Watts Bar Units 1 and 2 are within Group 2 which states that they do not have a Diverse Scram System, but have a Moderator Temperature Coefficient (MTC) that is consistent with the basis for the ATWS rule. As such, the implementation of the NRC approved ATWS analysis methodology is not required. Instead, the full power MTCs will continue to be confirmed on a cycle-specific basis.

ATWS mitigating capability in a Pressurized Water Reactor (PWR) is highly dependent on the MTC since its negative reactivity feedback mechanism acts to reduce reactor power as the RCS coolant temperature increases. Having a sufficiently negative MTC allows for the pressure increase to be maintained below the system limits in the absence of control rod insertion.

The effect of TPBARs on the MTC was initially studied in the DOE TPC Topical Report (Reference 7). In order to address the plant-specific interface issue 17, TVA submitted a plant-specific analysis (Reference 51), which was approved by the NRC (Reference 52). The analysis included a comparison of the full power MTCs as a function of cycle length was performed for a representative TPC and a Watts Bar core without TPBARs. The 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, since 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.

The ATWS issue was previously considered for Watts Bar Unit 2, as documented in Section 15.3.6 of NUREG-0847, Supplement 24 (Reference 54).

4.2 Post-LOCA Subcriticality Evaluation

The TPBAR post-LOCA subcriticality evaluation provided for the WBN Unit 2 Tritium Production Program LAR is based on the relevant WBN Unit 1 precedent documents (References 2, 17, 55, 56, 57, and 58): The methodology used for the evaluation and analysis is the same as used in support of WBN Unit 1 license amendment 107 (Reference 17).

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

The post-LOCA reflood phase has previously been evaluated for WBN Unit 1 and shown to be non-limiting for subcriticality, and therefore, a WBN Unit 2 analysis is not necessary. Also, the hot leg break LOCA is non-limiting for subcriticality, and therefore a WBN Unit 2 analysis is not necessary.

During the sump recirculation phase, ECCS flow is drawn from the containment sump following the transfer from the refueling water storage tank (RWST) to the containment sump as the source of coolant. In order to show that the sump water has sufficient boron concentration, the sump mixed mean boron concentration is calculated. The mixed mean boron concentration of the sump water is a function of the various water and boron contributors to the sump prior to start of sump recirculation, and after realignment of the ECCS for cold-leg and hot-leg recirculation. The initial Reactor Coolant System and ice condenser water masses and boron concentrations assumed in the analysis are conservative values. The boron concentration of the sump water must be sufficient to keep the core subcritical when the sump water is delivered to the reactor vessel during the cold-leg and hot-leg recirculation phases. The sump mixed mean boron concentration calculations are used to develop a post-LOCA subcriticality boron limit that is confirmed on a cycle-specific basis as part of the Westinghouse Reload Safety Evaluation Methodology (Reference 59).

The boron concentrations in the RWST and the safety injection accumulators used in the analysis were:

- Minimum RWST Boron Concentration - 3100 ppm
- Minimum Accumulator Boron Concentration - 3000 ppm

The evaluation of the TPBARs in Unit 2 considered plant operation with both Original Steam Generators (OSGs) and Replacement Steam Generators (RSGs).

The evaluation assumed TPBAR failure for all TPBARs with 3 percent lithium (Li)-6 leaching instantaneously for the first day and a maximum of 50 percent Li-6 leaching thereafter along with a loss of 12 inches of lithium aluminate (LiAlO2) pellets. The technical basis for the post-LOCA TPBAR lithium leaching effects is described in Section 4.2.1.

The evaluation assumed that no unborated ERCW or CCS water leakage into the containment sump co-incident with the LOCA. The plant modifications associated with this assumption are described in Section 4.2.2.

The PLS Reload Safety Evaluation limits applicable to TPBAR reload cores are presented in Figures 4.2-1 and 4.2-2. These limits are valid for plant operation with either OSGs or RSGs. These limits will be verified for the demonstration TPBAR core design and for each reload cycle. This verification will confirm that adequate boron exists to maintain subcriticality in the long-term post-LOCA.

4.2.1 Post-LOCA Lithium Leaching Assumptions

The lithium leach rate assumption for the WBN Unit 2 cold leg break scenario is the same assumption that was used for WBN Unit 1 licensee amendment 107 (Reference 17). The lithium leach rate assumption is a time dependent leaching rate of 3 percent for the first day and a maximum of 50 percent thereafter. The cold leg break scenario differs from the hot leg break scenario due to the assumption of TPBAR failure and the potential for sump dilution only being applicable for the cold leg break location. At hot-leg switchover (HLSO), the TPBAR failure assumption of an instantaneous loss of 3 percent lithium inventory is conservative because leaching of the TPBARs is not instantaneous. The bounding leaching rate is 3 percent per day; therefore, some value less than 3 percent of the lithium would have leached at the time of HLSO (3 hours).

PNNL conducted leach tests with irradiated pellets in TPBAR test rod segments at a post-loss of coolant accident (LOCA) temperature of 120 °C. These tests verified that pellet leach rates were less than 3 percent per day for the first 14 days after a TPBAR breach. In addition to the 120 °C data, earlier leach tests results were available for bare pellets at 93 °C for periods of 60 days and TPBAR test rod segments at 73 °C and 93 °C for periods of eight days. This additional data was combined with the 120 °C, 14-day data to support a conservative extrapolation of leach results to longer time periods. These extrapolations were used to support conclusions about maximum lithium leaching at 93°C for periods up to 120 days.

The tested configuration consisted of a sample TPBAR section in a pressure vessel as shown in Figure 4.2-3. Each sample TPBAR section was fabricated with uncoated cladding and getter stock, production inner material, and two irradiated lithium aluminate pellets ($LiAlO_2$). A buffered, pH-adjusted borated test solution (2,180 ppm boron and pH of 10) was used to simulate the sump water. The test solution was periodically sampled and replaced with fresh solution.

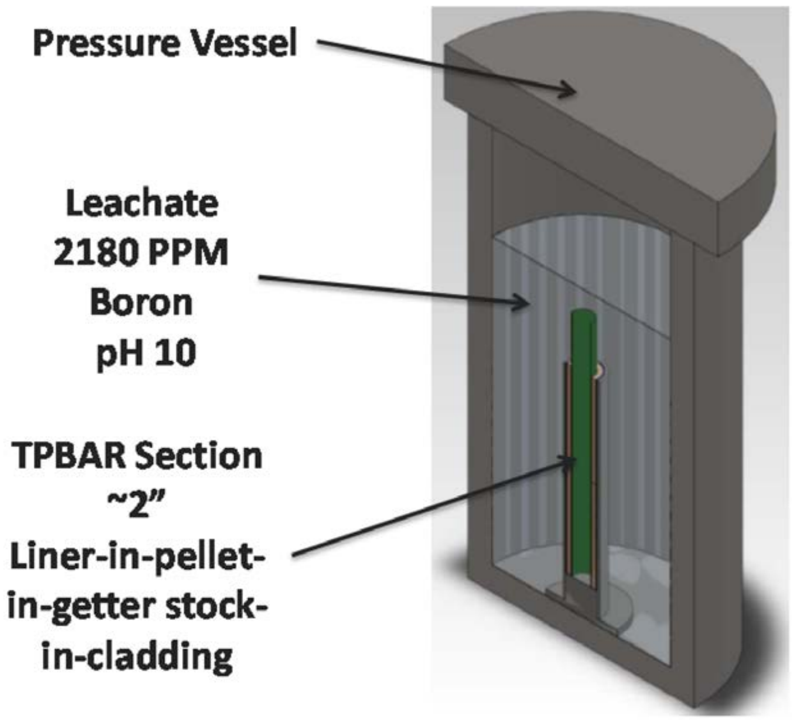

Figure 4.2-3: Schematic of the Pellet Leaching Test Configuration

The leach tests were performed in unstirred stainless steel pressure vessels at 120 °C and as-generated pressures. An actual TPBAR cladding breach creates a tortuous path that does not allow for free flowing coolant interaction with the pellet material that remains in the TPBAR. Therefore, no continuous flow past the TPBAR sections was used in the leach testing. Leaching of $LiAlO_2$ within a breached TPBAR occurs in stagnant conditions within the TPBAR. However, to conservatively bound the breach configuration relative to coolant interaction with the pellets, an open ended cladding section was used in the pellet leach test. The test included the mechanical loads due to temperature effects, thermal expansion, and pressure. The test was conducted at bounding post-LOCA temperature and pressure conditions.

Samples were taken at one day and 14 days. The average lithium leaching seen at one day with a 95 percent confidence upper bound was 2.05 weight percent (wt%) per day. The total lithium leached after 14 days with a 95 percent confidence upper bound was 8.73 wt%, or an overall average of 0.62 wt% per day.

The maximum TPBAR temperature after reflood (post-LOCA) is <248 °F (<120 °C) based on post-LOCA coolant temperature.

A burst TPBAR would result in about a two inch long fish-mouthed breach that clogs with a deformed mass of TPBAR components (getter, liner, pellets, and clad) and guide tube, resulting in a tortuous, obstructed path for fluid migration into a TPBAR. Full scale TPBAR burst testing has shown that an average of less than 12 inches of TPBAR pellet material is ejected as a result of a TPBAR breach. The safety analysis assumes that no pellet material remains near the breach location. During reflood, the TPBARs may fill with coolant but the potential for circulation of coolant past TPBAR pellet material is limited. Water stagnation in the TPBAR is the expected condition. When pellets are wet, swelling due to conversion to lower density phases further restricts the potential for coolant circulation. Thermal expansion and contraction of the TPBAR materials after reflood is otherwise minimal. Coolant would circulate outside a TPBAR, but the small quantity of coolant that gets to the TPBAR interior would not readily be exchanged.

The tested geometry consisted of a two inch long open ended TPBAR section containing irradiated pellets in a pressure vessel. The open end simulates a guillotine breach, which exposes the entire cross section to the test solution. Tests were conducted in stagnant test solution as that was the expected condition in the TPBAR and it facilitated testing protocols with irradiated samples. The results from the two inch section are conservatively applied to the entire length of the TPBAR even though the TPBAR geometry does not facilitate communication of the remaining pellets with the coolant. Although post-LOCA conditions would have high volumes of fresh coolant outside of the TPBAR, transfer of the small quantity of coolant present in the interior of the TPBAR would not occur in post-LOCA conditions due to the highly tortuous path at the post LOCA breach, the small pellet/getter/liner gaps, and the expected pellet swelling.

Use of irradiated pellets accounted for pellet irradiation damage and swelling. Past pellet leach tests show that initial pellet lithium leaching occurred via the conversion of $LiAlO_2$ into $LiAl_2(OH)_7{\bullet}2H_2O$ and LiOH with significant pellet swelling due to the formation of the lower density phases. High boron concentrations and a high pH both simulate the expected post-LOCA coolant condition. Large test solution/pellet ratios were used and when the test solutions were sampled, they were replaced with fresh solution. These conditions bounded the expected, stagnant condition.

The low Reynolds number post-LOCA coolant flows would cause only low energy vibration of the TPBAR. Vibration would have minimal effect on fluid exchange given the expected breach geometry.

The maximum Emergency Core Cooling System (ECCS) water temperature for WBN Unit 2 at any time post-LOCA is the saturation temperature at the bounding high post-LOCA containment pressure. This peak pressure is maintained for less than 24 hours and results in a maximum ECCS water temperature of < 244 °F (118 °C). TPBARs generate little heat as they have no fission products and are not significantly hotter than the ECCS water. The test vessel contents were maintained at 248 °F (120 °C) and as-generated pressures for the full duration of the test. As leaching increases with increasing temperature, testing at higher than expected temperatures for longer than expected times provides margin to the expected condition.

NRC conducted an audit of the TPBAR burst and lithium leaching test programs at PNNL (Reference 56). As a result of the audit, NRC requested that TVA submit certain portion of the TPBAR burst test document TTP-1-3010, Revision 0, "Phase Four Full-Length TPBAR Burst Testing." TVA submitted the following information (Reference 57):

- Statistical Reanalysis of Mark 9 .2 and Mark 8.1 TPBAR Test Pellet Length Loss (page v)
- Pellet Length Effect on Mark 9.2 TPBAR Burst Testing (page vi)
- Section 3.0, Description of SMART Facility (pages 5 - 9)
- Section 4.0, Test Method (pages 10 - 11)
- Subsection 5.3, Mark 9.2 Test Article Burst Test Summaries (pages 19 - 32)

This test information is also applicable to the Watts Bar Unit 2 TPC license amendment.

4.2.2 WBN Unit 2 Plant Modifications

The sump boron concentration curves are maintained as key safety parameters in the reload safety evaluation process. Three sump boron curves are generated corresponding to three different times following the break: (1) at initiation of cold leg recirculation, (2) at HLSO, and (3) at long term cooling. These sump boron curves account for the removal of an unborated dilution source that would enter the containment at a maximum rate of 40 gallons per minute and would be isolated within 16 hours after the break. The manual actions associated with isolating these dilution pathways within 16 hours were previously reviewed and approved by the NRC in Section 6.2.4 of NUREG-0847, Supplement 22 (Reference 13).

In order to eliminate this unborated dilution source and the associated manual actions, TVA is replacing the containment isolation thermal relief check valves on the lower compartment supply lines to the containment for WBN Unit 2 CCS and ERCW System with simple relief valves. The simple relief valves would only open to relieve an overpressure condition. The thermal relief check valves were the pathway for the unborated dilution source affecting the post-LOCA containment sump boron concentration.

The following Operator Manual Actions described in NUREG-0847, Supplement 22, Section 6.2.4 are being eliminated by the design change to replace the containment isolation thermal relief check valves on the lower compartment supply lines to the containment for WBN Unit 2 CCS and ERCW System with simple relief valves (i.e., passive devices).

1. Manual isolation of the six-inch ERCW supply to the lower containment cooler Group D from the main supply Header 2B by closing the safety-related valve 2-ISV-67-523B, which is located in the Auxiliary Building at elevation 692, within 16 hours after the postulated accident.

2. Manual isolation of the six-inch CCS supply and return lines for the reactor coolant pump oil cooler penetrating containment by closing safety-related valves 2-ISV-70-516 (supply) and 2-ISV-70-700 (return), as applicable within 16 hours of the accident, concurrent with the single failure of the outboard containment isolation valve to close.

The scaffolding to access ERCW supply valve 2-ISV-67-523B and the associated administrative controls will no longer be required following implementation of the valve replacement design change. Emergency Operating Instruction 2-E-0, "Reactor Trip of

Safety Injection," will be revised to remove the above Operator Manual Actions and operator training on the deletion of the Operator Manual Actions will be required.

TVA has concluded that there will be no changes to the control room interface (i.e., displays, controls, and alarms), simulator, and Safety Parameter Display System required to support the design change because the manual actions are performed remotely from the control room. These changes will be made as part of the design change to replace the valves in accordance with TVA procedure NPG-SPP-09.3, "Plant Modifications and Engineering Change Control." These requirements provide assurance that the affected Disciplines and Departments review the change as it is developed for impact to items under their control, such as procedures, training, control room interfaces, and the simulator.

4.2.3 Results

The core characteristics for fuel management are nearly identical between Watts Bar Unit 1 and Watts Bar Unit 2. Both plants operate at the same power level, utilize the same fuel product, and have similar inlet temperature and flow values. As a result, the same loading pattern developed to support the Watts Bar Unit 1 License Amendment 107 (Reference 2) is also applicable to Watts Bar Unit 2. The same analysis that led to the limiting boron concentration for Watts Bar Unit 1 post LOCA evaluation can be used and compared to the different limit for Watts Bar Unit 2.

4.2.3.1 Cold Leg Break Scenario - Hot Leg Switchover Assessment

In the cold leg break scenario, TPBAR failure is conservatively assumed to occur. The key assumptions for this scenario are the same as used for the WBN Unit 1 LAR supporting license amendment 107 (Reference 17):

a) a pre-condition of peak xenon to minimize the RCS boron concentration
b) cold conditions (50°F to 212°F)
c) TPBAR failure for all TPBARs with 3% lithium (Li)-6 leaching instantaneously for the first day and a maximum of 50% Li-6 leaching thereafter along with a loss of 12 inches of lithium aluminate (LiAlO2) pellets
d) an instantaneous loss of helium (He)-3 inventory
e) no control rod insertion
f) no credit for void feedback
g) sump dilution at the time of hot leg switchover (HLSO)
h) a conservative xenon credit at the time of HLSO (3 hours)
i) most reactive time in life

The limiting result for WBN Unit 2 are shown in Table 4.2.-1, along with the WBN Unit 1 results. The WBN Unit 2 minimum post-LOCA subcriticality margin at the time of HLSO is 116 ppm at the time-in-cycle corresponding to a burnup of 6000 MWd/MTU. This is comparable to the WBN Unit 1 result of 129 ppm. This results meets the minimum subcritical margin target of ≥100 ppm margin.

Table 4.2-1: Post-LOCA Subcriticality for a Cold Leg Break - HLSO Assessment for 1792 TPBAR Core

Burnup (MWD/MTU)	Pre-condition Boron Concentration HFP, Peak Xenon (ppm)	Sump Boron (ppm)	No Xenon, Cold Critical Boron (ppm)	Xenon Credit (ppm)	Cold Critical Boron with Xenon Credit (ppm)	Subcriticality Margin (ppm) for Watts Bar Unit 2	Subcriticality Margin (ppm) for Watts Bar Unit 1
150	357	1675	1628	210	1418	284	257
1000	420	1688	1664	210	1454	253	234
2000	508	1707	1716	210	1506	207	201
3000	578	1722	1759	210	1549	169	173
4000	621	1731	1790	210	1580	141	151
6000	644	1736	1817	210	1607	116	129
8000	605	1728	1811	210	1601	119	127
10000	525	1711	1770	210	1560	154	151

4.2.3.2 Cold Leg Break Scenario - Long-Term Cooling Assessment

The long-term cooling assessment for the cold leg break scenario uses the same assumptions as the HLSO assessment with the following exceptions:

a) long-term mixed sump boron concentration
b) no credit for xenon

The results for WBN Unit 2 are shown in Table 4.2-2, along with the WBN Unit 1 results for comparison. The WBN Unit 2 minimum post-LOCA subcriticality margin for the long-term assessment is 263 ppm at the time-in-cycle corresponding to a burnup of 6000 MWD/MTU. This is comparable to the WBN Unit 1 result of 212 ppm. This result meets the minimum subcritical margin target of ≥100 ppm.

In conclusion, the long-term assessment results are bounded by the HLSO results (Section 4.2.3.1) and WBN Unit 2 is bounded by WBN Unit 1. The minimum subcritical margin of ≥100 ppm is satisfied for all post-LOCA scenarios and conditions based on the current analysis inputs and methodology.

Table 4.2-1: Post-LOCA Subcriticality for a Cold Leg Break - Long Term Assessment for 1792 TPBAR Core

Burnup (MWD/MTU)	Pre-condition Boron Concentration HFP, Peak Xenon (ppm)	Sump Boron (ppm)	No Xenon, Cold Critical Boron (ppm)	Subcriticality Margin (ppm) for Watts Bar Unit 2	Subcriticality Margin (ppm) for Watts Bar Unit 1
150	357	2095	1728	432	367
1000	420	2103	1765	401	338
2000	508	2116	1819	355	297
3000	578	2126	1863	317	263
4000	621	2132	1895	289	237
6000	644	2135	1923	263	212
8000	605	2129	1917	265	212
10000	525	2118	1876	299	242

5.0 REGULATORY EVALUATION

5.1 Applicable Regulatory Requirements and Criteria

The TPBARs are described in the WBN UFSAR:

- Section 4.2.4, Tritium Producing Burnable Absorber Rod – Tritium Production Core
- Section 6.2.1.3.12, Tritium Production Core Evaluation
- Section 9.1.5, Tritium Producing Burnable Absorber Rods (TPBARs) Consolidation Activity
- Section 11.1, Source Terms
- Section 11.2, Liquid Waste Systems
- Section 11.3, Gaseous Waste Systems
- Section 11A, Tritium Control
- Section 12-A, Radiation Protection Features for the Tritium Production Program
- Section 13.7.3, Physical Security of TPBARs
- Section 15.3.5, Waste Gas Decay Tank Rupture
- Section 15.5, Environmental Consequences of Accidents

For these sections, the principal reviews performed by NRC are documented in the Safety Evaluation Report (SER), NUREG-0847, dated June 1982 (Reference 25) and various Supplements (References 12, 13, 31, 38, and 54). The assessment of these functions with respect to TPBAR irradiation is documented in the safety evaluation for WBN Unit 1 License Amendment 40 (Reference 1) and License Amendment 107 (Reference 2).

The information for TPBAR Interface Issue 4, "Reactor Vessel Integrity," addresses the operating experience described in NRC Regulatory Issue Summary 2014-11 (Reference 60).

The information for TPBAR Interface Issue 10, "New and Spent Fuel Storage," addresses the operating experience described in NEI 12-16 (Reference 61) and NEI 16-03 (Reference 62).

5.2 Precedents

TVA has determined that this request is similar to the following WBN Unit 1 License Amendments, which have been approved by the NRC:

1. NRC Letter to TVA dated July 29, 2016, "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)" (ML16159A057)

2. NRC letter to TVA dated September 23, 2002, "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)," (ML022540925).

The information provided for the WBN Unit 2 Tritium Production Program LAR is based on the applicable WBN Unit 1 precedent documents, as described in Section 4.0.

5.3 Significant Hazard Consideration

The Tennessee Valley Authority (TVA) proposes to revise the current licensing basis of Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant (WBN), Unit 2 by revising the WBN Unit 2 Technical Specifications (TS) 4.2.1, "Fuel Assemblies," to add a limit on the number of Tritium Producing Burnable Absorber Rods (TPBARs) that can be irradiated in the core. This proposed change will support a plan to irradiate TPBARs after WBN Unit 2 Cycle 4 refueling outage in the fall of 2020 to support national security needs.

This license amendment also provides proposed changes related to the new criticality analyses performed for the spent fuel storage racks. The proposed change would revise WBN Units 1 and 2 TS 3.7.15, “Spent Fuel Assembly Storage,” to simplify the fuel storage limitations on fuel assemblies by eliminating the burnup-related criteria. The proposed change would add WBN Units 1 and 2 TS 3.7.18, “Fuel Storage Pool Boron Concentration,” to specify the minimum fuel storage pool boron concentration when fuel is stored in the pool. The proposed change would revise WBN Units 1 and 2 TS 3.9.9, “Spent Fuel Pool Boron Concentration,” to modify the minimum fuel storage pool boron concentration during refueling operations when fuel is stored in the pool. The proposed change would revise WBN Units 1 and 2 TS 4.3, “Fuel Storage,” to replace the storage limitations on fuel assembly burnup and storage location with a single requirement to maintain a specified boron concentration in the spent fuel pool. The proposed change would add WBN Units 1 and 2 TS 5.7.2.21, “Spent Fuel Storage Rack Neutron Absorber Monitoring Program.” The proposed technical specification changes ensure continued compliance with 10 CFR 50.36, “Technical specifications.”

TVA has concluded that the changes to WBN Unit 1 TSs 3.7.15, 3.7.18, 3.9.9, and 4.3 and the WBN Unit 2 TSs 3.7.15, 3.7.18, 3.9.9, 4.2.1, 4.3, and 5.7.2.21 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 Unit 2 TS 4.2.1 adds a limit on the number of TPBARs that can be irradiated in the core. The safety analyses demonstrated sufficient reactivity control after a postulated loss of coolant accident (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.

The modifications to eliminate potential sources of post-LOCA sump dilution described in this amendment request restore the original design basis of the plant. These modifications eliminate operator actions credited to isolate the unborated water lines in the event of a design basis accident. The modifications eliminate the potential for human error associated with the required manual actions.

The results of the revised spent fuel criticality analysis show that the maximum k_{eff} of the PWR BORAL^{TM} spent fuel racks loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.00 with no credit for soluble boron for normal conditions; and less than 0.95 with credit for soluble boron for both normal and accident conditions, all for 95 percent probability at a 95 percent confidence level. For accident conditions, a soluble boron level of 500 ppm is sufficient to ensure that the maximum k_{eff} is below regulatory limit. The proposed change to WBN Units 1 and 2 TS 3.7.15, 3.7.18, 3.9.9, and 4.3 change the limit on minimum boron concentration in the spent fuel pool to be consistent with an updated criticality analysis for the spent fuel storage racks when fuel is stored in the pool.

The proposed change incorporates a new program as TS 5.7.2.21 to monitor the condition of the neutron absorber material used in the spent fuel pool storage racks to ensure it will continue to perform its assumed design functions. SFP storage rack neutron absorber monitoring is an administrative requirement that does not affect the ability of any structures, systems, and components (SSCs) to perform a design function. A SFP storage rack neutron absorber monitoring program is not an initiator to any accident previously evaluated and does not affect the consequences of any accident previously evaluated.

Based on the above discussions, the proposed changes do 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 2 TS 4.2.1 adds a limit on the number of TPBARs that can be irradiated in the core. The safety analyses demonstrated sufficient reactivity control after a postulated loss of coolant accident (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.

These modifications eliminate operator actions credited to isolate the unborated water lines in the event of a design basis accident. The modifications eliminate the potential for human error associated with the required manual actions. Because the TPBARs are manufactured to the same quality standards as the other core components, the possibility of a new or different kind of an accident is not created.

The results of the revised spent fuel criticality analysis show that the maximum k_{eff} of the PWR BORAL^{TM} spent fuel racks loaded with fuel of the highest

anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.00 with no credit for soluble boron for normal conditions; and less than 0.95 with credit for soluble boron for both normal and accident conditions, all for 95 percent probability at a 95 percent confidence level. For accident conditions, a soluble boron level of 500 ppm is sufficient to ensure that the maximum k_{eff} is below regulatory limit. The proposed change to WBN Units 1 and 2 TS 3.7.15, 3.7.18, 3.9.9, and 4.3 change the limit on minimum boron concentration in the spent fuel pool when fuel is stored in the pool to be consistent with an updated criticality analysis for the spent fuel storage racks.

The proposed change incorporates a new program as TS 5.7.2.21 to monitor the condition of the neutron absorber material used in the spent fuel pool storage racks to ensure it will continue to perform its assumed design functions. SFP storage rack neutron absorber monitoring is an administrative requirement that does not alter the design function or operation of the SSCs involved, and does not involve installation of a new SSCs. The proposed change will not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

Therefore, the proposed changes do 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 Unit 2 TS 4.2.1 adds a 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. The proposed number of TPBARs, in conjunction with the current boron concentration values, has been demonstrated to provide an adequate level of reactivity control for accident mitigation. This conclusion will be verified for each core that contains TPBARs as part of the normal reload analysis. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

The modifications to eliminate potential sources of post-LOCA sump dilution described in this amendment request restore the original design basis of the plant. These modifications eliminate operator actions credited to isolate the unborated water lines in the event of a design basis accident. The modifications eliminate the potential for human error associated with the required manual actions.

The results of the revised spent fuel criticality analysis show that the maximum k_{eff} of the PWR BORAL^{TM} spent fuel racks loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.00 with no credit for soluble boron for normal conditions; and less than 0.95 with credit for soluble boron for both normal and accident conditions, all for 95 percent probability at a 95 percent confidence level. For accident

conditions, a soluble boron level of 500 ppm is sufficient to ensure that the maximum k_{eff} is below regulatory limit. The proposed change to WBN Units 1 and 2 TS 3.7.15, 3.7.18, 3.9.9, and 4.3 change the limit on minimum boron concentration in the spent fuel pool to be consistent with an updated criticality analysis for the spent fuel storage racks when fuel is stored in the pool.

The proposed change incorporates a new program as TS 5.7.2.21 to monitor the condition of the neutron absorber material used in the spent fuel pool storage racks to ensure it will continue to perform its assumed design functions. SFP storage rack neutron absorber monitoring is an administrative requirement that does not affect the ability of any SSCs to perform a design function. No safety limits are affected. No Limiting Conditions for Operation or Surveillance limits are affected. The new Technical Specification requirements assure sufficient criticality safety margins are maintained. The proposed change does not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. As such, there are no changes being made to safety analysis assumptions, safety limits, or limiting safety system settings that would adversely affect plant safety.

Accordingly, the proposed changes do not involve a significant reduction in a margin of safety.

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 Tennessee Valley Authority's (TVA's) Watts Bar Nuclear Plant (WBN), Unit 1 were initially assessed in a 1999 "Final Environmental Impact Statement (EIS) for the Production of Tritium in a Commercial Light Water Reactors" (DOE/EIS0288) prepared by the Department of Energy (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 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 the 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.[1] 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

1 See https://nnsa.energy.gov/aboutus/ouroperations/generalcounsel/nepaoverview/nepa/tritiumseis

requirements. 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 TPBAR irradiation for facility workers or the public. For all analyzed alternatives (including TPBAR irradiation at both Watts Bar Nuclear Plant units), estimated radiation exposures would remain well below regulatory limits. In accordance with 40 CFR 1506.3(c) of the Council on Environmental Quality regulations, TVA independently reviewed the EIS prepared by DOE, found it to be adequate, and adopted the EIS, as published in the Federal Register at 81 FR 11557 (March 4, 2016). TVA's Record of Decision on Production of Tritium in a Commercial Light Water Reactors (including both Watts Bar Nuclear Plant units) was published in the Federal Register at 82 FR 16653 (April 5, 2017).[2]

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 21). NRC used this information in their Environmental Assessment and Finding of No Significant Impact for WBN Unit 1 License Amendment 40 (Reference 64).

TVA also provided updated information to NRC regarding the environmental impacts associated with tritium production from as many as 1,792 TPBARs to support WBN Unit 1 License Amendment 107 (Reference 42). NRC used this information in their Environmental Assessment and Finding of No Significant Impact for WBN Unit 1 License Amendment 107 (Reference 65).

Sections 4.1.5 and 4.1.14 provide updates to this information to address the environmental effects of TPBAR irradiation at both Watts Bar Nuclear Plant units. Compliance with 10 CFR 50.68, as described in Section 4.1.10, ensures that the Technical Specification changes involving the spent fuel rack have no environmental impacts.

Based on the 1999 EIS and 2016 SEIS prepared by the DOE, the information provided to NRC (References 21 and 42) for WBN Unit 1 License Amendments 40 and 107 and the corresponding NRC Environmental Assessments and Findings of No Significant Impact (References 62 and 65), and the updated evaluation information provided for this proposed amendment in Sections 4.1.5 and 4.1.14, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a 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.

[2] See https://www.tva.gov/Environment/Environmental-Stewardship/Environmental-Reviews/Production-of-Tritium-in-a-Commercial-Light-Water-Reactor

7.0 REFERENCES

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)
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)
3. NRC letter to TVA, "Issuance of Amendment on Tritium Producing Burnable Absorber Lead Test Assemblies (TAC No. M98615)," dated September 15, 1997 (ML020780128)
4. NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Revision of Boron Requirements for Cold Leg Accumulators and Refueling Water Storage Tank (TAC No. MB9480)," dated October 8, 2003 (ML032880062)
5. NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding the Maximum Number of Tritium Producing Burnable Assembly Rods in the Reactor Core (TAC No. MD5430)," dated January 18, 2008 (ML073520546)
6. NRC letter to TVA, "Watts Bar Nuclear Plant, Unit 1 – Issuance of Amendment Regarding the Maximum Number of Tritium Producing Burnable Assembly Rods in the Reactor Core (TAC No. MD9396)," dated May 4, 2009 (ML090920506)
7. NDP-98-181, Revision 1, "Tritium Production Core (TPC) Topical Report," Westinghouse Electric Company, February 8, 1999," (ML16077A093)
8. NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report on the Tritium Production Core," May 1999 (9907210095).
9. TVA Letter to NRC, “Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores -Technical Specification Change No. TVA-WBN-TS-00-015” dated August 20, 2001 (ML012390106 and ML012390115)
10. TVA Letter to NRC, “Responses to Request for Additional Information (RAI) Regarding Tritium Production Interface Item Numbers 1, 6, 7, 10, 11, and 12,” dated October 29, 2001 (ML020320146)
11. TVA Letter to NRC, “Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Generic Letter (GL) 81-07 - NUREG-0612 - Control of Heavy Loads at Nuclear Power Plants – Revised Response - License Condition (LC) 39 - (TACc Nos. M77560 and M77561)” dated July 28, 1993 (ML073230481)”
12. NUREG-0847, Supplement 13, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Units 1 and 2,” April 1994 (ML072060484)
13. NUREG-0847, Supplement 22, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2,” February 2011 (ML110390197)
14. TVA Letter to NRC, “Request For Additional Information (RAI) Regarding Tritium Production -Interface Issue Number 2 -Procurement and Fabrication Issues” dated April 5, 2002 (ML021000361)
15. TVA Letter to NRC, “Responses to Request for Additional Information (RAI) Regarding Tritium Production Interface Item Number 3 - Compliance with DNB Criterion” dated September 21, 2001 (ML012670465)
16. TVA Letter to NRC, “Responses to Request for Additional Information (RAI) Regarding Tritium Production Interface Item Number 3 - Compliance with DNB Criterion Clarification to Previous RAI Response dated September 21, 2001” dated December 19, 2001 (ML013620072)

17. 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)
18. TVA Letter to NRC, CNL-15-232, “Application to Revise Technical Specification 4.2.1, “Fuel Assemblies” (WBN-TS-15-03) - Response to NRC Request for Additional Information - Nuclear Performance and Code Review Branch” dated November 30, 2015 (ML15335A468)
19. WCAP-15306-NP-A, “VIPRE-01 Modeling Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” October 1999 (ML993160096)
20. NRC Letter to TVA, “Request For Additional Information (RAI) Regarding Tritium Production -Interface Issue Number 5 -Control Room Habitability Systems” dated May 21, 2002(ML021440135)
21. NRC Letter to TVA, “Request For Additional Information (RAI) Regarding Radiological Impact” dated May 23, 2002 (ML021490139)
22. NUREG-0847, Supplement 25, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2,” December 2011 (ML12011A024)
23. 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)
24. TVA Letter to NRC, CNL-15-192, “Application to Revise Technical Specification 4.2.1, "Fuel Assemblies" (WBN-TS-15-03) - Response to NRC Request for Additional Information - Radiation Protection and Consequence Branch,” dated September 25, 2015(ML15268A568)
25. NUREG-0847, "Safety Evaluation Report related to the operation of Watts Bar Nuclear Plant, Units 1 and 2," June 1982 (ML072060490)
26. TVA Letter to NRC, “Responses to RAI Regarding Tritium Producing Burnable Absorber Rods (TPBARs),” dated February 21, 2002 (ML020580005)
27. TVA Letter to NRC, “Request for Additional Information Regarding Tritium Production - Interface Issue Number 7 - Light Load Handling Systems,” dated May 21, 2002 (ML021440142)
28. TVA Letter to NRC, “Revision to the Spent Fuel Pool Cooling Analysis Methodology,” dated April 20, 2001 (ML011170181)
29. TVA Letter to NRC, “Watts Bar Nuclear Plant -Responses to RAI Regarding Spent Fuel Pool Cooling Analysis Methodology” dated November 14, 2001 (ML020150176)
30. NRC Letter to TVA, “Issuance of Amendment Regarding Spent Fuel Pool Cooling Analysis Methodology Change” dated February 21, 2002 (ML020580612)
31. NUREG-0847, Supplement 23, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2,” July 2011 (ML11206A499)
32. TVA Letter to NRC, “Request for Additional Information Regarding Tritium Production -Interface Issues 1, 6, 7, 10, 11, and 12 - Holtec Analysis” dated November 21, 2001 (ML020170164 and ML020170174)
33. TVA Letter to NRC, “Request for Additional Information Regarding Tritium Production - Holtec Analysis” dated February 19, 2002 (ML020580082)
34. TVA Letter to NRC, “Watts Bar Nuclear Plant - Unit 1 - Technical Specification Change No. 98-002 - Changes to New Fuel Vault Storage Requirements for Increased Fuel Enrichment” dated May 6, 1998 (ML073201046)
35. TVA Letter to NRC, “Watts Bar Nuclear Plant - Unit 1 - Technical Specification Change No. 98-002 - Changes to New Fuel Vault Storage Requirements for Increased Fuel Enrichment” dated June 5, 1998 (ML073201051)

36. NRC Letter to TVA, “Issuance of Amendment on Increase of Enrichment Limit to Five Percent for New Fuel Storage” dated December 1, 1998 (ML020780171)
37. TVA Letter to NRC, “Watts Bar Nuclear Plant (WBN) - Unit 2 - Final Safety Analysis Report (FSAR), Section 9.1, "Fuel Storage and Handling" dated December 21, 2010 (ML103610285)
38. NUREG-0847, Supplement 26, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2,” June 2013 (ML13205A136)
39. TVA Letter to NRC, “Responses To RAI Regarding Tritium Production -Interface Issues 14 and 15” dated December 7, 2001 (ML013520461)
40. 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)
41. 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)
42. TVA Letter to NRC, CNL-16-038, “Application to Revise Technical Specification 4.2.1, "Fuel Assemblies" (WBN-TS-15-03) – 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)
43. TVA Letter 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)
44. NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors" (ML112720411)
45. TVA Letter to NRC, "Groundwater Protection - Data Collection Questionnaire for Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants" dated August 4, 2006 (ML062280598).
46. NEI 07-07, "Ground Water Protection Initiative," (ML072610036)
47. TVA letter to NRC, "Watts Bar Nuclear Plant, Annual Radioactive Effluent Release Report - 2015" dated May 1, 2016 (ML15117A395).
48. TVA Letter, “Sequoyah (SQN) and Watts Bar (WBN) Nuclear Plants - Tritium Program” dated June 23, 2000 (ML003727549)
49. TVA Letter, “Sequoyah (SQN) and Watts Bar (WBN) Nuclear Plants - Tritium Program” dated October 5, 2000 (ML012210085)
50. NRC Letter to TVA, “Safety Evaluation of LOCTA_JR CODE for Loss-of-Coolant Accident Analysis of Fuel Rods” dated January 17, 2001 (ML010170152)
51. TVA Letter to NRC, “Tritium Production Program Anticipated Transients Without Scram (ATWS)” dated September 29, 2000 (ML003759282)
52. NRC Letter to TVA, “Tritium Production Program - NUREG-1672 Interface Issue 17 - Anticipated Transient Without Scram Analyses” dated March 16, 2001 (ML010750049)
53. WCAP-15831-NP-A, Revision 2, “WOG Risk-Informed ATWS Assessment and Licensing Implementation Process,” August 2007 (ML072550560)
54. NUREG-0847, Supplement 24, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant Unit 2,” September 2011 (ML11277A148)
55. TVA Letter to NRC, CNL-15-172, “Application to Revise Technical Specification 4.2.1, "Fuel Assemblies" (WBN-TS-15-03) - Response to NRC Request for

Additional Information - Reactor Systems Branch" - Responses to RAI 4," dated September 14, 2015 (ML15258A204)
56. NRC Letter to TVA, "Watts Bar Nuclear Plant, Unit 1 - Audit Report Related to License Amendment Request to Revise Technical Specification 4.2.1, Fuel Assemblies (CAC No. MF6050)," dated December 23, 2015 (ML15345A424)
57. TVA Letter to NRC, CNL-15-263, "Supplemental Information Related to the Onsite Regulatory Audit at Pacific Northwest National Laboratory" - Response to Audit Open Item #2," dated December 29, 2015 (ML16004A161)
58. TVA Letter to NRC, CNL-15-092, "Response to NRC Request to Supplement the Application to Revise Technical Specification 4.2.1, "Fuel Assemblies" (WBN-TS-15-03)" dated May 27, 2015 (ML15147A611)
59. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
60. NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," (ML14149A165)
61. NEI 12-16, Revision 2 Draft B, Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants (ML17009A343)
62. NEI letter to NRC, "Submittal of NEI 16-03, Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools, Revision 0, dated August 2016," dated May 26, 2017 (ML17263A133)
63. NRC letter to NEI, "Final Safety Evaluation for Nuclear Energy Institute Topical Report NEI 16-03 - Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools (CAC NO. MF8122)," dated March 3, 2017 (ML16354A486)
64. 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).
65. 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).

ATTACHMENT 1

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

1. Affected TS Pages
   TS 3.7-31
   TS 3.7-39
   TS 3.7-40
   TS 3.9-16
   TS 4.0-2
   TS 4.0-3
   TS 4.0-9
   TS 4.0-10
   TS 5.0-25a

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Assembly Storage

LCO 3.7.15 The ~~combination of~~ initial enrichment ~~and burnup~~ of each ~~spent~~ fuel assembly stored shall be in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------------NOTE----------------- LCO 3.0.3 is not applicable. -- Initiate action to move the noncomplying fuel assembly.	 Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment ~~and burnup~~ of the fuel assembly is in accordance with Specification 4.3.1.1.	Prior to storing the fuel assembly.

3.7 PLANT SYSTEMS

3.7.18 Fuel Storage Pool Boron Concentration

LCO 3.7.18 The fuel storage pool boron concentration shall be ≥ 2300 ppm

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. -- A.1 Initiate action to restore fuel storage pool boron concentration to within limit.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify the fuel storage pool boron concentration is within limit.	72 hours

3.9 REFUELING OPERATIONS

3.9.9 Spent Fuel Pool Boron Concentration

LCO 3.9.9 Boron concentration of the spent fuel pool shall be ≥ ~~2000~~ 2300 ppm.

APPLICABILITY: ~~During fuel movement~~Whenever any fuel assembly is stored in the flooded spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 ~~Suspend fuel movement.~~Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.9.1	Verify boron concentration in the spent fuel pool is ≥ ~~2000~~ 2300 ppm.	~~Prior to movement of fuel in the spent fuel pool~~ ~~AND~~ 72 hours ~~thereafter~~

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent (wt%) (nominally 4.95 ± 0.05 wt% U-235);

b. $k_{eff} \leq 0.95$ if fully flooded with ~~unborated~~ 2300 ppm borated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR, and a k_{eff} less than critical when flooded with unborated water;

c. Distances between fuel assemblies are a nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules.

~~d. Fuel assemblies with initial enrichments less than a maximum of 5 wt% U-235 enrichment (nominally 4.95± 0.05 wt% U235) may be stored in the spent fuel racks in any one of four arrangements with specific limits as identified below:~~

~~1. Fuel assemblies may be stored in the racks in an all cell arrangement provided the burnup of each assembly is in the acceptable domain identified in Figure 4.3-3, depending upon the specified initial enrichment.~~

~~2. New and spent fuel assemblies may be stored in a checkerboard arrangement of 2 new and 2 spent assemblies, provided that each spent fuel assembly has accumulated a minimum burnup in the acceptable domain identified in Figure 4.3-4.~~

~~3. New fuel assemblies may be stored in 4-cell arrays with 1 of the 4 cells remaining empty of fuel (i.e. containing only water or water with up to 75 percent by volume of non-fuel bearing material).~~

(continued)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage (continued)

~~4. New fuel assemblies with a minimum of 32 integral fuel burnable absorber (IFBA) rods may be stored without further restriction, provided the loading of ZrB_2 in the coating of each IFBA rod is minimum of 1.25x (1.9625mg/in).~~

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements. A water cell is defined as a cell containing water or non-fissile material ~~with no more than 75 percent of the water displaced~~.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum enrichment of 5.0 weight percent U-235 and shall be maintained with the arrangement of 120 storage locations shown in Figure 4.3-2;

b. $k_{eff} \leq$ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq$ 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

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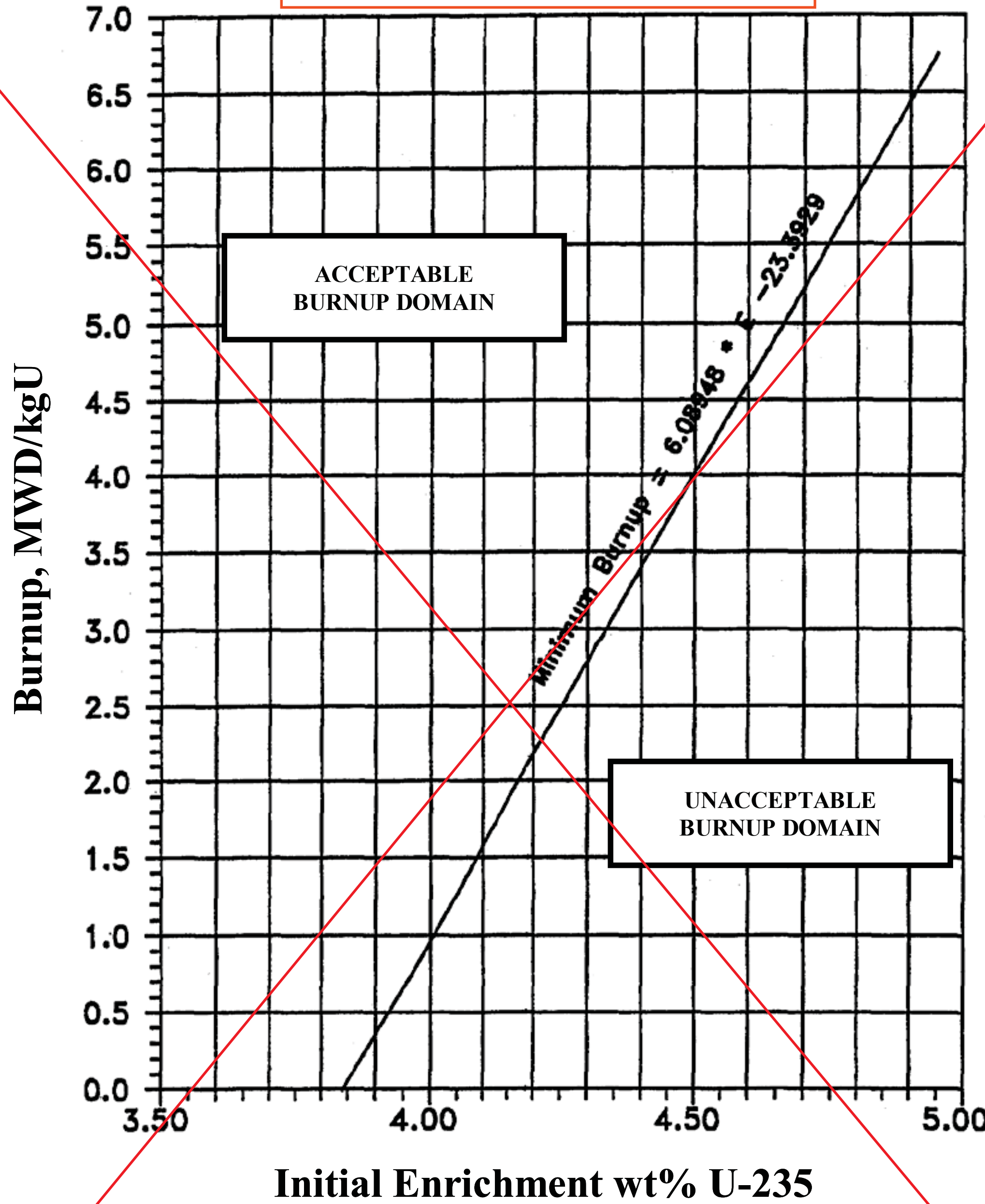

FIGURE 4.3-3
MINIMUM REQUIRED BURNUP FOR UNRESTRICTED STORAGE OF FUEL OF VARIOUS INITIAL ENRICHMENTS

(continued)

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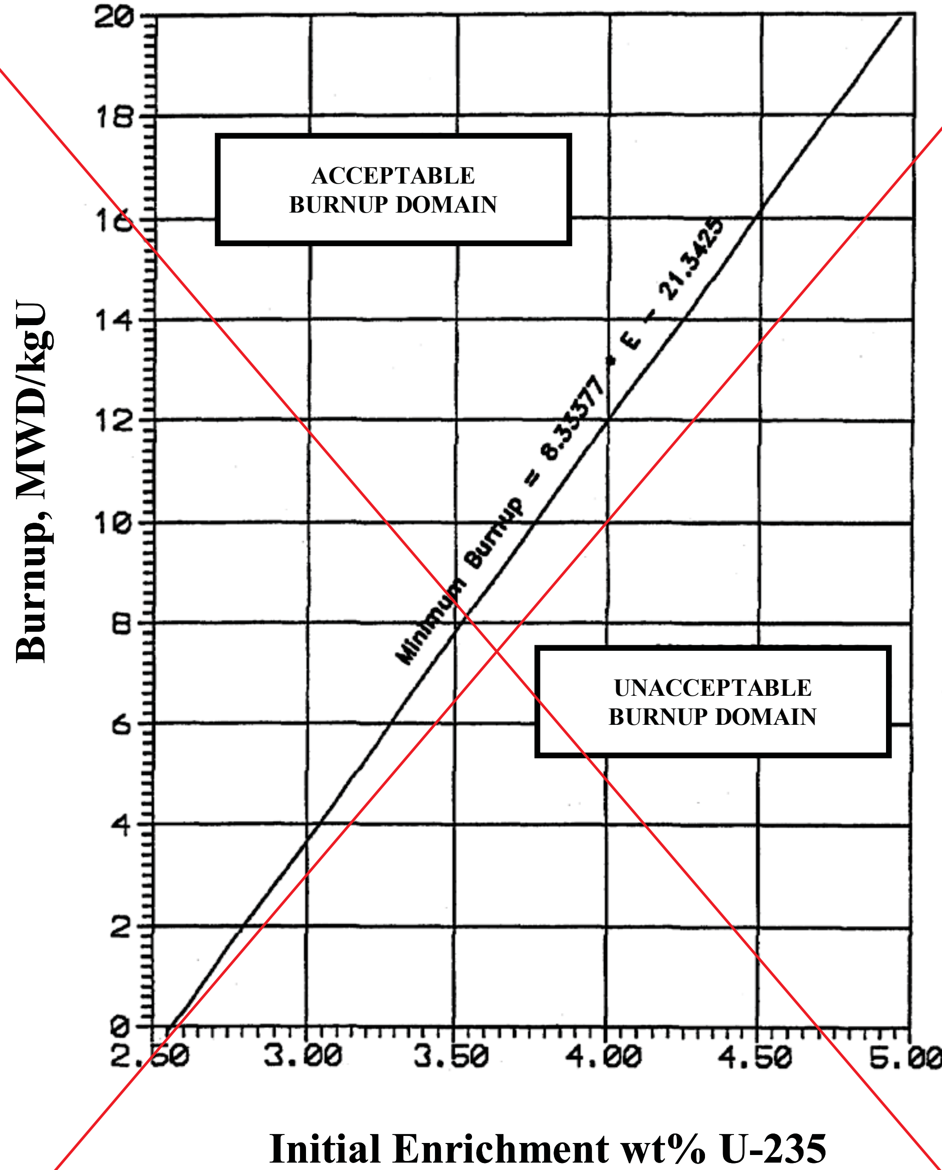

FIGURE 4.3-4
MINIMUM REQUIRED BURNUP FOR A CHECKERBOARD ARRANGEMENT OF 2 SPENT AND 2 NEW FUEL ASSEMBLIES OF 5wt% U-235 ENRICHMENT (MAXIMUM)

5.7 Procedures, Programs, and Manuals

5.7.2.20 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate defined in the Ventilation Filter Testing Program (VFTP), at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.7.2.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the neutron absorber density is consistent with the assumptions in the spent fuel pool criticality analysis. The Program shall be in accordance with NEI 16-03-A, “Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools,” Revision 0, May 2017.

ATTACHMENT 2

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

1. Affected TS Pages
   TS 3.7-30
   TS 3.7-38
   TS 3.7-39
   TS 3.9-12
   TS 4.0-1
   TS 4.0-2
   TS 4.0-3
   TS 4.0-8
   TS 4.0-9
   TS 5.0-27

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Assembly Storage

LCO 3.7.15 The ~~combination of~~ initial enrichment ~~and burnup~~ of each ~~spent~~ fuel assembly stored shall be in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ------------NOTE------------- LCO 3.0.3 is not applicable. ---------------------------------- Initiate action to move the noncomplying fuel assembly.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment ~~and burnup~~ of the fuel assembly is in accordance with Specification 4.3.1.1.	Prior to storing the fuel assembly.

3.7 PLANT SYSTEMS

3.7.18 Fuel Storage Pool Boron Concentration

LCO 3.7.18 The fuel storage pool boron concentration shall be ≥ 2300 ppm

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. --	
	A.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify the fuel storage pool boron concentration is within limit.	72 hours

3.9 REFUELING OPERATIONS

3.9.9 Spent Fuel Pool Boron Concentration

LCO 3.9.9 Boron concentration of the spent fuel pool shall be ≥ ~~2000~~ 2300 ppm.

APPLICABILITY: ~~During fuel movement~~Whenever any fuel assembly is stored in the flooded spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 ~~Suspend fuel movement.~~Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.9.1	Verify boron concentration in the spent fuel pool is ≥ ~~2000~~ 2300 ppm.	~~Prior to movement of fuel in the spent fuel pool~~ ~~AND~~ 72 hours ~~thereafter~~

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

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 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 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)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent (wt%) (nominally 4.95 ± 0.05 wt% U-235);

b. $k_{eff} \leq 0.95$ if fully flooded with ~~unborated~~ 2300 ppm borated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR, and a k_{eff} less than critical when flooded with unborated water;

c. Distances between fuel assemblies are a nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules.

d. ~~Fuel assemblies with initial enrichments less than a maximum of 5 wt% U-235 enrichment (nominally 4.95 ± 0.05 wt% U-235) may be stored in the spent fuel racks in any one of four arrangements with specific limits as identified below:~~

~~1. Fuel assemblies may be stored in the racks in an all cell arrangement provided the burnup of each assembly is in the acceptable domain identified in Figure 4.3-3, depending upon the specified initial enrichment.~~

~~2. New and spent fuel assemblies may be stored in a checkerboard arrangement of 2 new and 2 spent assemblies, provided that each spent fuel assembly has accumulated a minimum burnup in the acceptable domain identified in Figure 4.3-4.~~

~~3. New fuel assemblies may be stored in 4-cell arrays with 1 of the 4 cells remaining empty of fuel (i.e. containing only water or water with up to 75 percent by volume of non-fuel bearing material).~~

(continued)

4.3 Fuel Storage (continued)

~~4. New fuel assemblies with a minimum of 32 integral fuel burnable absorber (IFBA) rods may be stored without further restriction, provided the loading of ZrB_2 in the coating of each IFBA rod is minimum of 1.25x (1.9625mg/in).~~

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements. A water cell is defined as a cell containing water or non-fissile material ~~with no more than 75 percent of the water displaced~~.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum enrichment of 5.0 weight percent U-235 and shall be maintained with the arrangement of 120 storage locations shown in Figure 4.3-2;

b. $k_{eff} \leq$ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq$ 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Elevation 747 feet - 1 1/2 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in 24 flux trap rack modules.

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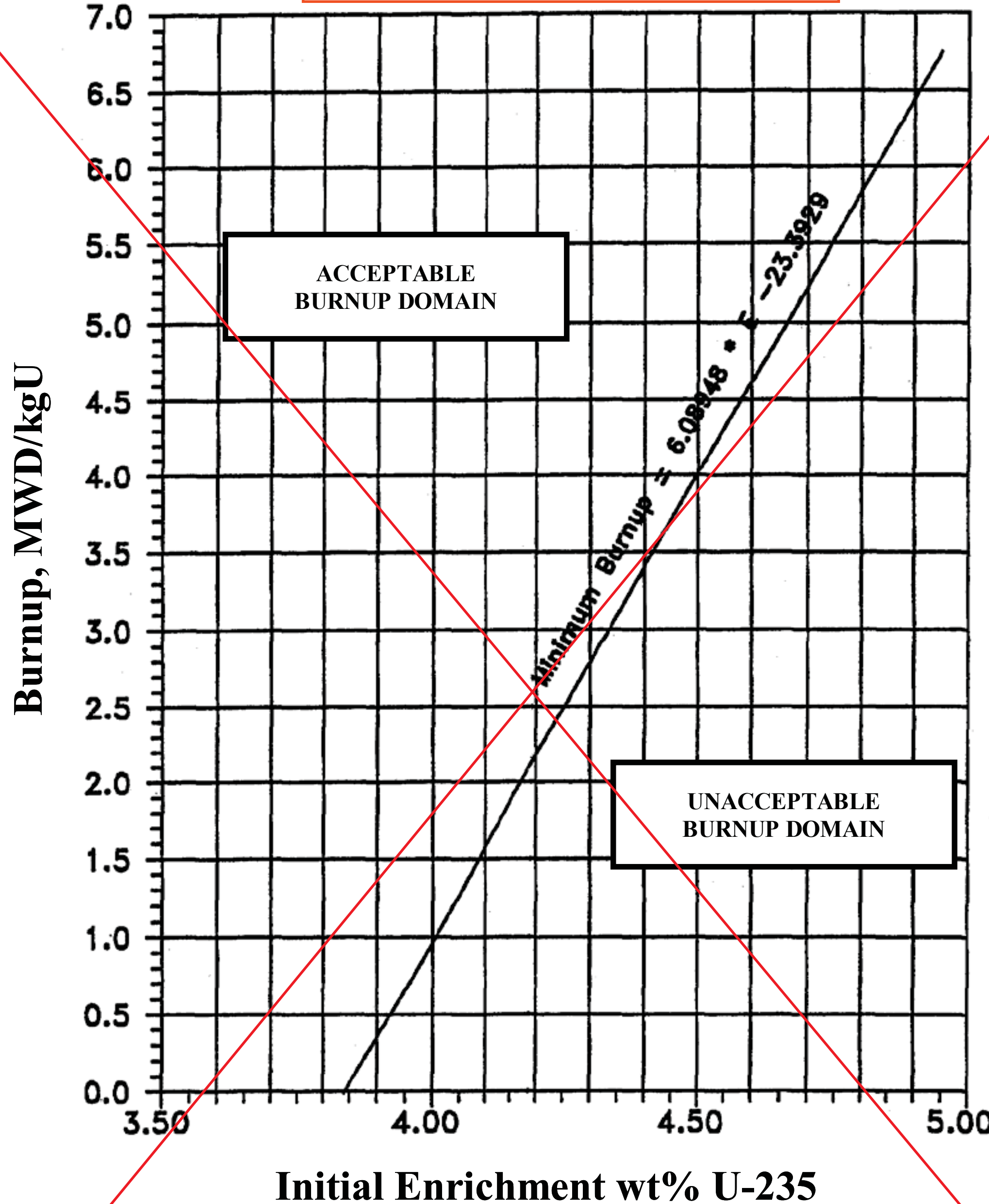

FIGURE 4.3-3
MINIMUM REQUIRED BURNUP FOR UNRESTRICTED STORAGE
OF FUEL OF VARIOUS INITIAL ENRICHMENTS

(continued)

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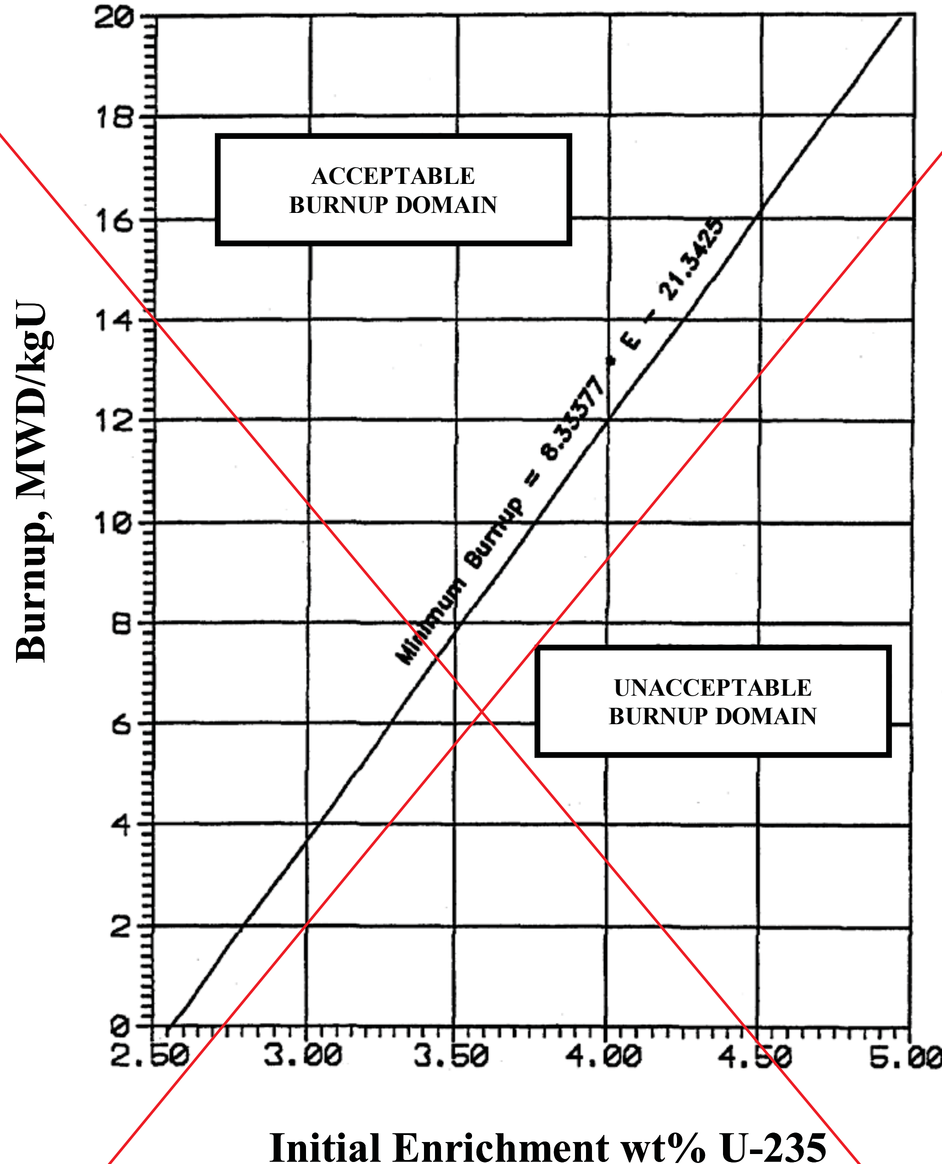

FIGURE 4.3-4
MINIMUM REQUIRED BURNUP FOR A CHECKERBOARD ARRANGEMENT OF 2 SPENT AND 2 NEW FUEL ASSEMBLIES OF 5 wt% U-235 ENRICHMENT (MAXIMUM)

5.7 Procedures, Programs, and Manuals

5.7.2.20 Control Room Envelope Habitability Program (continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate defined in the Ventilation Filter Testing Program (VFTP), at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.7.2.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the neutron absorber density is consistent with the assumptions in the spent fuel pool criticality analysis. The Program shall be in accordance with NEI 16-03-A, “Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools,” Revision 0, May 2017.

ATTACHMENT 3

Proposed TS Changes (Final Typed) for WBN Unit 1

1. Affected TS Pages
   TS 3.7-31
   TS 3.7-39
   TS 3.7-40
   TS 3.9-16
   TS 4.0-2
   TS 4.0-3
   TS 5.0-25a

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Assembly Storage

LCO 3.7.15 The initial enrichment of each fuel assembly stored shall be in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------------NOTE----------------- LCO 3.0.3 is not applicable. -- Initiate action to move the noncomplying fuel assembly.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment of the fuel assembly is in accordance with Specification 4.3.1.1.	Prior to storing the fuel assembly.

3.7 PLANT SYSTEMS

3.7.18 Fuel Storage Pool Boron Concentration

LCO 3.7.18 The fuel storage pool boron concentration shall be ≥ 2300 ppm

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. -- A.1 Initiate action to restore fuel storage pool boron concentration to within limit.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify the fuel storage pool boron concentration is within limit.	72 hours

3.9 REFUELING OPERATIONS

3.9.9 Spent Fuel Pool Boron Concentration

LCO 3.9.9 Boron concentration of the spent fuel pool shall be ≥ 2300 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the flooded spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.9.1	Verify boron concentration in the spent fuel pool is ≥ 2300 ppm.	72 hours

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent (wt%) (nominally 4.95 ± 0.05 wt% U-235);

b. $k_{eff} \leq 0.95$ if fully flooded with 2300 ppm borated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR, and a k_{eff} less than critical when flooded with unborated water;

c. Distances between fuel assemblies are a nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules.

(continued)

4.3 Fuel Storage (continued)

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements. A water cell is defined as a cell containing water or non-fissile material.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum enrichment of 5.0 weight percent U-235 and shall be maintained with the arrangement of 120 storage locations shown in Figure 4.3-2;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.20 Control Room Envelope Habitability Program (continued)

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate defined in the Ventilation Filter Testing Program (VFTP), at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.7.2.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the neutron absorber density is consistent with the assumptions in the spent fuel pool criticality analysis. The Program shall be in accordance with NEI 16-03-A, “Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools,” Revision 0, May 2017.

ATTACHMENT 4

Proposed TS Changes (Final Typed) for WBN Unit 2

1. Affected TS Pages
   TS 3.7-30
   TS 3.7-38
   TS 3.7-39
   TS 3.9-12
   TS 4.0-1
   TS 4.0-2
   TS 4.0-3
   TS 5.0-27

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Assembly Storage

LCO 3.7.15 The initial enrichment of each fuel assembly stored shall be in accordance with Specification 4.3.1.1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	------------NOTE------------- LCO 3.0.3 is not applicable. ---------------------------------- Initiate action to move the noncomplying fuel assembly.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment of the fuel assembly is in accordance with Specification 4.3.1.1.	Prior to storing the fuel assembly.

3.7 PLANT SYSTEMS

3.7.18 Fuel Storage Pool Boron Concentration

LCO 3.7.18 The fuel storage pool boron concentration shall be ≥ 2300 ppm

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	--------------------NOTE-------------------- LCO 3.0.3 is not applicable. --- A.1 Initiate action to restore fuel storage pool boron concentration to within limit.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify the fuel storage pool boron concentration is within limit.	72 hours

3.9 REFUELING OPERATIONS

3.9.9 Spent Fuel Pool Boron Concentration

LCO 3.9.9 Boron concentration of the spent fuel pool shall be ≥ 2300 ppm.

APPLICABILITY: Whenever any fuel assembly is stored in the flooded spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.9.1	Verify boron concentration in the spent fuel pool is ≥ 2300 ppm.	72 hours

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries

The site and exclusion area boundaries shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone (LPZ)

The LPZ shall be as shown in Figure 4.1-2 (within the 3-mile circle).

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 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 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)

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks (shown in Figure 4.3-1) are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent (wt%) (nominally 4.95 ± 0.05 wt% U-235);

b. $k_{eff} \leq 0.95$ if fully flooded with 2300 ppm borated water, which, includes an allowance for uncertainties as described in Sections 4.3.2.7 and 9.1 of the FSAR, and a k_{eff} less than critical when flooded with unborated water;

c. Distances between fuel assemblies are a nominal 10.375 inch center-to-center spacing in the twenty-four flux trap rack modules.

(continued)

4.3 Fuel Storage (continued)

A water cell is less reactive than any cell containing fuel and therefore a water cell may be used at any location in the loading arrangements. A water cell is defined as a cell containing water or non-fissile material.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum enrichment of 5.0 weight percent U-235 and shall be maintained with the arrangement of 120 storage locations shown in Figure 4.3-2;

b. $k_{eff} \leq$ 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR;

c. $k_{eff} \leq$ 0.98 if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below Elevation 747 feet - 1 1/2 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1386 fuel assemblies in 24 flux trap rack modules.

5.7.2.20 Control Room Envelope Habitability Program (continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate defined in the Ventilation Filter Testing Program (VFTP), at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.7.2.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This Program provides controls for monitoring the condition of the neutron absorber used in the spent fuel pool storage racks to verify the neutron absorber density is consistent with the assumptions in the spent fuel pool criticality analysis. The Program shall be in accordance with NEI 16-03-A, “Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools,” Revision 0, May 2017.

ATTACHMENT 5

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

(for information only)

1. Affected TS Bases Pages
   B 3.7-75
   B 3.7-76
   B 3.7-77
   B 3.7-91
   B 3.7-92
   B 3.7-93
   B 3.9-33
   B 3.9-34

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Assembly Storage

BASES

BACKGROUND

The spent fuel pool contains flux trap rack modules with 1386 storage positions ~~and~~ that are designed to accommodate fuel with a maximum enrichment ~~as high as~~of ~~3.8~~4.95 ± 0.05 weight percent U-235 without restrictions. ~~Storage of fuel assemblies with enrichment between 3.8 and 5.0 weight percent requires either fuel burnup in accordance with paragraph 4.3.1.1 or placement in storage locations which have face adjacent storage cells containing either water or fuel assemblies with accumulated burnup of at least 20.0 MWD/KgU in accordance with Specification 4.3.1.1.~~

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. ~~However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains the storage racks in a subcritical condition during normal operation with the racks fully loaded.~~ The double contingency principle discussed in ANSI N-16.1-1975, and the April 1978 NRC letter (Reference 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. ~~For example, an abnormal scenario could be associated with the improper~~

(continued)

BASES

BACKGROUND (continued)

~~loading of a relatively high enrichment, low exposure fuel assembly. This could potentially increase the criticality of the storage racks.~~ To mitigate ~~these~~ postulated criticality-related events, boron is dissolved in the pool water. ~~Safe the pool, it is necessary to perform SR 3.9.9.1.~~

APPLICABLE SAFETY ANALYSES

~~The hypothetical events can only take place during or as a result of the movement of an assembly. For these occurrences, the presence of soluble boron in the spent fuel storage pool, (controlled by LCO 3.9.9, "Spent Fuel Pool Boron Concentration,") prevents criticality in the storage rack regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential occurrences may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, the operation may be under the auspices of the accompanying LCO.~~ The accident analyses are provided in the FSAR.

The ~~configuration~~ initial enrichment of fuel assemblies in the fuel storage pool along with the concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the ~~NRC Policy Statement~~10 CFR 50.36(c)(2)(ii).

LCO

The restrictions on the ~~placement~~ initial enrichment of fuel assemblies within the spent fuel pool in accordance with ~~paragraph~~ Specification 4.3.1.1 in the accompanying LCO, ensures the k_{eff} will always remain ~~≤0.95~~subcritical, assuming the pool to be flooded with unborated water.

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.

(continued)

BASES (continued)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

If unable to move irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in Mode 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the ~~configuration~~ initial enrichment of fuel assemblies stored in the spent fuel storage pool is not in accordance with Specification 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movements to bring the configuration into compliance with Specification 4.3.1.1.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment ~~and burnup~~ of the fuel assembly is in accordance with Specification 4.3.1.1 in the accompanying LCO.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. FSAR Section 4.3.2.7

B 3.7 PLANT SYSTEMS

B 3.7.18 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND	In the BORAL™ flux trap rack design, the spent fuel storage pool is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt % U-235, or spent fuel regardless of the discharge fuel burnup. The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. Analysis demonstrates that the effective neutron multiplication factor (k_{eff}) of the spent fuel pool loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with unborated water, and does not exceed 0.95 for the pool flooded with borated water with 500 ppm soluble boron (an additional 50 ppm of soluble boron has been added to account for grid growth.) The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related events, boron is dissolved in the pool water.
APPLICABLE SAFETY ANALYSES	The following accident conditions have been evaluated: • The effect of SFP temperature exceeding the normal range • A dropped fuel assembly • A misloaded fuel assembly (a fuel assembly in the wrong location within the storage rack) • A mislocated fuel assembly (a fuel assembly in the wrong location outside the storage rack) • Rack movement due to seismic activity The results of the evaluation show that a soluble boron concentration of 500 ppm is sufficient to ensure that the maximum k_{eff} is below the regulatory limit of 0.95. The boron dilution analysis assumes an initial boron concentration of 2,300 ppm for the limiting evaluation. The accident analyses are provided in the FSAR, Section 4.3.2.7 (Ref. 2) The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The fuel storage pool boron concentration is required to be ≥2300 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

A.1

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. However, prior to resuming movement of the fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.7.18.1 This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 72 hour Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.
REFERENCES	1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A) 2. FSAR, Section 4.3.2.7

B 3.9 REFUELING OPERATIONS

B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND	The spent fuel storage rack criticality analysis assumes ~~2000~~ 2300 ppm soluble boron in the fuel pool ~~during a dropped/misplaced fuel assembly event~~when fuel is being stored.
APPLICABLE SAFETY ANALYSES	This requirement ensures the presence of at least ~~2000~~ 2300 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for normal storage and a dropped~~/misplaced~~ fuel assembly event. The RCS boron concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to ~~2000~~ 2300 ppm ~~during fuel movement~~ anytime fuel is being stored in the pool.
APPLICABILITY	This LCO is applicable when the spent fuel pool is flooded and fuel is ~~being moved~~in the pool. ~~Once fuel movement begins, the movement is considered in progress until the configuration of the assemblies in the storage racks~~The assembly is verified to comply with the criticality loading criteria specified in Specification 4.3.1.1 before placing it in the Spent Fuel Pool.
ACTIONS	A.1 If the spent fuel pool boron concentration does not meet the above requirements, ~~fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis.~~action must be initiated to restore fuel storage pool boron concentration to within limits. Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.9.1

This SR requires that the spent fuel pool boron concentration be verified greater than or equal to ~~2000~~ 2300 ppm. This surveillance is to be performed ~~prior to movement of fuel~~ when fuel is stored in the spent fuel pool and at least once every 72 hours thereafter ~~during the movement of fuel in the spent fuel pool~~.

The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the sample. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Watts Bar FSAR, Section ~~15, "Accident Analysis."~~4.3.2.7

ATTACHMENT 6

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

(for information only)

1. Affected TS Bases Pages
   B 3.7-74
   B 3.7-75
   B 3.7-76
   B 3.7-89
   B 3.7-90
   B 3.7-91
   B 3.9-24
   B 3.9-25

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Pool Assembly Storage

BASES

BACKGROUND

The spent fuel pool contains flux trap rack modules with 1386 storage positions that are designed to accommodate ~~new~~ fuel with a maximum enrichment of 4.95 ± 0.05 weight percent U-235 ~~and fuel of various initial enrichments when stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~without restrictions.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. The double contingency principle discussed in ANSI N-16.1-1975, and the April 1978 NRC letter (Reference 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time.~~However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design is based on the use of unborated water, which maintains the storage racks in a subcritical condition during normal operation with the racks fully loaded. The double contingency principle discussed and implemented in Reference 2, was based on the Proposed Revision 2 of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," specifically Appendix A, "Nuclear Criticality Safety", section 1.4, which states;~~ ~~*At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures or operating limit violations.*~~ ~~ANSI N-16.1-1975, and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, an abnormal scenario could be associated with the improper loading of a relatively high enrichment, low exposure fuel assembly. This could potentially increase the criticality of the storage racks.~~ To mitigate ~~these~~ postulated criticality-related events, boron is dissolved in the pool water. ~~Safe operation of the spent fuel storage design with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO. Prior to movement of an assembly in the pool, it is necessary to perform SR 3.9.9.1.~~

BASES (continued)

APPLICABLE SAFETY ANALYSES	~~The hypothetical events can only take place during or as a result of the movement of an assembly. For these occurrences, the presence of soluble boron in the spent fuel storage pool, (controlled by LCO 3.9.9, "Spent Fuel Pool Boron Concentration") prevents criticality in the storage rack regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential occurrences may be limited to a small fraction of the total operating time. During the remaining time period with no potential for such events, the operation may be under the auspices of the accompanying LCO.~~ The accident analyses are provided in the FSAR. The ~~configuration~~ initial enrichment of fuel assemblies in the fuel storage pool along with the concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The restrictions on the ~~placement~~ initial enrichment of fuel assemblies within the spent fuel pool in accordance with Specification 4.3.1.1 in the accompanying LCO, ensures the k_{eff} will always remain ~~≤ 0.95~~ subcritical, assuming the pool to be flooded with unborated water.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel storage pool.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown. When the ~~configuration~~ initial enrichment of fuel assemblies stored in the spent fuel storage pool is not in accordance with Specification 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movements to bring the configuration into compliance with Specification 4.3.1.1.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR verifies by administrative means that the initial enrichment ~~and burnup~~ of the fuel assembly is in accordance with Specification 4.3.1.1 in the accompanying LCO.

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. ~~TVA, Document Number: PFE-R07, "Criticality Analysis Summary Report Watts Bar Nuclear Plant (WBN)," Revision 0, October 1996.~~FSAR Section 4.3.2.7

B 3.7 PLANT SYSTEMS

B 3.7.18 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

In the BORAL™ flux trap rack design, the spent fuel storage pool is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt % U-235, or spent fuel regardless of the discharge fuel burnup.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. Analysis demonstrates that the effective neutron multiplication factor (k_{eff}) of the spent fuel pool loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with unborated water, and does not exceed 0.95 for the pool flooded with borated water with 500 ppm soluble boron (an additional 50 ppm of soluble boron has been added to account for grid growth.) The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. To mitigate postulated criticality related events, boron is dissolved in the pool water.

APPLICABLE SAFETY ANALYSES

The following accident conditions have been evaluated:

- The effect of SFP temperature exceeding the normal range
- A dropped fuel assembly
- A misloaded fuel assembly (a fuel assembly in the wrong location within the storage rack)
- A mislocated fuel assembly (a fuel assembly in the wrong location outside the storage rack)
- Rack movement due to seismic activity

The results of the evaluation show that a soluble boron concentration of 500 ppm is sufficient to ensure that the maximum k_{eff} is below the regulatory limit of 0.95. The boron dilution analysis assumes an initial boron concentration of 2,300 ppm for the limiting evaluation. The accident analyses are provided in the FSAR, Section 4.3.2.7 (Ref. 2)

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO	The fuel storage pool boron concentration is required to be ≥2300 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.
ACTIONS	A.1

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. However, prior to resuming movement of the fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.7.18.1 This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 72 hour Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.
REFERENCES	1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A) 2. FSAR, Section 4.3.2.7

BASES (continued)

B 3.9 REFUELING OPERATIONS

B 3.9.9 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND	The spent fuel storage rack criticality analysis assumes ~~2000~~ 2300 ppm soluble boron in the fuel pool ~~during a dropped/misplaced fuel assembly event (Ref. 1)~~when fuel is being stored.
APPLICABLE SAFETY ANALYSES	This requirement ensures the presence of at least ~~2000~~ 2300 ppm soluble boron in the spent fuel pool water as assumed in the spent fuel rack criticality analysis for normal storage and a dropped~~/misplaced~~ fuel assembly event. The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that the boron concentration in the spent fuel pool be greater than or equal to ~~2000~~ 2300 ppm ~~during fuel movement~~anytime fuel is being stored in the pool.
APPLICABILITY	This LCO is applicable when the spent fuel pool is flooded and fuel is ~~being moved~~in the pool. ~~Once fuel movement begins, the movement is considered in progress until the configuration of the assemblies in the storage racks~~The assembly is verified to comply with the criticality loading criteria specified in Specification 4.3.1.1 before placing it in the Spent Fuel Pool.

BASES (continued)

ACTIONS

A.1

If the spent fuel pool boron concentration does not meet the above requirements, ~~fuel handling in the spent fuel pool must be suspended immediately. This action precludes a fuel handling accident, when conditions are outside those assumed in the accident analysis.~~action must be initiated to restore fuel storage pool boron concentration to within limits.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.9.1

This SR requires that the spent fuel pool boron concentration be verified greater than or equal to ~~2000~~ 2300 ppm. This surveillance is to be performed ~~prior to movement of fuel~~when fuel is stored in the spent fuel pool and at least once every 72 hours thereafter ~~during the movement of fuel in the spent fuel pool~~.

The Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of the sample. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. Watts Bar FSAR, Section ~~9.1.2, "Spent Fuel Storage" and Section 15, "Accident Analysis."~~4.3.2.7

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

WCAP-18191-NP, Revision 0, “Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations”

Subject: 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)

Westinghouse Non-Proprietary Class 3

WCAP-18191-NP
Revision 0

May 2017

Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations

WCAP-18191-NP
Revision 0

Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations

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RECORD OF REVISION

Revision 0: Original Issue

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure-temperature (P-T) limit curves for normal operation of the Watts Bar Unit 2 reactor vessel. The analyses herein consider implementation of Tritium-Producing Burnable Absorber Rods (TPBARs) at the beginning of Cycle 4. The heatup and cooldown P-T limit curves were generated using the limiting Adjusted Reference Temperature (ART) values for Watts Bar Unit 2. The limiting ART values were those of Intermediate Shell Forging 05 (Position 1.1) at both 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations.

The P-T limit curves were generated for 32 effective full-power years (EFPY) using the K_{Ic} methodology detailed in the 1998 through the 2000 Addenda Edition of the ASME Code, Section XI, Appendix G. The P-T limit curve generation methodology is consistent with the NRC-approved methodology documented in WCAP-14040-A, Revision 4. Heatup rates of 60 and 100°F/hr, and cooldown rates of 0 (steady-state), 20, 40, 60, and 100°F/hr were used to generate the P-T limit curves, with the flange requirements and without margins for instrumentation errors. The Watts Bar Unit 2 End of License (EOL) corresponding to 40 years of operation is 32 EFPY. The EOL P-T limit curves can be found in Figures 8-1 and 8-2.

Appendix A contains the thermal stress intensity factors for the maximum heatup and cooldown rates at 32 EFPY.

Appendix B contains a P-T limit evaluation of the reactor vessel inlet and outlet nozzles based on a 1/4T flaw postulated at the inside surface of the reactor vessel nozzle corner, where T is the thickness of the nozzle corner region. As discussed in Appendix B, the P-T limit curves generated based on the limiting cylindrical beltline material (Intermediate Shell Forging 05) bound the P-T limit curves for the reactor vessel inlet and outlet nozzles for Watts Bar Unit 2 at 32 EFPY.

Appendix C contains discussion of the other ferritic Reactor Coolant Pressure Boundary (RCPB) components relative to P-T limits. As discussed in Appendix C, all of the other ferritic RCPB components meet the applicable requirements of Section III of the ASME Code.

Appendix D contains an upper-shelf energy (USE) evaluation for all Watts Bar Unit 2 reactor vessel beltline and extended beltline materials. Per Appendix D, all beltline and extended beltline materials are projected to maintain USE values above the 50 ft-lb screening criterion per 10 CFR 50 Appendix G at 32 EFPY.

Appendix E contains a pressurized thermal shock (PTS) evaluation for all Watts Bar Unit 2 reactor vessel beltline and extended beltline materials. Per Appendix E, all beltline and extended beltline materials have projected RT_{PTS} values below the screening criteria set forth in 10 CFR 50.61. Additionally, Watts Bar Unit 2 will remain in Category I of the Emergency Response Guidelines through 32 EFPY.

Appendix F contains an updated surveillance capsule withdrawal schedule. Per Appendix F, three surveillance capsules are recommended to be withdrawn from the Watts Bar Unit 2 reactor before end of license.

1 INTRODUCTION

The purpose of this report is to present the calculations and the development of the Watts Bar Unit 2 heatup and cooldown P-T limit curves for 32 EFPY. This report documents the calculated Adjusted Reference Temperature (ART) values and the development of the P-T limit curves for normal operation. The analyses herein consider implementation of Tritium-Producing Burnable Absorber Rods (TPBARs) at the beginning of Cycle 4.

Heatup and cooldown P-T limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} ($RT_{NDT(U)}$) is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The U.S. Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2 [Ref. 1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of ART values ($RT_{NDT(U)}$ + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The calculated ART values for 32 EFPY are documented in Section 7 of this report. The fluence projections used in calculation of the ART values are provided in Section 2 of this report

The heatup and cooldown P-T limit curves documented in this report were generated using the most limiting ART values (plus an additional margin to account for future perturbations such as an uprate or surveillance capsule results) and the NRC-approved methodology documented in WCAP-14040-A, Revision 4 [Ref. 2], which allows use of ASME Code Cases N-641 and N-640. Specifically, the K_{Ic} methodology of the 1998 through the 2000 Addenda Edition of ASME Code, Section XI, Appendix G [Ref. 3] was used. The K_{Ic} curve is a lower bound static fracture toughness curve obtained from test data gathered from several different heats of pressure vessel steel. The limiting material is indexed to the K_{Ic} curve so that allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits are then determined using the allowable stress intensity factors.

The P-T limit curves herein were generated without instrumentation errors. The reactor vessel flange requirements of 10 CFR 50, Appendix G [Ref. 4] have been incorporated in the P-T limit curves. The P-T limit curves generated in Section 8 bound the P-T limit curves for the reactor vessel inlet and outlet nozzles generated in Appendix B for Watts Bar Unit 2 at 32 EFPY. Discussion of the other ferritic RCPB components relative to P-T limits is contained in Appendix C.

Appendices D, E, and F provide supplemental reactor vessel integrity analyses including a USE evaluation (Appendix D), a PTS and Emergency Response Guideline (ERG) analysis (Appendix E), and a revised surveillance capsule withdrawal schedule (Appendix F).

2 CALCULATED NEUTRON FLUENCE

2.1 INTRODUCTION

A discrete ordinates (S_n) transport analysis was performed for the Watts Bar Unit 2 reactor to determine the neutron radiation environment within the beltline and extended beltline regions of the reactor pressure vessel. In this analysis, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis.

All of the calculations described in this section were based on nuclear cross-section data derived from the Evaluated Nuclear Data File (ENDF) database (specifically, ENDF/B-VI). Furthermore, these neutron transport methodologies adhere to the guidelines and meet the requirements of the U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.190 [Ref. 5]. Additionally, the methods used to determine the pressure vessel neutron exposure are consistent with the USNRC approved methodology described in WCAP-14040-A [Ref. 2].

2.2 DISCRETE ORDINATES ANALYSIS

In performing the fast neutron exposure evaluations for the Watts Bar Unit 2 reactor vessel, plant-specific forward transport calculations were carried out using the following three-dimensional flux synthesis technique:

$$\phi(r,\theta,z) = \phi(r,\theta) \times \frac{\phi(r,z)}{\phi(r)}$$

where $\phi(r,\theta,z)$ is the synthesized three-dimensional neutron flux distribution, $\phi(r,\theta)$ is the transport solution in [r,θ] geometry, $\phi(r,z)$ is the two-dimensional solution for a cylindrical reactor model using the actual axial core power distribution, and $\phi(r)$ is the one-dimensional solution for a cylindrical reactor model using the same source per unit height as that used in the [r,θ] two-dimensional calculation.

For the transport calculations, the [r,θ] models depicted in Figure 2-1 through Figure 2-3 were utilized since, with the exception of the neutron pads, the reactor is octant symmetric. These [r,θ] models included representations of the core, the reactor internals, the neutron pads, the pressure vessel cladding and vessel wall, the insulation external to the pressure vessel, and the primary biological shield wall. The difference among the models is the azimuthal extent of the neutron pad and the configuration of surveillance capsules on those pads. The reactor model including a neutron pad segment without surveillance capsules and associated structures is shown in Figure 2-1. The geometric model of the neutron pad segment with a single surveillance capsule holder attached is shown in Figure 2-2, while a neutron pad segment with a dual surveillance capsule holder attached is shown in Figure 2-3. In developing these analytical models, nominal design dimensions were employed for the various structural components. Likewise, 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. The geometric mesh description of the [r,θ] reactor model consisted of 170 radial by 98 azimuthal intervals. 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 [r,θ] calculations was set at a value of 0.001.

The [r,z] model used for the Watts Bar Unit 2 calculations is shown in Figure 2-4 and extends radially from the centerline of the reactor core out to a location interior to the primary biological shield. Axially, the model extends from an elevation of about 1.6 feet below the bottom head ring to bottom peel circumferential weld (lowest analyzed point) to the centerline of the outlet and inlet nozzles. Therefore, the core, core baffle, former plates, core barrel, neutron pad, downcomer water, pressure vessel, cladding, inlet plenum, lower core support plates, lower support columns, lower core plates, nozzle legs, plates, gaps, upper core plates, and outlet plenums are explicitly included in the model. The analyzed locations as measured from the midplane of the active fuel can be seen in Table 2-1. As in the case of the [r,θ] models, nominal design dimensions and full power coolant densities were employed in the calculations. In this case, the homogenous core region was treated as an equivalent cylinder with a volume equal to that of the active core zone. The [r,z] geometric mesh description of these reactor models consisted of 153 radial by 158 axial intervals. As in the case of the [r,θ] calculations, 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 [r,z] calculations was also set at a value of 0.001.

The one-dimensional radial model used in the synthesis technique consisted of the same 153 radial mesh intervals included in the [r,z] model. Thus, radial synthesis factors could be determined on a meshwise basis throughout the entire geometry.

The core power distributions used in the plant-specific transport analysis for each of the first 7 projected fuel cycles at Watts Bar Unit 2 included cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions. Note that Watts Bar Unit 2 is currently operating in Cycle 1; therefore, no fuel cycles have been completed. Cycles 1-6 are based on the expected core design for these cycles and Cycle 7 is a representative equilibrium fuel cycle that is then extended to 32 and 40 Effective Full Power Years (EFPY). These calculations take into account implementation of TPBARs at the beginning of Cycle 4. The TPBAR core design was reflected in the radial power distributions that are used as critical inputs into the reactor vessel fluence calculation. In general, the TPBAR core designs show a higher relative power in the peripheral assemblies than that of a low leakage core design, which causes an increase in the reactor pressure vessel fast neutron fluence exposure in those TPBAR cycles. The increases can be seen in Table 2-4. The reactor vessel integrity evaluations herein consider the TPBAR core design and justify safe operation through 32 Effective Full Power Years with respect to reactor vessel integrity. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of fuel cycle-averaged neutron fluence 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.

All of the transport calculations were carried out using the DORT discrete ordinates code [Ref. 7] coupled with the BUGLE-96 cross-section library [Ref. 8]. The BUGLE-96 library provides a 67-group coupled

neutron-gamma ray cross-section data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering was treated with a P_5 Legendre expansion and the 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.

The data tabulations include both plant- and fuel-cycle-specific calculated neutron exposures at the end of Cycles (EOC) 1 through 7 and at further projections to 32 and 40 EFPY. The calculations account for the expected core power of 3411 MWt in Cycle 1 and 3459 MWt in Cycles 2-7. The projections after Cycle 7 are based on the assumption that the core power distributions and associated plant operating characteristics from Cycle 7 are representative of future plant operations.

Selected results from the neutron transport analyses are provided in Table 2-2 through Table 2-10. The calculated fast neutron exposure rate and total exposure values at the geometric center of the surveillance capsule test specimen are provided in Table 2-2 and Table 2-3, respectively. Likewise, Table 2-4 through 2-6 show the maximum calculated exposure and exposure rate at the clad/base metal interface. Azimuthally, angles 0.00, 20.00 through 22.00 at 0.25 degrees intervals, 23.00, 30.00, and 45.00 were analyzed. For Cycles 1-6, Table 2-5 and Table 2-6 show neutron exposure at the azimuthal angle of 45 degrees and a radial distance of 220.11 cm from the pressure vessel centerline, which is at the clad/base metal interface, because this resulted in the maximum neutron fluence and dpa. For Cycle 7 and on, Table 2-5 and Table 2-6 show neutron exposure at the azimuthal angle of 22 degrees and a radial distance of 220.11 cm from the pressure vessel centerline, which is at the clad/base metal interface, because this resulted in the maximum neutron fluence and dpa. Table 2-7 through Table 2-10 show the maximum neutron exposure at the pressure vessel clad/base metal interface at various axial points of interest. The data tabulation includes the projected plant-specific calculated fluence at the end of Cycles 1 through 7 and projections for an operating time extending to 32 and 40 EFPY.

2.3 CALCULATIONAL UNCERTAINTIES

An uncertainty analysis that includes comparisons of calculations with test and power reactor benchmarks and an analytical uncertainty study has been completed and documented in an NRC-approved topical report [Ref. 2 and 6]. The overall uncertainty in the transport calculations was demonstrated to be 13% (1σ). This level of uncertainty meets the 20% (1σ) requirement of Regulatory Guide 1.190.

Table 2-1 Pressure Vessel Material Locations

Location	Distance from Midplane of Active Fuel (cm)
Flange Mating Surface	556.823
Centerline of Inlet and Outlet Nozzles (Upper extent of r,z model)	343.463
Lowest Extent of Outlet Nozzle to Shell Weld	276.263
Lowest Extent of Inlet Nozzle to Shell Weld	265.263
Center of Intermediate Shell to Nozzle Shell Circumferential Weld	226.423
Bottom of Upper Core Plate	216.463
Top of Active Fuel	182.880
Center of Lower Shell A to Intermediate Shell Circumferential Weld	12.023
Midplane of Active Fuel	0.000
Bottom of Active Fuel	-182.880
Center of Lower Shell A to Lower Shell B Circumferential Weld	-201.677
Top of Lower Core Plate	-191.206
Center of Lower Ring to Lower Shell B Circumferential Weld	-314.777
Lowest extent of r,z model	-363.296
Lower Head to Lower Ring Circumferential Weld	-401.977

Table 2-2 Calculated Fast Neutron Exposure Rate at the Geometric Center of the Surveillance Capsules at Core Midplane

Cycle	Cumulative Operating Time (EFPY)	Neutron Flux (E > 1.0 MeV) [n/cm²-s]			Iron Atom Displacement Rate [dpa/s]		
		31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single	31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single
1	**1.27**	9.077E+10	1.075E+11	1.091E+11	1.787E-10	2.150E-10	2.226E-10
2	**2.61**	6.987E+10	8.059E+10	8.174E+10	1.360E-10	1.593E-10	1.648E-10
3	**4.14**	5.622E+10	6.520E+10	6.614E+10	1.093E-10	1.289E-10	1.333E-10
4	**5.53**	7.263E+10	8.412E+10	8.534E+10	1.417E-10	1.667E-10	1.725E-10
5	**6.91**	8.054E+10	9.590E+10	9.734E+10	1.578E-10	1.909E-10	1.976E-10
6	**8.30**	8.107E+10	9.563E+10	9.705E+10	1.588E-10	1.903E-10	1.970E-10
7	**9.69**	7.870E+10	9.059E+10	9.191E+10	1.541E-10	1.802E-10	1.864E-10
Future	**32.00**	7.870E+10	9.059E+10	9.191E+10	1.541E-10	1.802E-10	1.864E-10
Future	**40.00**	7.870E+10	9.059E+10	9.191E+10	1.541E-10	1.802E-10	1.864E-10
Surveillance Capsule Center = 207.32 cm							

Table 2-3 Calculated Fast Neutron Exposure at the Geometric Center of the Surveillance Capsules at Core Midplane

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm²]			Iron Atom Displacement [dpa]		
		31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single	31.5 Degrees Dual	34.0 Degrees Dual	34.0 Degrees Single
1	**1.27**	3.636E+18	4.307E+18	4.370E+18	7.158E-03	8.615E-03	8.917E-03
2	**2.61**	6.590E+18	7.714E+18	7.826E+18	1.291E-02	1.535E-02	1.589E-02
3	**4.14**	9.304E+18	1.086E+19	1.102E+19	1.819E-02	2.157E-02	2.232E-02
4	**5.53**	1.248E+19	1.454E+19	1.475E+19	2.438E-02	2.887E-02	2.987E-02
5	**6.91**	1.600E+19	1.873E+19	1.901E+19	3.128E-02	3.721E-02	3.851E-02
6	**8.30**	1.955E+19	2.291E+19	2.325E+19	3.822E-02	4.553E-02	4.712E-02
7	**9.69**	2.301E+19	2.690E+19	2.729E+19	4.500E-02	5.345E-02	5.532E-02
Future	**32.00**	7.842E+19	9.069E+19	9.200E+19	1.535E-01	1.803E-01	1.866E-01
Future	**40.00**	9.829E+19	1.136E+20	1.152E+20	1.924E-01	2.258E-01	2.336E-01
Surveillance Capsule Center = 207.32 cm							

Table 2-4 Calculated Maximum Fast Neutron Exposure Rate at the Pressure Vessel Clad/Base Metal Interface

Cycle	Cumulative Operating Time (EFPY)	Neutron Flux (E > 1.0 MeV) [n/cm^2-s]	Iron Atom Displacement Rate [dpa/s]
1	1.27	2.193E+10	3.466E-11
2	2.61	1.614E+10	2.547E-11
3	4.14	1.351E+10	2.132E-11
4	5.53	1.687E+10	2.663E-11
5	6.91	2.004E+10	3.162E-11
6	8.30	1.962E+10	3.098E-11
7	9.69	1.946E+10	2.974E-11
Future	32.00	1.946E+10	2.974E-11
Future	40.00	1.946E+10	2.974E-11

Table 2-5 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface (Neutron Fluence)

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]					
		0.0 Degrees	22.0 Degrees	23.0 Degrees	30.0 Degrees	45.0 Degrees	Max.
1	1.27	4.513E+17	8.262E+17	8.250E+17	7.019E+17	8.787E+17	8.787E+17
2	2.61	8.036E+17	1.467E+18	1.470E+18	1.277E+18	1.560E+18	1.560E+18
3	4.14	1.191E+18	2.074E+18	2.077E+18	1.811E+18	2.212E+18	2.212E+18
4	5.53	1.583E+18	2.797E+18	2.800E+18	2.433E+18	2.950E+18	2.950E+18
5	6.91	1.968E+18	3.560E+18	3.565E+18	3.108E+18	3.826E+18	3.826E+18
6	8.30	2.411E+18	4.354E+18	4.359E+18	3.793E+18	4.684E+18	4.684E+18
7	9.69	2.894E+18	5.210E+18	5.210E+18	4.484E+18	5.478E+18	5.478E+18
Future	32.00	1.064E+19	1.891E+19	1.882E+19	1.555E+19	1.821E+19	1.891E+19
Future	40.00	1.341E+19	2.382E+19	2.370E+19	1.952E+19	2.277E+19	2.382E+19
Base Metal Inner Radius = 220.11 cm							

Table 2-6 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Clad/Base Metal Interface (Iron Displacement)

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacement [dpa]					
		0.0 Degrees	22.0 Degrees	23.0 Degrees	30.0 Degrees	45.0 Degrees	Max.
1	**1.27**	6.990E-04	1.262E-03	1.261E-03	1.092E-03	1.388E-03	1.388E-03
2	**2.61**	1.246E-03	2.244E-03	2.249E-03	1.987E-03	2.464E-03	2.464E-03
3	**4.14**	1.846E-03	3.174E-03	3.180E-03	2.818E-03	3.493E-03	3.493E-03
4	**5.53**	2.455E-03	4.281E-03	4.287E-03	3.785E-03	4.659E-03	4.659E-03
5	**6.91**	3.052E-03	5.448E-03	5.458E-03	4.835E-03	6.041E-03	6.041E-03
6	**8.30**	3.739E-03	6.665E-03	6.674E-03	5.901E-03	7.395E-03	7.395E-03
7	**9.69**	4.488E-03	7.973E-03	7.974E-03	6.976E-03	8.651E-03	8.651E-03
Future	**32.00**	1.649E-02	2.891E-02	2.879E-02	2.418E-02	2.876E-02	2.891E-02
Future	**40.00**	2.080E-02	3.642E-02	3.625E-02	3.036E-02	3.597E-02	3.642E-02
Base Metal Inner Radius = 220.11 cm							

Table 2-7 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Circumferential Welds (Neutron Fluence)

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			
		Bottom Head Ring to Bottom Peel Circ. Weld	Lower Shell to Bottom Head Ring Circ. Weld	Intermediate to Lower Shell Circ. Weld	Upper to Intermediate Shell Circ. Weld
1	**1.27**	4.197E+13	1.195E+17	8.708E+17	5.241E+16
2	**2.61**	8.044E+13	2.168E+17	1.545E+18	9.055E+16
3	**4.14**	1.211E+14	3.145E+17	2.191E+18	1.328E+17
4	**5.53**	1.637E+14	4.250E+17	2.921E+18	1.766E+17
5	**6.91**	2.107E+14	5.507E+17	3.785E+18	2.253E+17
6	**8.30**	2.559E+14	6.726E+17	4.632E+18	2.719E+17
7	**9.69**	2.919E+14	7.724E+17	5.413E+18	2.952E+17
Future	**32.00**	8.985E+14	2.454E+18	1.861E+19	1.097E+18
Future	**40.00**	1.121E+15	3.071E+18	2.344E+19	1.385E+18
Bottom Head Ring to Bottom Peel Circ. Weld Elevation in r,z model: z = -314.777 cm, r = 220.11 cm Lower Shell to Bottom Head Ring Circ. Weld Elevation in r,z model: z = -201.677 cm, r = 220.11 cm Intermediate to Lower Shell Circ. Weld Elevation in r,z model: z = 12.023 cm, r = 220.11 cm Upper to Intermediate Shell Circ. Weld Elevation in r,z model: z = 226.423 cm, r = 217.56 cm					

Table 2-8 Calculated Maximum Fast Neutron Exposure at the Pressure Vessel Circumferential Welds (Iron Displacement)

Cycle	Cumulative Operating Time (EFPY)	Iron Atom Displacement [dpa]			
		Bottom Head Ring to Bottom Peel Circ. Weld	Lower Shell to Bottom Head Ring Circ. Weld	Intermediate to Lower Shell Circ. Weld	Upper to Intermediate Shell Circ. Weld
1	1.27	1.892E-07	1.900E-04	1.376E-03	8.335E-05
2	2.61	3.496E-07	3.448E-04	2.440E-03	1.440E-04
3	4.14	5.121E-07	4.999E-04	3.461E-03	2.110E-04
4	5.53	6.899E-07	6.754E-04	4.612E-03	2.805E-04
5	6.91	8.916E-07	8.752E-04	5.976E-03	3.580E-04
6	8.30	1.087E-06	1.069E-03	7.313E-03	4.321E-04
7	9.69	1.250E-06	1.228E-03	8.547E-03	5.082E-04
Future	32.00	3.874E-06	3.782E-03	2.846E-02	1.728E-03
Future	40.00	4.851E-06	4.734E-03	3.584E-02	2.165E-03

Bottom Head Ring to Bottom Peel Circ. Weld Elevation in r,z model: z = -314.777 cm, r = 220.11 cm
Lower Shell to Bottom Head Ring Circ. Weld Elevation in r,z model: z = -201.677 cm, r = 220.11 cm
Intermediate to Lower Shell Circ. Weld Elevation in r,z model: z = 12.023 cm, r = 220.11 cm
Upper to Intermediate Shell Circ. Weld Elevation in r,z model: z = 226.423 cm, r = 217.56 cm

Table 2-9 Calculated Maximum Fast Neutron Exposure at Pressure Vessel Nozzle Welds

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]		Iron Atom Displacement [dpa]	
		Inlet Nozzle Weld	Outlet Nozzle Weld	Inlet Nozzle Weld	Outlet Nozzle Weld
1	1.27	3.764E+15	1.947E+15	6.63E-06	3.571E-06
2	2.61	6.668E+15	3.459E+15	1.17E-05	6.322E-06
3	4.14	9.924E+15	5.176E+15	1.74E-05	9.409E-06
4	5.53	1.334E+16	6.959E+15	2.34E-05	1.265E-05
5	6.91	1.667E+16	8.690E+15	2.92E-05	1.581E-05
6	8.30	2.004E+16	1.044E+16	3.52E-05	1.902E-05
7	9.69	2.407E+16	1.255E+16	4.22E-05	2.286E-05
Future	32.00	8.861E+16	4.630E+16	1.55E-04	8.423E-05
Future	40.00	1.118E+17	5.841E+16	1.96E-04	1.062E-04

Outlet Nozzle Weld Elevation in r,z model: z = 276.263 cm, θ = 22°, r = 217.56 cm
Inlet Nozzle Weld Elevation in r,z model: z = 265.263 cm, θ = 23°, r = 217.56 cm

Table 2-10 Calculated Maximum Fast Neutron Exposure at Pressure Vessel Shells

Cycle	Cumulative Operating Time (EFPY)	Neutron Fluence (E > 1.0 MeV) [n/cm^2]			Iron Atom Displacement [dpa]		
		Intermediate Shell	Lower Shell	Bottom Head Ring	Intermediate Shell	Lower Shell	Bottom Head Ring
1	**1.27**	8.708E+17	8.787E+17	1.195E+17	1.376E-03	1.388E-03	1.900E-04
2	**2.61**	1.545E+18	1.560E+18	2.168E+17	2.440E-03	2.464E-03	3.448E-04
3	**4.14**	2.191E+18	2.212E+18	3.145E+17	3.461E-03	3.493E-03	4.999E-04
4	**5.53**	2.921E+18	2.950E+18	4.250E+17	4.612E-03	4.659E-03	6.754E-04
5	**6.91**	3.785E+18	3.826E+18	5.507E+17	5.976E-03	6.041E-03	8.752E-04
6	**8.30**	4.632E+18	4.684E+18	6.726E+17	7.313E-03	7.395E-03	1.069E-03
7	**9.69**	5.413E+18	5.478E+18	7.724E+17	8.547E-03	8.651E-03	1.228E-03
Future	**32.00**	1.861E+19	1.891E+19	2.454E+18	2.846E-02	2.891E-02	3.782E-03
Future	**40.00**	2.344E+19	2.382E+19	3.071E+18	3.584E-02	3.642E-02	4.734E-03
Bottom Head Ring Spans: -314.777 cm to -201.677 cm Lower Shell Spans: -201.677 cm to 12.023 cm Intermediate Shell Spans: 12.023 cm to 226.423 cm							

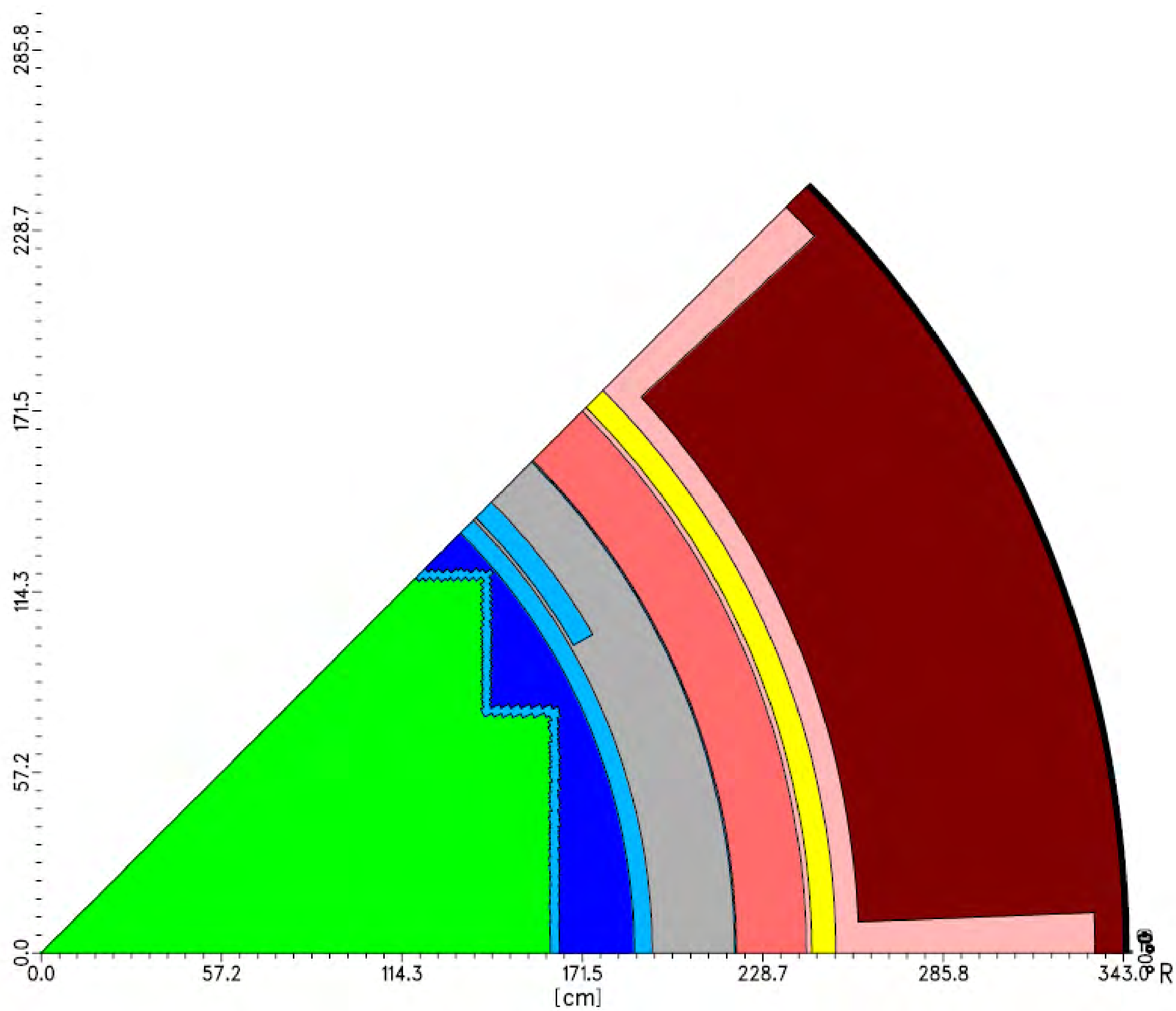

Figure 2-1 Watts Bar Unit 2 Reactor Geometry; [r,θ] Plan View, 15° Neutron Pad Configuration without Surveillance Capsules

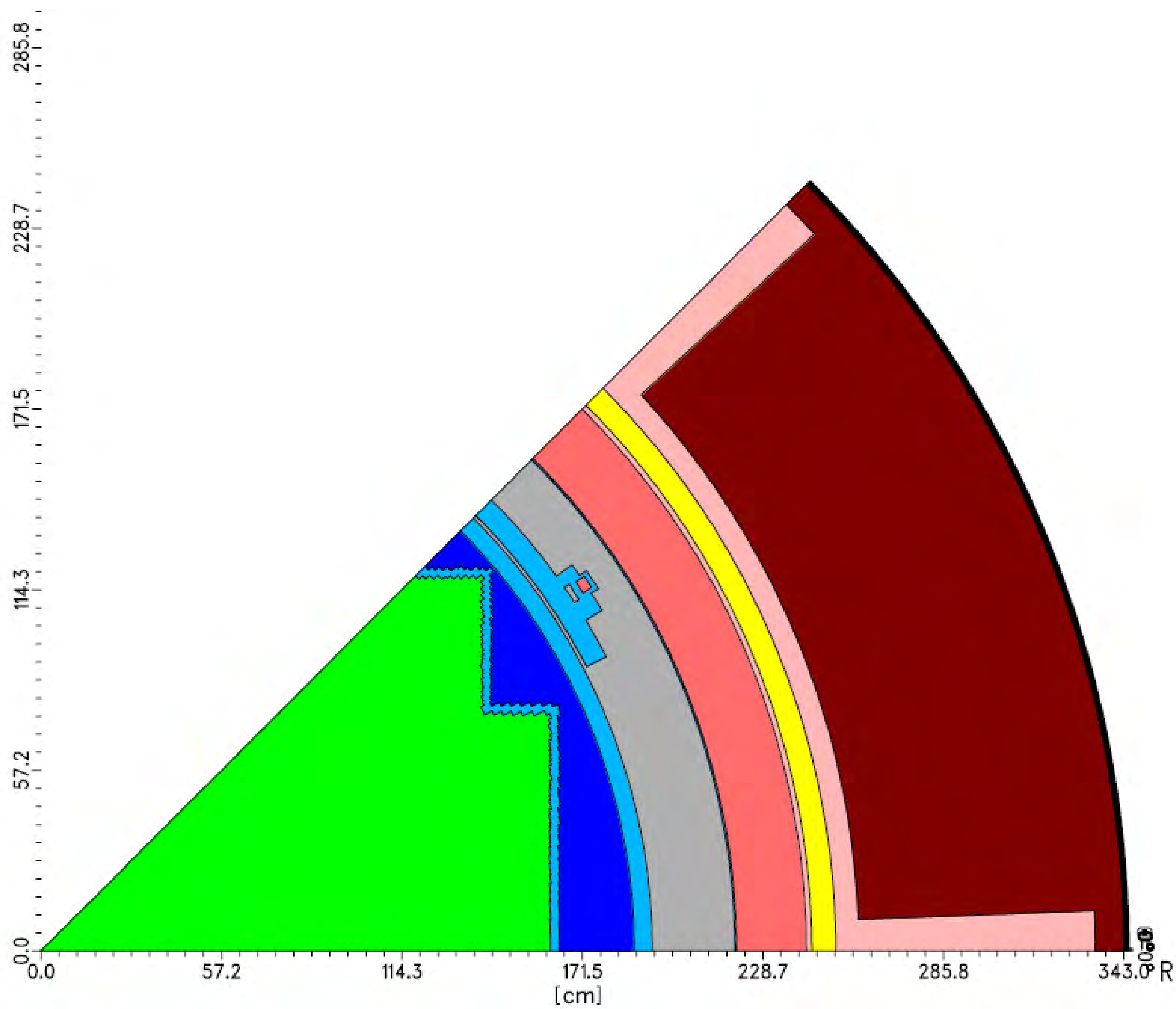

Figure 2-2 Watts Bar Unit 2 Reactor Geometry; [r,θ] Plan View, 17.5° Neutron Pad Configuration with a Single Capsule Holder

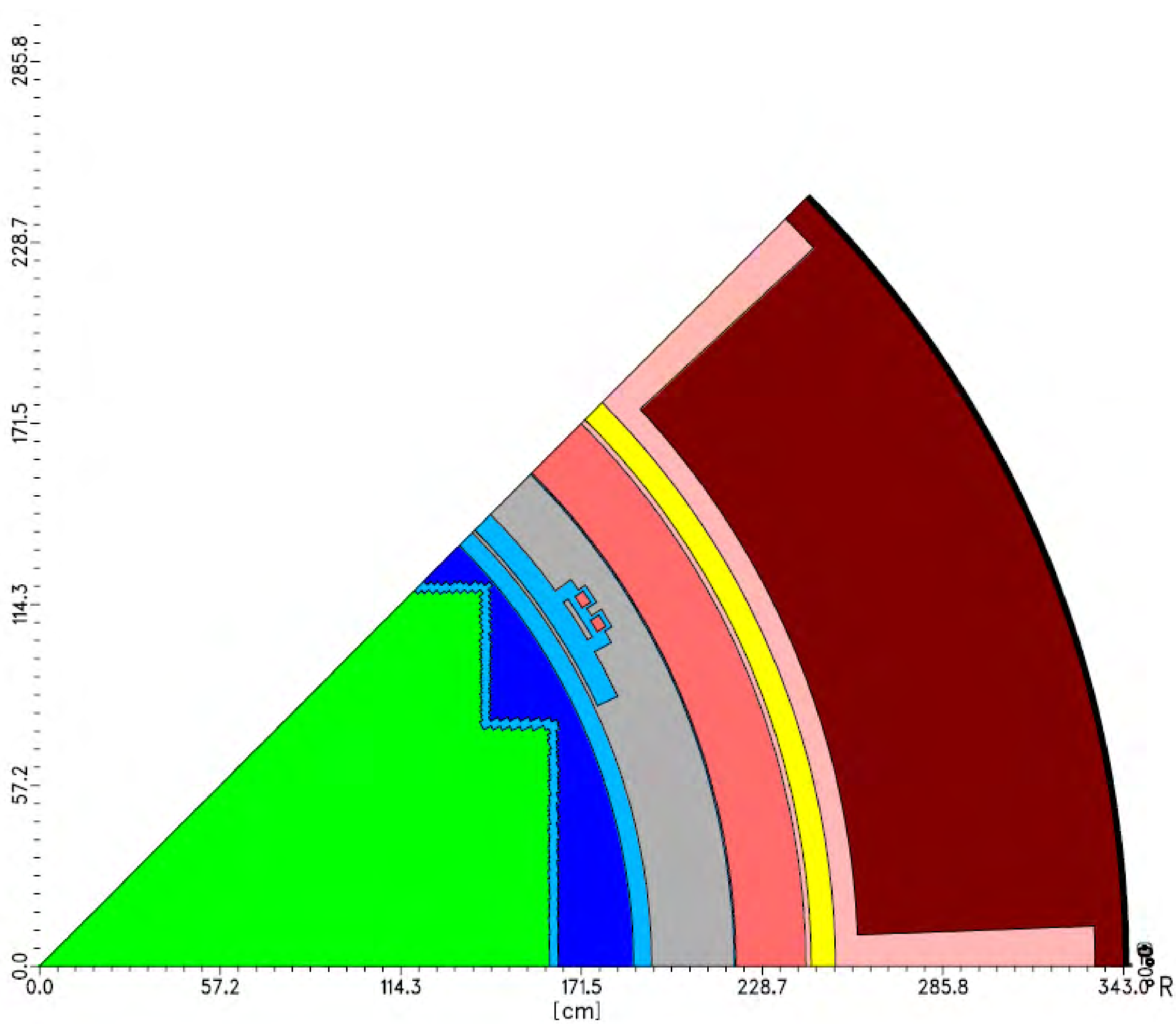

Figure 2-3 Watts Bar Unit 2 Reactor Geometry; [r,θ] Plan View 20° Neutron Pad Configuration with a Dual Capsule Holder

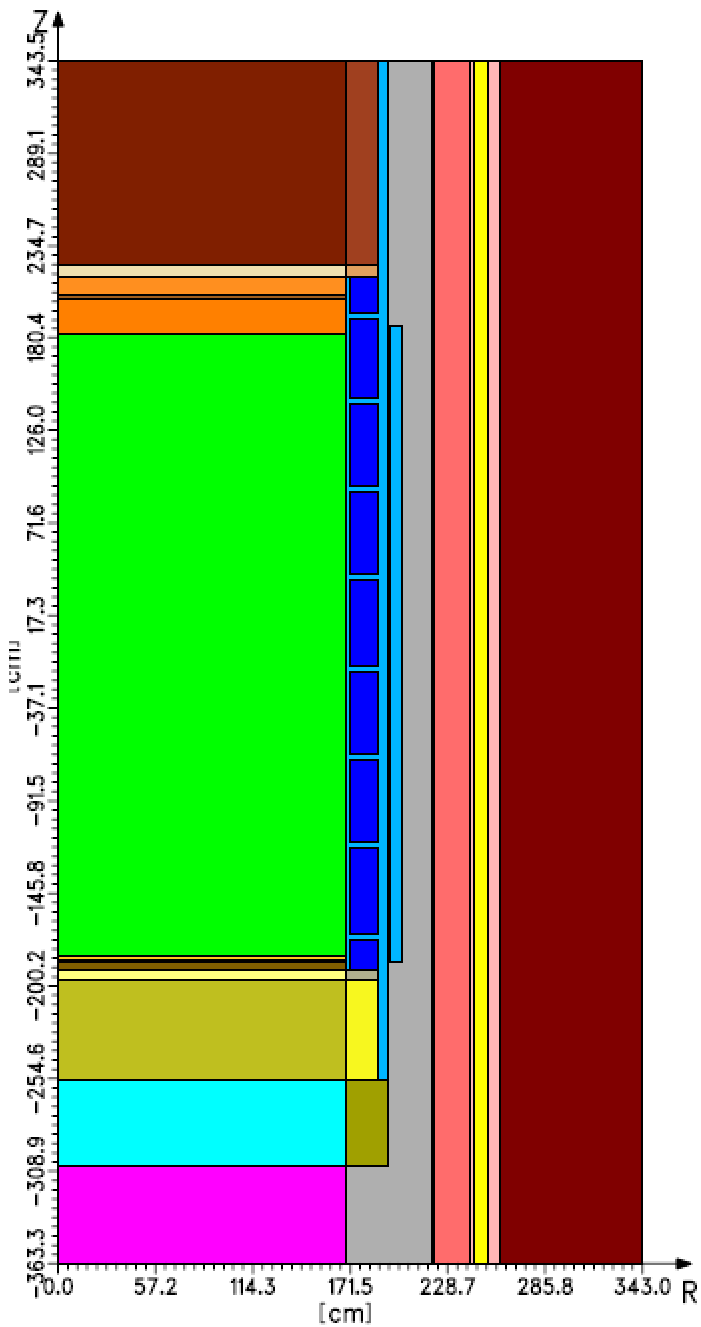

Figure 2-4 Watts Bar Unit 2 Reactor Geometry; [r,z] Section View without Surveillance Capsules

3 FRACTURE TOUGHNESS PROPERTIES

The requirements for P-T limit curve development are specified in 10 CFR 50, Appendix G [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 Unit 2 beltline materials traditionally included Intermediate Shell Forging 05, Lower Shell Forging 04, and Intermediate to Lower Shell Circumferential Weld W05; however, as described in NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. 9] and TLR-RES/DE/CIB-2013-01 [Ref. 10], any reactor vessel materials that are predicted to experience a neutron fluence exposure greater than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of P-T limit curves. The additional materials that exceed this fluence threshold are referred to as extended beltline materials and are evaluated to ensure that the applicable acceptance criteria are met. As seen from Tables 2-7, 2-9, and 2-10 of this report, the extended beltline materials include Upper Shell Forging 06, Bottom Head Ring 03, Upper to Intermediate Shell Circumferential Weld W06, and Lower Shell to Bottom Head Ring Circumferential Weld W04. Note that the Upper to Intermediate Shell Circumferential Weld W06 fluence value is conservatively applied to the Upper Shell Forging 06 herein; this approach is conservative. Although the reactor vessel nozzles are not a part of the extended beltline, per NRC RIS 2014-11, the nozzle materials must be evaluated for their potential effect on P-T limit curves regardless of exposure – See Appendix B for more details.

As part of this P-T limit curve development effort, the initial RT_{NDT} and initial USE values for the Watts Bar Unit 2 reactor vessel beltline and extended beltline base metal materials are required. Since this document represents the first definition of the extended beltline materials for Watts Bar Unit 2, the initial RT_{NDT} and USE values for these materials are determined herein. The initial RT_{NDT} values for each of the extended beltline forgings were determined per BTP 5-3, Paragraph B1.1(3) [Ref. 11] in conjunction with ASME Code, Section III, Subsection NB-2300 [Ref. 12]. These initial RT_{NDT} values were determined using both BTP 5-3 Position 1.1(3)(a) and Position 1.1(3)(b), and the more limiting initial RT_{NDT} value was chosen for each material. The initial USE values were determined in accordance with the methodology described in ASTM E185-82 [Ref. 19]. For each of the extended beltline forgings, use of BTP 5-3 Paragraph 1.2 [Ref. 11] was necessary. A summary of the best-estimate copper (Cu), nickel (Ni), Manganese (Mn), and Phosphorus (P) contents, in units of weight percent (wt. %), as well as initial RT_{NDT} and initial USE values for the reactor vessel beltline and extended beltline materials are provided in Table 3-1 for Watts Bar Unit 2. Table 3-2 contains a summary of the initial RT_{NDT} values of the reactor vessel flange and reactor vessel closure head flange. These values serve as input to the P-T limit curves "flange-notch" per Appendix G of 10 CFR 50 – See Section 6.3 for details.

Note that data from a surveillance program at another plant is often called 'sister-plant' data; this term will be utilized herein.

Table 3-1 Summary of the Best-Estimate Chemistry, Initial RT_{NDT}, and Initial USE Values for the Watts Bar Unit 2 Reactor Vessel Materials

Reactor Vessel Material and Identification Number	Heat Number	Chemical Composition[(a)]				Fracture Toughness Property	
		Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	Initial RT_{NDT} (°F)	Initial Upper-Shelf Energy (ft-lb)
Reactor Vessel Beltline Materials							
Intermediate Shell Forging 05	527828	0.05[(b)]	0.78[(b)]	0.72	0.012	14[(b)]	90[(b)]
Lower Shell Forging 04	528658	0.05[(b)]	0.81[(b)]	0.72	0.006	5[(b)]	105[(b)]
Intermediate to Lower Shell Circumferential Weld W05	895075	0.05[(b)]	0.70[(b)]	1.97	0.010	-50[(b)]	127[(b)]
Reactor Vessel Extended Beltline Materials							
Upper Shell Forging 06	411572	0.07	0.91	0.70	0.005	-14[(c)]	94[(c)]
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	0.03	0.75	1.97	0.009	10[(d)]	92[(e)]
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	899680	0.03	0.75	1.97	0.009	10[(d)]	92[(e)]
Bottom Head Ring 03	5329	0.06	0.86	0.72	0.009	-40[(c)]	105[(c)]
Surveillance Materials[(f)]							
Intermediate Shell Forging 05	527828	0.05	0.78	0.72	0.012	- - -	90
Watts Bar Unit 2 Surveillance Weld	895075[(g)]	0.033	0.70	1.89	0.013	- - -	144.3
Catawba Unit 1 Surveillance Weld		0.05	0.73	- - -	- - -	- - -	- - -
Watts Bar Unit 1 Surveillance Weld		0.03	0.75	- - -	- - -	- - -	- - -
McGuire Unit 2 Surveillance Weld		0.04	0.74	- - -	- - -	- - -	- - -

Notes (continued on following page):

(a) All chemistry values are obtained from the Watts Bar Unit 2 Certified Material Test Reports (CMTRs), unless otherwise noted. Wt. % Mn and wt. % P values are not necessary for reactor vessel integrity evaluations; these values are provided for future use, if needed.

(b) Chemistry and initial RT_{NDT} values are taken from Table 2-1 of WCAP-17035-NP, Revision 2 [Ref. 13]. Initial USE values are taken from Table B-1 of WCAP-17035-NP, Revision 2.

(c) The extended 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 [Ref. 11] Position 1.1(3)(a) and (b), with the more limiting $RT_{NDT(U)}$ value being selected. 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 Position 1.2 methodology.

(d) Value consistent with the initial RT_{NDT} value for weld Heat # 899680 as reported in WCAP-17455-NP [Ref. 14] for McGuire Unit 2 and WCAP-17669-NP, Revision 1 [Ref. 15] for Catawba Unit 1. See Note (e) below.

(e) 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 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 should be used in accordance with Section B.1.2 of NUREG-0800 Branch Technical Position 5-3 [Ref. 11]. However, since the first capsule has not yet been tested for Watts Bar Unit 2, the limiting USE value from sister-plant Heat # 899680 analyses is used herein. The sister-plant analyses are contained in WCAP-17455-NP [Ref. 14] for McGuire Unit 2 and WCAP-17669-NP, Revision 1 [Ref. 15] for Catawba Unit 1. This value should be revisited after test results from the first capsule are available.

(f) The Watts Bar Unit 2 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 2 surveillance weld material was made using Heat # 895075 with Grau L.O. (LW 320) flux, lot P46, which is identical to the Watts Bar Unit 2 Intermediate to Lower Shell Circumferential Weld per WCAP-9455, Revision 3 [Ref. 16]. The chemistry values for the Watts Bar Unit 2 surveillance weld are the average of the values in Table A-2 of WCAP-9455, Revision 3. The USE value for the Watts Bar Unit 2 surveillance weld is taken from Section 3 of WCAP-9455, Revision 3.

(g) Surveillance data exists for weld Heat # 895075 from multiple sources; see Section 4 for more details. The data for the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance welds was taken from Table 3-1 of WCAP-17669-NP, Revision 1 [Ref. 15].

Table 3-2 Summary of Watts Bar Unit 2 Reactor Vessel Closure Head Flange and Vessel Flange Initial RT_{NDT} Values

Reactor Vessel Material	Initial RT_{NDT} [(a)] (°F)
Closure Head Flange	-40
Vessel Flange	-22

Notes:

(a) Values taken from WCAP-17035-NP, Revision 2 [Ref. 13] and are consistent with those documented in WCAP-13830, Revision 1 [Ref. 17].

4 SURVEILLANCE DATA

Per Regulatory Guide 1.99, Revision 2 [Ref. 1], calculation of Position 2.1 chemistry factors requires data from the plant-specific surveillance program. In addition to the plant-specific surveillance data, surveillance data generated from surveillance programs at other plants which include a reactor vessel beltline or extended beltline material should also be considered when calculating Position 2.1 chemistry factors.

The surveillance capsule plate material for Watts Bar Unit 2 is from Intermediate Shell Forging 05. The surveillance capsule weld material for Watts Bar Unit 2 is Heat # 895075, which is applicable to the intermediate to lower shell circumferential weld.

While no surveillance capsules have yet to be removed and tested from the Watts Bar Unit 2 reactor, weld Heat # 895075 is contained in the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance programs. Thus, the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 data will be used in the calculation of the Position 2.1 chemistry factor value for Watts Bar Unit 2 weld Heat # 895075. Table 4-1 summarizes the applicable surveillance capsule data pertaining to weld Heat # 895075. Per Appendix A of WCAP-17669-NP, Revision 1 [Ref. 15], the combined Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance weld data (Heat # 895075) is deemed credible. This credibility conclusion is applicable to the Watts Bar Unit 2 weld Heat # 895075 evaluated herein, since no additional surveillance data was included in this analysis. Therefore, a reduced margin term will be utilized in the ART and PTS calculations contained in Section 7 and Appendix E, respectively.

Table 4-1 Catawba Unit 1, Watts Bar 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)	Inlet Temperature (°F)	Temperature Adjustment[b] (°F)
Catawba Unit 1 Data	Z	0.286	1.91	553	-3
	Y	1.29	17.79		
	V	2.27	26.5		
Watts Bar Unit 1 Data	U	0.447	0.0	560	4
	W	1.08	30.5		
	X	1.71	25.8		
	Z	2.40	13.9		
McGuire Unit 2 Data	V	0.302	38.51	557	1
	X	1.38	35.93		
	U	1.90	23.81		
	W	2.82	43.76		

Notes:

(a) All data taken from WCAP-17669-NP, Revision 1 [Ref. 15] Table 4-1 or 4-2, unless otherwise noted.

(b) Temperature adjustment = 1.0*($T_{capsule}$ - T_{plant}), where T_{plant} = 556°F for Watts Bar Unit 2 (applied to the weld ΔRT_{NDT} data for each of the Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 capsules in the Position 2.1 chemistry factor calculation – See Section 5 for more details).

5 CHEMISTRY FACTORS

The chemistry factors (CFs) were calculated using Regulatory Guide 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 Regulatory Guide 1.99, Revision 2. The best-estimate copper and nickel weight percent values for the Watts Bar Unit 2 reactor vessel materials are provided in Table 3-1 of this report.

The Position 2.1 chemistry factor calculation is presented in Table 5-1 for the Watts Bar Unit 2 Heat # 895075 material. This value was calculated using the surveillance data summarized in Section 4 of this report, which includes available surveillance program results from sister plants. The calculation is performed using the method described in Regulatory Guide 1.99, Revision 2. All of the surveillance data is adjusted for irradiation temperature and chemical composition differences in accordance with the guidance presented at an industry meeting held by the NRC on February 12 and 13, 1998 [Ref. 18]. Margin will be applied to the ART calculations in Section 7 and PTS calculations in Appendix E according to the conclusions of the credibility evaluation for the surveillance material, as documented in Section 4.

The Position 1.1 chemistry factors are summarized along with the Position 2.1 chemistry factors in Table 5-2 for Watts Bar Unit 2.

Table 5-1 Calculation of Watts Bar Unit 2 Chemistry Factor Value for Weld Heat # 895075 Using Surveillance Capsule Data

Weld Metal Heat # 895075	Capsule	Capsule f[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Catawba Unit 1 Data	Z	0.286	0.658	0.00[(d)] (1.91)	0.00	0.433
	Y	1.29	1.071	14.79 (17.79)	15.84	1.147
	V	2.27	1.222	23.50 (26.5)	28.71	1.493
Watts Bar Unit 1 Data	U	0.447	0.776	6.64 (0.0)	5.15	0.602
	W	1.08	1.022	57.27 (30.5)	58.50	1.044
	X	1.71	1.148	49.47 (25.8)	56.77	1.317
	Z	2.40	1.236	29.71 (13.9)	36.73	1.528
McGuire Unit 2 Data	V	0.302	0.672	49.78 (38.51)	33.45	0.452
	X	1.38	1.089	46.53 (35.93)	50.69	1.187
	U	1.90	1.176	31.26 (23.81)	36.75	1.382
	W	2.82	1.276	56.40 (43.76)	71.95	1.628
				SUM:	394.56	12.211
	$CF_{Weld\ Heat\ \#\ 895075} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (394.56) \div (12.211) =$ **32.3°F**					

Notes:

(a) f = fluence.

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. The ΔRT_{NDT} values are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences in the surveillance weld chemistry and the reactor vessel weld chemistry (pre-adjusted values are listed in parentheses and were taken from Table 4-1 of this report). The temperature adjustments are listed in Table 4-1. Ratio applied to the Catawba Unit 1 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 68°F / 68°F = 1.00. Ratio applied to the Watts Bar Unit 1 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 68°F / 41°F = 1.66. Ratio applied to the McGuire Unit 2 surveillance data = $CF_{Vessel\ Weld}$ / $CF_{Surv.\ Weld}$ = 68°F / 54°F = 1.26.

(d) A negative ΔRT_{NDT} value was calculated (-1.09°F). Physically, this should not occur; thus, a conservative value of 0°F was utilized for this calculation.

Table 5-2 Summary of Watts Bar Unit 2 Positions 1.1 and 2.1 Chemistry Factors

Reactor Vessel Material and Identification Number	Heat Number	Chemistry Factor (°F)	
		Position 1.1[(a)]	Position 2.1
Reactor Vessel Beltline Materials			
Intermediate Shell Forging 05	527828	31.0	- - -
Lower Shell Forging 04	528658	31.0	- - -
Intermediate to Lower Shell Circumferential Weld W05	895075	68.0	32.3[(b)]
Reactor Vessel Extended Beltline Materials			
Upper Shell Forging 06	411572	44.0	- - -
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	- - -
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04	899680	41.0	- - -
Bottom Head Ring 03	5329	37.0	- - -
Surveillance Weld Data			
Watts Bar Unit 2	895075	44.9	- - -
Catawba Unit 1		68.0	- - -
Watts Bar Unit 1		41.0	- - -
McGuire Unit 2		54.0	- - -

Notes:

(a) Position 1.1 chemistry factors were calculated using the copper and nickel weight percent values presented in Table 3-1 of this report and Tables 1 and 2 of Regulatory Guide 1.99, Revision 2 [Ref. 1].

(b) Position 2.1 chemistry factor was taken from Table 5-1 of this report. As discussed in Section 4, the surveillance weld data was deemed credible.

6 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

6.1 OVERALL APPROACH

The ASME (American Society of Mechanical Engineers) approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in the 1998 Edition through the 2000 Addenda of Section XI, Appendix G of the ASME Code [Ref. 3]. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} (ksi√in.) = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

6.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \quad (3)$$

where, M_m for an inside axial surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

and, M_m for an outside axial surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

Similarly, M_m for an inside or an outside circumferential surface flaw is given by:

M_m = 0.89 for $\sqrt{t} < 2$,
M_m = $0.443\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 1.53 for $\sqrt{t} > 3.464$

where,

p = internal pressure (ksi), R_i = vessel inner radius (in), and t = vessel wall thickness (in).

For bending stress, the corresponding K_I for the postulated axial or circumferential defect is:

K_{Ib} = M_b * Maximum Bending Stress, where M_b is two-thirds of M_m (4)

The maximum K_I produced by radial thermal gradient for the postulated axial or circumferential inside surface defect of G-2120 is:

$$K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5} \quad (5)$$

where CR is the cooldown rate in °F/hr., or for a postulated axial or circumferential outside surface defect

$$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5} \quad (6)$$

where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from ASME Code, Section XI, Appendix G, Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Code, Section XI, Appendix G, Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness axial or circumferential inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \quad (7)$$

or similarly, K_{It} during heatup for a ¼-thickness outside axial or circumferential surface defect using the relationship:

$$K_{It} = (1.043\,C_0 + 0.630\,C_1 + 0.481\,C_2 + 0.401\,C_3) * \sqrt{\pi a} \quad (8)$$

where the coefficients C_0, C_1, C_2, and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (9)$$

and x is a variable that represents the radial distance (in) from the appropriate (i.e., inside or outside) surface to any point on the crack front, and a is the maximum crack depth (in).

Note that Equations 3, 7, and 8 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. The P-T curve methodology is the same as that described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" [Ref. 2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1). Finally, the reactor vessel metal temperature at the crack tip of a postulated flaw is determined based on the methodology contained in Section 2.6.1 of WCAP-14040-A, Revision 4 (equation 2.6.1-1). This equation is solved utilizing values for thermal diffusivity of 0.518 ft^2/hr at 70°F and 0.379 ft^2/hr at 550°F and a constant convective heat-transfer coefficient value of 7000 Btu/hr-ft^2-°F.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw (the postulated flaw has a depth of 1/4 of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, Paragraph G-2120), the appropriate value for RT_{NDT}, and the reference fracture toughness curve (Equation 1). The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference 1/4T flaw of Appendix G to Section XI of the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the vessel wall because the thermal gradients, which increase with increasing cooldown rates, produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Allowable pressure-temperature curves are generated for steady-state (zero-rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) across the vessel wall developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures could be lower if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the inside 1/4T flaw during heatup is lower than the K_{Ic} for the flaw during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The third portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the least of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

6.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G [Ref. 4] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure head regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure, which is calculated to be 621 psig. The initial RT_{NDT} values of the reactor vessel closure head and vessel flange are documented in Table 3-2. The limiting unirradiated RT_{NDT} of -22°F is associated with the vessel flange of the Watts Bar Unit 2 vessel, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig (without margins for instrument uncertainties). Consistent with the previous P-T limit curves [Ref. 13], the minimum allowable temperature of the flange region was rounded up to 100°F. This limit is shown in Figures 8-1 and 8-2.

6.4 BOLTUP TEMPERATURE REQUIREMENTS

The minimum boltup temperature is the minimum allowable temperature at which the reactor vessel closure head bolts can be preloaded. It is determined by the highest reference temperature, RT_{NDT}, in the closure flange region. This requirement is established in Appendix G to 10 CFR 50 [Ref. 4]. Per the NRC-approved methodology in WCAP-14040-A, Revision 4 [Ref. 2], the minimum boltup temperature should be 60°F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher. Since the limiting unirradiated RT_{NDT} of this region is below 60°F per Table 3-2, the minimum boltup temperature for the Watts Bar Unit 2 reactor vessel is 60°F (without margins for instrument uncertainties). This limit is shown in Figures 8-1 and 8-2.

7 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2 [Ref. 1], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (10)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code [Ref. 12]. If measured values of the initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used, provided if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \quad (11)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth:

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \quad (12)$$

where x inches (reactor vessel cylindrical shell beltline thickness is 8.465 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 11 to calculate the ΔRT_{NDT} at the specific depth.

The projected reactor vessel neutron fluence was updated for this analysis and documented in Section 2 of this report. The evaluation methods used in Section 2 are consistent with the methods presented in WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves” [Ref. 2].

Table 7-1 contains the surface fluence values at 32 EFPY, which were used for the development of the P-T limit curves contained in this report. Table 7-1 also contains the 1/4T and 3/4T calculated fluence values and fluence factors (FFs), per Regulatory Guide 1.99, Revision 2. The values in this table will be used to calculate the 32 EFPY ART values for the Watts Bar Unit 2 reactor vessel materials.

Margin is calculated as $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term (σ_I) is 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings when surveillance data is not used or is non-credible, and 8.5°F (half the value) for plates or forgings when credible surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used or is non-credible, and is 14°F (half the value) when credible surveillance capsule data is used. The value for σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Contained in Tables 7-2 and 7-3 are the 32 EFPY ART calculations at the 1/4T and 3/4T locations for generation of the Watts Bar Unit 2 heatup and cooldown curves.

The inlet and outlet nozzle forging weld materials for Watts Bar Unit 2 have projected fluence values that do not exceed the 1 x 10^{17} n/cm^2 fluence threshold at 32 EFPY per Table 2-9. The projected fluence values for the inlet and outlet nozzle forging weld materials provide conservative estimates of the fluence values of the inlet and outlet nozzles at the lowest extent of the nozzle; therefore, per NRC RIS 2014-11 [Ref. 9], neutron radiation embrittlement need not be considered herein for the nozzle forging or weld materials. Thus, ART calculations for the inlet and outlet nozzle forging and weld materials utilizing the 1/4T and 3/4T fluence values are excluded from Tables 7-2 and 7-3, respectively. Limiting ART values for the nozzle forging materials are contained in Appendix B. Finally, the second conclusion of TLR-RES/DE/CIB-2013-01 [Ref. 10] states that if ΔRT_{NDT} is calculated to be less than 25°F, then embrittlement need not be considered. This conclusion was applied, as necessary, to the ART calculations documented in Tables 7-2 and 7-3 for the extended beltline materials.

The limiting ART values for Watts Bar Unit 2 to be used in the generation of the P-T limit curves are based on Intermediate Shell Forging 05 (Position 1.1). In order to provide an additional margin of conservatism, the limiting calculated ART values were rounded and increased by 10°F. The increased limiting ART values, using the “Axial Flaw” methodology, for Intermediate Shell Forging 05 are summarized in Table 7-4.

Table 7-1 Fluence Values and Fluence Factors for the Vessel Surface, 1/4T and 3/4T Locations for the Watts Bar Unit 2 Reactor Vessel Materials at 32 EFPY

Reactor Vessel Material	Surface Fluence, f [(a)] (n/cm^2, E > 1.0 MeV)	1/4T f (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T f (n/cm^2, E > 1.0 MeV)	3/4T FF
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	1.861 x 10^{19}	1.120 x 10^{19}	1.032	0.4055 x 10^{19}	0.7497
Lower Shell Forging 04	1.891 x 10^{19}	1.138 x 10^{19}	1.036	0.4121 x 10^{19}	0.7540
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	1.861 x 10^{19}	1.120 x 10^{19}	1.032	0.4055 x 10^{19}	0.7497
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.1097 x 10^{19}	0.0660 x 10^{19}	0.3390	0.02390 x 10^{19}	0.1919
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	0.1097 x 10^{19}	0.0660 x 10^{19}	0.3390	0.02390 x 10^{19}	0.1919
Lower Shell to Bottom Head Ring Circumferential Weld Seam W04 (Heat # 899680)	0.2454 x 10^{19}	0.1477 x 10^{19}	0.4993	0.05347 x 10^{19}	0.3035
Bottom Head Ring 03	0.2454 x 10^{19}	0.1477 x 10^{19}	0.4993	0.05347 x 10^{19}	0.3035

Note:

(a) 32 EFPY fluence values are documented in Tables 2-7 and 2-10.

Table 7-2 Adjusted Reference Temperature Evaluation for the Watts Bar Unit 2 Reactor Vessel Beltline Materials through 32 EFPY at the 1/4T Location

Reactor Vessel Material and ID Number	Heat Number	CF (°F)	1/4T Fluence (n/cm^2, E > 1.0 MeV)	1/4T FF	$RT_{NDT(U)}$ [a] (°F)	ΔRT_{NDT} [b] (°F)	σ_I [a] (°F)	σ_Δ [c] (°F)	Margin (°F)	ART [d] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	527828	31.0	1.120×10^{19}	1.032	14	32.0	0.0	16.0	32.0	78.0
Lower Shell Forging 04	528658	31.0	1.138×10^{19}	1.036	5	32.1	0.0	16.1	32.1	69.2
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	68.0	1.120×10^{19}	1.032	-50	70.2	0.0	28.0	56.0	76.2
Using credible Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance data	895075	32.3	1.120×10^{19}	1.032	-50	33.3	0.0	14.0	28.0	11.3
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	411572	44.0	0.0660×10^{19}	0.3390	-14	0.0 (14.9)	0.0	0.0	0.0	-14
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	0.0660×10^{19}	0.3390	10	0.0 (13.9)	0.0	0.0	0.0	10
Lower Shell to Bottom Head Ring Weld Seam W04	899680	41.0	0.1477×10^{19}	0.4993	10	0.0 (20.5)	0.0	0.0	0.0	10
Bottom Head Ring 03	5329	37.0	0.1477×10^{19}	0.4993	-40	0.0 (18.5)	0.0	0.0	0.0	-40

Notes:

(a) Initial RT_{NDT} values taken from Table 3-1. All initial RT_{NDT} values are measured value; thus, $\sigma_I = 0°F$ for each material.

(b) As discussed in Section 7, calculated extended beltline ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 10]. Actual calculated ΔRT_{NDT} values are listed in parentheses.

(c) As discussed in Section 4, the surveillance weld data for Heat # 895075 data was deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 1], the base metal $\sigma_\Delta = 17°F$ for Position 1.1. Also per Regulatory Guide 1.99, Revision 2 [Ref. 1], the weld metal $\sigma_\Delta = 28°F$ for Position 1.1, and the weld metal $\sigma_\Delta = 14°F$ for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(d) The Regulatory Guide 1.99, Revision 2 [Ref. 1] methodology was used to calculate ART values. $ART = RT_{NDT(U)} + \Delta RT_{NDT} + Margin$.

Table 7-3 Adjusted Reference Temperature Evaluation for the Watts Bar Unit 2 Reactor Vessel Beltline Materials through 32 EFPY at the 3/4T Location

Reactor Vessel Material and ID Number	Heat Number	CF (°F)	3/4T Fluence (n/cm^2, E > 1.0 MeV)	3/4T FF	$RT_{NDT(U)}$ [a] (°F)	ΔRT_{NDT} [b] (°F)	σ_I [a] (°F)	σ_Δ [c] (°F)	Margin (°F)	ART [d] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	527828	31.0	0.4055×10^{19}	0.7497	14	23.2	0.0	11.6	23.2	60.5
Lower Shell Forging 04	528658	31.0	0.4121×10^{19}	0.7540	5	23.4	0.0	11.7	23.4	51.7
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	68.0	0.4055×10^{19}	0.7497	-50	51.0	0.0	25.5	51.0	52.0
Using credible Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance data	895075	32.3	0.4055×10^{19}	0.7497	-50	24.2	0.0	12.1	24.2	-1.6
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	411572	44.0	0.02390×10^{19}	0.1919	-14	0.0 (8.4)	0.0	0.0	0.0	-14
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	0.02390×10^{19}	0.1919	10	0.0 (7.9)	0.0	0.0	0.0	10
Lower Shell to Bottom Head Ring Weld Seam W04	899680	41.0	0.05347×10^{19}	0.3035	10	0.0 (12.4)	0.0	0.0	0.0	10
Bottom Head Ring 03	5329	37.0	0.05347×10^{19}	0.3035	-40	0.0 (11.2)	0.0	0.0	0.0	-40

Notes:

(a) Initial RT_{NDT} values taken from Table 3-1. All initial RT_{NDT} values are measured value; thus, $\sigma_I = 0°F$ for each material.

(b) As discussed in Section 7, calculated extended beltline ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. 10]. Actual calculated ΔRT_{NDT} values are listed in parentheses.

(c) As discussed in Section 4, the surveillance weld data for Heat # 895075 data was deemed credible. Per the guidance of Regulatory Guide 1.99, Revision 2 [Ref. 1], the base metal $\sigma_\Delta = 17°F$ for Position 1.1. Also per Regulatory Guide 1.99, Revision 2 [Ref. 1], the weld metal $\sigma_\Delta = 28°F$ for Position 1.1, and the weld metal $\sigma_\Delta = 14°F$ for Position 2.1 with credible surveillance data. However, σ_Δ need not exceed $0.5*\Delta RT_{NDT}$.

(d) The Regulatory Guide 1.99, Revision 2 [Ref. 1] methodology was used to calculate ART values. $ART = RT_{NDT(U)} + \Delta RT_{NDT} + Margin$.

Table 7-4 Summary of the Increased Limiting ART Values Used in the Generation of the Watts Bar Unit 2 Heatup and Cooldown Curves at 32 EFPY

1/4T Limiting ART[a]	3/4T Limiting ART[a]
88°F	71°F
Intermediate Shell Forging 05 (Position 1.1)	

Notes:

(a) The ART values used for P-T limit curve development in this report are the limiting ART values calculated in Tables 7-2 and 7-3 rounded and increased by 10°F to add additional margin; this approach is conservative.

8 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel cylindrical beltline region using the methods discussed in Sections 6 and 7 of this report. This approved methodology is also presented in WCAP-14040-A, Revision 4 [Ref. 2].

Figure 8-1 presents the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60 and 100°F/hr applicable for 32 EFPY, with the flange requirements and using the "Axial Flaw" methodology. Figure 8-2 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60, and 100°F/hr applicable for 32 EFPY, with the flange requirements and using the "Axial Flaw" methodology. The heatup and cooldown curves were generated using the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 8-1 and 8-2. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 8-1 (heatup curve only). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in the 1998 through the 2000 Addenda ASME Code Section XI, Appendix G as follows:

$$1.5\ K_{Im} < K_{Ic} \qquad (13)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress [see page 6-2, Equation (3)],

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T\ -\ RT_{NDT})]}$ [see page 6-1 Equation (1)],

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation in order to provide additional margin during actual power production. The pressure-temperature limits for core operation (except for low power physics tests) are that: 1) the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and 2) the reactor vessel must be at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 6 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the inservice hydrostatic leak tests for the Watts Bar Unit 2 reactor vessel at 32 EFPY is 149°F. This temperature is the minimum permissible temperature at which design pressure can be reached during a hydrostatic test per Equation (13). The vertical line drawn from these points on the pressure-temperature

curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 8-1 and 8-2 define all of the above limits for ensuring prevention of non-ductile failure for the Watts Bar Unit 2 reactor vessel for 32 EFPY with the flange requirements and without instrumentation uncertainties. The data points used for developing the heatup and cooldown P-T limit curves shown in Figures 8-1 and 8-2 are presented in Tables 8-1 and 8-2. The P-T limit curves shown in Figures 8-1 and 8-2 were generated based on the limiting ART values for the cylindrical beltline and extended beltline reactor vessel materials rounded and increased by 10°F to add additional margin; this approach is conservative. As discussed in Appendix B, the P-T limits developed for the cylindrical beltline region bound the P-T limits for the reactor vessel inlet and outlet nozzles for Watts Bar Unit 2 at 32 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Forging 05 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 32 EFPY: 1/4T, 88°F (Axial Flaw)
3/4T, 71°F (Axial Flaw)

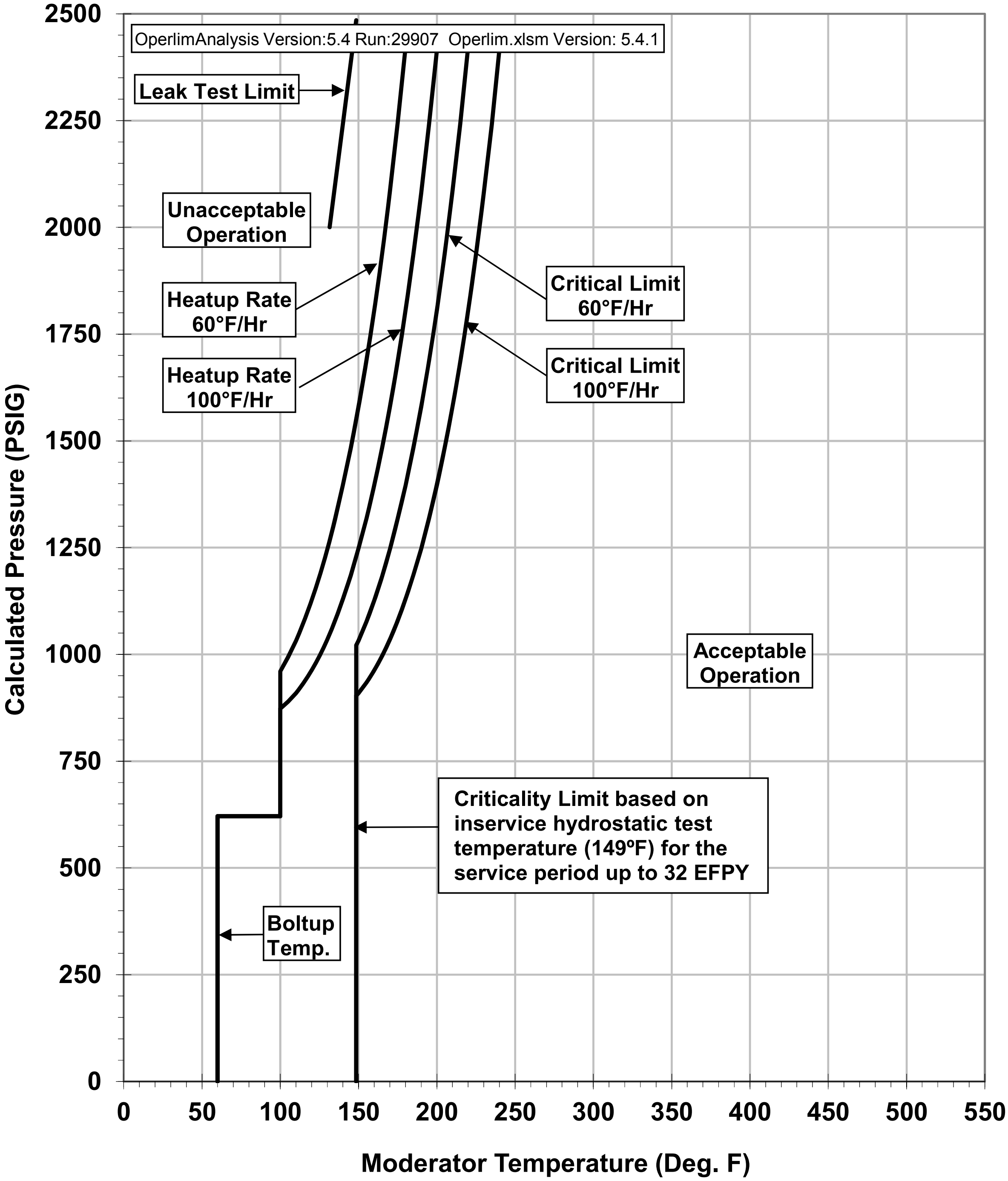

Figure 8-1 Watts Bar Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 32 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate Shell Forging 05 using Regulatory Guide 1.99 Position 1.1 data
LIMITING ART VALUES AT 32 EFPY: 1/4T, 88°F (Axial Flaw)
3/4T, 71°F (Axial Flaw)

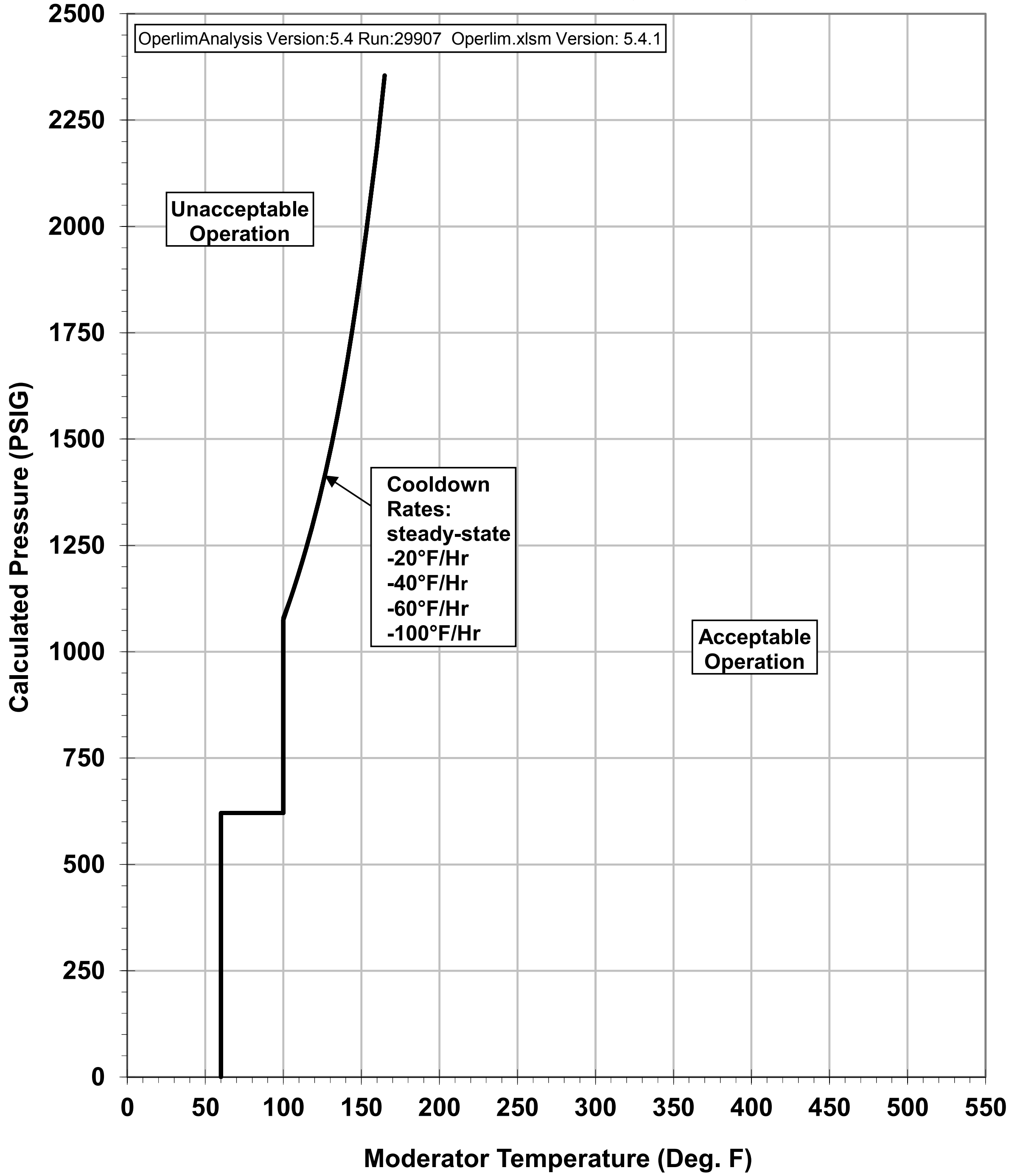

Figure 8-2 Watts Bar Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60, and 100°F/hr) Applicable for 32 EFPY (with Flange Requirements and without Margins for Instrumentation Errors) using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic})

Table 8-1 Watts Bar Unit 2 32 EFPY Heatup Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	149	0	60	0	149	0
60	621	149	1021	60	621	149	904
65	621	150	1032	65	621	150	909
70	621	155	1076	70	621	155	934
75	621	160	1126	75	621	160	963
80	621	165	1183	80	621	165	996
85	621	170	1246	85	621	170	1035
90	621	175	1317	90	621	175	1079
95	621	180	1395	95	621	180	1129
100	621	185	1483	100	621	185	1185
100	960	190	1580	100	873	190	1248
105	993	195	1687	105	889	195	1318
110	1032	200	1806	110	909	200	1396
115	1076	205	1938	115	934	205	1483
120	1126	210	2083	120	963	210	1580
125	1183	215	2244	125	996	215	1686
130	1246	220	2421	130	1035	220	1805
135	1317			135	1079	225	1936
140	1395			140	1129	230	2080
145	1483			145	1185	235	2240
150	1580			150	1248	240	2417
155	1687			155	1318		
160	1806			160	1396		
165	1938			165	1483		
170	2083			170	1580		
175	2244			175	1686		
180	2421			180	1805		
				185	1936		
				190	2080		
				195	2240		
				200	2417		

Leak Test Limit	
T (°F)	P (psig)
132	2000
149	2485

Table 8-2 Watts Bar Unit 2 32 EFPY Cooldown Curve Data Points using the 1998 through the 2000 Addenda App. G Methodology (w/ K_{Ic}, w/ Flange Requirements, and w/o Margins for Instrumentation Errors)

Steady-State		20°F/hr Cooldown		40°F/hr Cooldown		60°F/hr Cooldown		100°F/hr Cooldown	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	621	60	621
65	621	65	621	65	621	65	621	65	621
70	621	70	621	70	621	70	621	70	621
75	621	75	621	75	621	75	621	75	621
80	621	80	621	80	621	80	621	80	621
85	621	85	621	85	621	85	621	85	621
90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621
100	1080	100	1075	100	1075	100	1075	100	1075
105	1130	105	1130	105	1130	105	1130	105	1130
110	1185	110	1185	110	1185	110	1185	110	1185
115	1247	115	1247	115	1247	115	1247	115	1247
120	1315	120	1315	120	1315	120	1315	120	1315
125	1390	125	1390	125	1390	125	1390	125	1390
130	1473	130	1473	130	1473	130	1473	130	1473
135	1564	135	1564	135	1564	135	1564	135	1564
140	1665	140	1665	140	1665	140	1665	140	1665
145	1777	145	1777	145	1777	145	1777	145	1777
150	1901	150	1901	150	1901	150	1901	150	1901
155	2037	155	2037	155	2037	155	2037	155	2037
160	2188	160	2188	160	2188	160	2188	160	2188
165	2355	165	2355	165	2355	165	2355	165	2355

9 REFERENCES

1. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.
2. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.
3. Appendix G to the 1998 through the 2000 Addenda Edition of the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure."
4. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.
5. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.
6. Westinghouse Report WCAP-16083-NP, Revision 1, "Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," April 2013.
7. RSICC Computer Code Collection CCC-650, "DOORS 3.2: One, Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System," April 1998.
8. RSICC Data Library Collection DLC-185, "BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996.
9. NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[Agencywide Document Management System (ADAMS) Accession Number ML14149A165]*
10. U.S. NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], November 2014. *[ADAMS Accession Number ML14318A177]*
11. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.
12. ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, "Class 1 Components."
13. Westinghouse Report WCAP-17035-NP, Revision 2, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," December 2009.
14. Westinghouse Report WCAP-17455-NP, Revision 0, "McGuire Units 1 and 2 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," February 2012.
15. Westinghouse Report WCAP-17669-NP, Revision 1, "Catawba Unit 1 Measurement Uncertainty Recapture (MUR) Power Uprate: Reactor Vessel Integrity and Neutron Fluence Evaluations," October 2015.
16. Westinghouse Report WCAP-9455, Revision 3, "Tennessee Valley Authority Watts Bar Unit No. 2 Reactor Vessel Radiation Surveillance Program," September 2009.

17. Westinghouse Report WCAP-13830, Revision 1, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 2," February 1995.

18. K. Wichman, M. Mitchell, and A. Hiser, U.S. NRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, *NRC/Industry Workshop on RPV Integrity Issues*, February 12, 1998. *[ADAMS Accession Number ML110070570]*

19. ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.

APPENDIX A THERMAL STRESS INTENSITY FACTORS (K_{It})

Tables A-1 and A-2 contain the thermal stress intensity factors (K_{It}) for the maximum heatup and cooldown rates at 32 EFPY for Watts Bar Unit 2. The reactor vessel cylindrical shell radii to the 1/4T and 3/4T locations are as follows:

- 1/4T Radius = 88.768 inches
- 3/4T Radius = 93.001 inches

Table A-1 K_{It} Values for Watts Bar Unit 2 at 32 EFPY 100°F/hr Heatup Curves (w/ Flange Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (ksi √in.)	Vessel Temperature at 3/4T Location for 100°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (ksi √in.)
60	56.015	-0.994	55.047	0.478
65	58.635	-2.438	55.318	1.443
70	61.728	-3.675	56.029	2.419
75	65.038	-4.846	57.225	3.331
80	68.620	-5.851	58.848	4.142
85	72.314	-6.766	60.858	4.864
90	76.193	-7.556	63.213	5.501
95	80.170	-8.276	65.868	6.069
100	84.280	-8.903	68.790	6.572
105	88.475	-9.472	71.945	7.019
110	92.767	-9.970	75.302	7.416
115	97.129	-10.424	78.839	7.773
120	101.561	-10.823	82.530	8.092
125	106.051	-11.189	86.359	8.379
130	110.592	-11.512	90.308	8.637
135	115.181	-11.809	94.363	8.870
140	119.806	-12.074	98.512	9.080
145	124.470	-12.318	102.742	9.271
150	129.161	-12.537	107.045	9.445
155	133.883	-12.740	111.410	9.603
160	138.625	-12.924	115.832	9.748
165	143.391	-13.096	120.303	9.882
170	148.173	-13.252	124.817	10.005
175	152.974	-13.399	129.369	10.119
180	157.787	-13.534	133.955	10.225
185	162.615	-13.662	138.570	10.324
190	167.452	-13.780	143.211	10.417
195	172.301	-13.894	147.875	10.504
200	177.156	-13.999	152.559	10.587
205	182.021	-14.101	157.261	10.665
210	186.891	-14.197	161.979	10.739

Table A-2 K_{It} Values for Watts Bar Unit 2 at 32 EFPY 100°F/hr Cooldown Curves (w/ Flange Requirements, and w/o Margins for Instrument Errors)

Water Temp. (°F)	Vessel Temperature at 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (ksi √in.)
210	236.000	16.310
205	230.917	16.244
200	225.833	16.178
195	220.750	16.111
190	215.666	16.046
185	210.582	15.979
180	205.497	15.913
175	200.413	15.846
170	195.329	15.780
165	190.244	15.713
160	185.160	15.647
155	180.075	15.581
150	174.990	15.514
145	169.906	15.448
140	164.821	15.382
135	159.737	15.316
130	154.653	15.250
125	149.568	15.183
120	144.484	15.118
115	139.400	15.052
110	134.316	14.986
105	129.232	14.921
100	124.148	14.855
95	119.064	14.790
90	113.981	14.725
85	108.897	14.659
80	103.814	14.595
75	98.731	14.530
70	93.647	14.465
65	88.565	14.401
60	83.483	14.336

APPENDIX B REACTOR VESSEL INLET AND OUTLET NOZZLES

As described in NRC Regulatory Issue Summary (RIS) 2014-11 [Ref. B-1], reactor vessel non-beltline materials may define pressure-temperature (P-T) limit curves that are more limiting than those calculated for the reactor vessel cylindrical shell beltline materials. Reactor vessel nozzles, penetrations, and other discontinuities have complex geometries that can exhibit significantly higher stresses than those for the reactor vessel beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperatures (RT_{NDT}) for these components are not as high as those of the reactor vessel beltline shell materials that have simpler geometries.

The methodology contained in WCAP-14040-A, Revision 4 [Ref. B-2] was used in the main body of this report to develop P-T limit curves for the limiting Watts Bar Unit 2 cylindrical shell beltline material; however, WCAP-14040-A, Revision 4 does not consider ferritic materials in the area adjacent to the beltline, specifically the stressed inlet and outlet nozzles. Due to the geometric discontinuity, the inside corner regions of these nozzles are the most highly stressed ferritic component outside the beltline region of the reactor vessel; therefore, these components are analyzed in this Appendix. P-T limit curves are determined for the reactor vessel nozzle corner region for Watts Bar Unit 2 and compared to the P-T limit curves for the reactor vessel traditional beltline region in order to determine if the nozzles can be more limiting than the reactor vessel beltline as the plant ages and the vessel accumulates more neutron fluence. The increase in neutron fluence as the plant ages causes a concern for embrittlement of the reactor vessel above the beltline region. Therefore, the P-T limit curves are developed for the nozzle inside corner region since the geometric discontinuity results in high stresses due to internal pressure and the cooldown transient. The cooldown transient is analyzed as it results in tensile stresses at the inside surface of the nozzle corner.

A 1/4T axial flaw is postulated at the inside surface of the reactor vessel nozzle corner, and stress intensity factors are determined based on the rounded curvature of the nozzle geometry. The allowable pressure is then calculated based on the fracture toughness of the nozzle material and the stress intensity factors for the 1/4T flaw.

B.1 CALCULATION OF ADJUSTED REFERENCE TEMPERATURES

The fracture toughness (K_{Ic}) used for the inlet and outlet nozzle material is defined in Appendix G of the Section XI ASME Code, as discussed in Section 6 of this report. The K_{Ic} fracture toughness curve is dependent on the Adjusted Reference Temperature (ART) value for irradiated materials. The ART values for the inlet and outlet nozzle materials are determined using the methodology contained in Regulatory Guide 1.99, Revision 2 [Ref. B-3], which is described in Section 7 of this report, as well as weight percent (wt. %) copper (Cu) and nickel (Ni) values, initial RT_{NDT} values, and projected neutron fluence as inputs. The material properties for each of the reactor vessel inlet and outlet nozzle forging materials are documented in Table B-1 and a summary of the limiting inlet and outlet nozzle ART values for Watts Bar Unit 2 is presented in Table B-2.

Nozzle Material Properties

The Watts Bar Unit 2 nozzle material properties are provided in Table B-1. Copper (Cu), Nickel (Ni), Manganese (Mn), and Phosphorus (P) weight percent (wt. %) values were obtained from the Watts Bar Unit 2 CMTRs for each of the Watts Bar Unit 2 reactor vessel inlet and outlet nozzles.

The Charpy V-Notch forging specimen orientation for the inlet and outlet nozzles was not reported in the Watts Bar Unit 2 CMTRs; thus, it was conservatively assumed that the orientation was the "strong direction" for each nozzle forging. Due to the assumed lack of transverse test data, the initial RT_{NDT} values were determined for each of the Watts Bar Unit 2 reactor vessel inlet and outlet nozzle forging materials using the Branch Technical Position (BTP) 5-3, Position 1.1(3) methodology [Ref. B-4]. The initial RT_{NDT} values for all of the nozzle materials were determined directly from the data or by using CVGRAPH, Version 6.02 hyperbolic tangent curve fits through the minimum data points, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraph (a)(4) [Ref. B-5]. The initial RT_{NDT} values were determined using both BTP 5-3 Position 1.1(3)(a) and Position 1.1(3)(b), and the more limiting initial RT_{NDT} value was chosen for each nozzle forging material. The initial USE values were determined in accordance with the methodology described in ASTM E185-82 [Ref. B-6]. For each of the nozzle forging materials, use of BTP 5-3 Paragraph 1.2 [Ref. B-5] was necessary. The Watts Bar Unit 2 initial RT_{NDT} and initial USE values for the inlet and outlet nozzles materials are summarized in Table B-1.

Nozzle Calculated Neutron Fluence Values

The maximum fast neutron (E > 1 MeV) exposure of the Watts Bar Unit 2 reactor vessel materials is discussed in Section 2 of this report. The fluence values used in the inlet and outlet nozzle ART calculations were calculated at the lowest extent of the nozzles (i.e., the nozzle to nozzle shell weld locations) and were chosen at an elevation lower than the actual elevation of the postulated flaw, which is at the inside corner of the nozzle, for conservatism.

Per Table 2-9, the inlet nozzles are determined to receive a projected maximum fluence of 8.861×10^{16} n/cm^2 (E > 1 MeV) at the lowest extent of the nozzles at 32 EFPY. Similarly, the outlet nozzles are projected to achieve a maximum fluence value of 4.630×10^{16} n/cm^2 (E > 1 MeV) at the lowest extent of the nozzles at 32 EFPY. Thus, the maximum neutron fluence values for the nozzle materials are not projected to exceed a fluence of 1×10^{17} n/cm^2 (E > 1.0 MeV) at 32 EFPY. Per NRC RIS 2014-11 [Ref. B-1], embrittlement of reactor vessel materials, with projected fluence values less than 1×10^{17} n/cm^2 (E > 1.0 MeV), does not need to be considered. Therefore, the initial RT_{NDT} values documented in Table B-1 are identical to the nozzle ART values.

The neutron fluence values used in the ART calculations for the Watts Bar Unit 2 inlet and outlet nozzle forging materials are summarized in Table B-1. The limiting nozzle ART values used for determination of the nozzle P-T limit curves are summarized in Table B-2.

The use of the embrittlement conclusion of NRC RIS 2014-11 [Ref. B-1], and thus the limiting ART values summarized in Table B-2, will remain unchanged as long as the fluence values assigned to the inlet and outlet nozzles remain below 1.0×10^{17} n/cm^2 (E > 1.0 MeV). If these fluence values are reached, the Watts Bar Unit 2 nozzle material ART values should be re-evaluated.

Table B-1 Summary of the Watts Bar Unit 2 Reactor Vessel Nozzle Material Initial RT_{NDT}, Chemistry, and Fluence Values at 32 EFPY

Reactor Vessel Material	Chemical Composition[a]				Fracture Toughness Properties[b]		Fluence at Lowest Extent of Nozzle[d] (n/cm^2, E > 1.0 MeV)
	Wt. % Cu	Wt. % Ni	Wt. % Mn	Wt. % P	$RT_{NDT(U)}$ (°F)	Initial USE (ft-lb)	
Inlet Nozzle 11 (Heat # 5328)	0.05	0.83	0.75	0.008	-22	78	8.861×10^{16}
Inlet Nozzle 12 (Heat # 5330)	0.06	0.85	0.77	0.011	-14	67	8.861×10^{16}
Inlet Nozzle 13 (Heat # 5331)	0.06	0.82	0.75	0.010	-8	61	8.861×10^{16}
Inlet Nozzle 14 (Heat # 5335)	0.04	0.79	0.77	0.009	-13	87	8.861×10^{16}
Outlet Nozzle 15 (Heat # 5319)	0.06	0.86	0.69	0.009	-22	90	4.630×10^{16}
Outlet Nozzle 16 (Heat # 5324)	0.06	0.84	0.71	0.011	-13	72	4.630×10^{16}
Outlet Nozzle 17 (Heat # 5327)	0.05	0.86	0.76	0.009	-39	84	4.630×10^{16}
Outlet Nozzle 18 (Heat # 5334)	0.04	0.80	0.77	0.009	-31	>84[c]	4.630×10^{16}

Notes:

(a) Chemistry values were taken from the Watts Bar Unit 2 CMTRs.

(b) The nozzle 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 [Ref. B-4] Position 1.1(3)(a) and (b), with the more limiting $RT_{NDT(U)}$ value being selected. 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 Position 1.2 methodology, unless otherwise noted.

(c) 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%, reduced to 65% of its original value per NUREG-0800, BTP 5-3 Position 1.2 methodology.

(d) Fluence values conservatively correspond to 32 EFPY fluence values at the lowest extent of the nozzle weld.

Table B-2 Summary of the Limiting ART Values for the Watts Bar Unit 2 Inlet and Outlet Nozzle Materials

EFPY	Nozzle Material and ID Number	Limiting ART Value (°F)
32	Inlet Nozzle 13 (Heat # 5331)	-8
	Outlet Nozzle 16 (Heat # 5324)	-13

B.2 NOZZLE COOLDOWN PRESSURE-TEMPERATURE LIMITS

Allowable pressures are determined for a given temperature based on the fracture toughness of the limiting nozzle material along with the appropriate pressure and thermal stress intensity factors. The Watts Bar Unit 2 nozzle fracture toughness used to determine the P-T limits is calculated using the limiting inlet and outlet nozzle ART values from Table B-2. The stress intensity factor correlations used for the nozzle corners are provided in Oak Ridge National Laboratory study, ORNL/TM-2010/246 [Ref. B-7], and are consistent with ASME PVP2011-57015 [Ref. B-8]. The methodology includes postulating an inside surface 1/4T nozzle corner flaw, and calculating through-wall nozzle corner stresses for a cooldown rate of 100ºF/hour.

The through-wall stresses at the nozzle corner location were fitted based on a third-order polynomial of the form:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

where,

σ = through-wall stress distribution

x = through-wall distance from inside surface

A_0, A_1, A_2, A_3 = coefficients of polynomial fit for the third-order polynomial, used in the stress intensity factor expression discussed below

The stress intensity factors generated for a rounded nozzle corner for the pressure and thermal gradient were calculated based on the methodology provided in ORNL/TM-2010/246. The stress intensity factor expression for a rounded corner is:

$$K_I = \sqrt{\pi a}\left[0.706A_0 + 0.537\left(\frac{2a}{\pi}\right)A_1 + 0.448\left(\frac{a^2}{2}\right)A_2 + 0.393\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

where,

K_I = stress intensity factor for a circular corner crack on a nozzle with a rounded inner radius corner

a = crack depth at the nozzle corner, for use with 1/4T (25% of the wall thickness)

The Watts Bar Unit 2 reactor vessel inlet and outlet nozzle P-T limit curves are shown in Figures B-1 and B-2, respectively, based on the stress intensity factor expression discussed above; also shown in these figures are the traditional cylindrical beltline cooldown P-T limit curves from Figure 8-2. The nozzle P-T limit curves are provided for a cooldown rate of 100°F/hr, along with a steady-state curve.

An outside surface flaw in the nozzle was not considered because the pressure stress is significantly lower at the outside surface than the inside surface. A heatup nozzle P-T limit curve is also not provided since it would be less limiting than the cooldown nozzle P-T limit curve in Figures B-1 and B-2 for an inside surface flaw. Additionally, the cooldown transient is more limiting than the heatup transient since it results in tensile stresses at the inside surface of the nozzle corner.

Conclusion

Based on the results shown in Figures B-1 and B-2, it is concluded that the nozzle P-T limits are bounded by the traditional cylindrical beltline curves. Therefore, the P-T limits provided in Section 8 for 32 EFPY remain limiting for the beltline and non-beltline reactor vessel components.

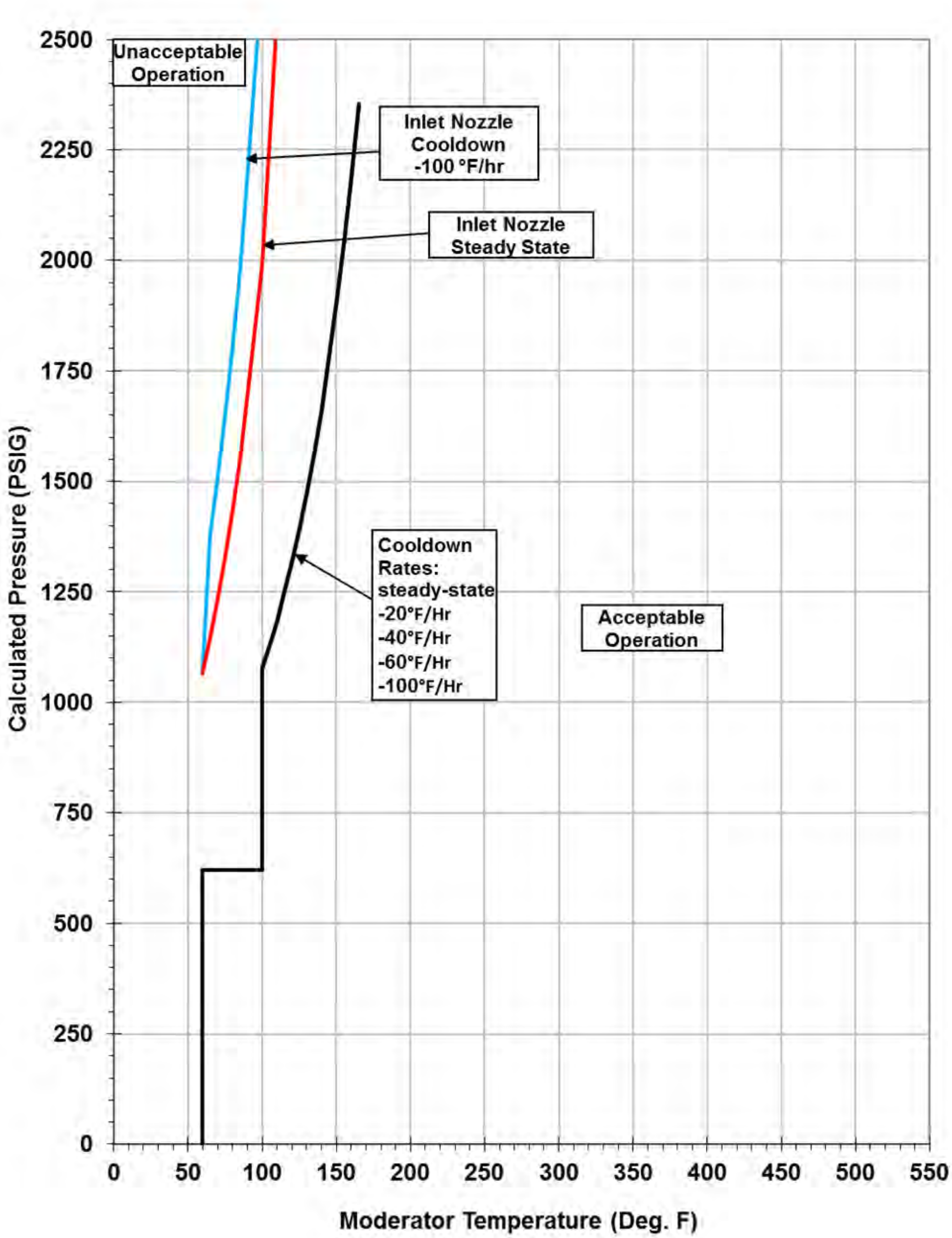

Figure B-1 Comparison of Watts Bar Unit 2 32 EFPY Beltline P-T Limits to 32 EFPY Inlet Nozzle P-T Limits, Without Margins for Instrumentation Errors

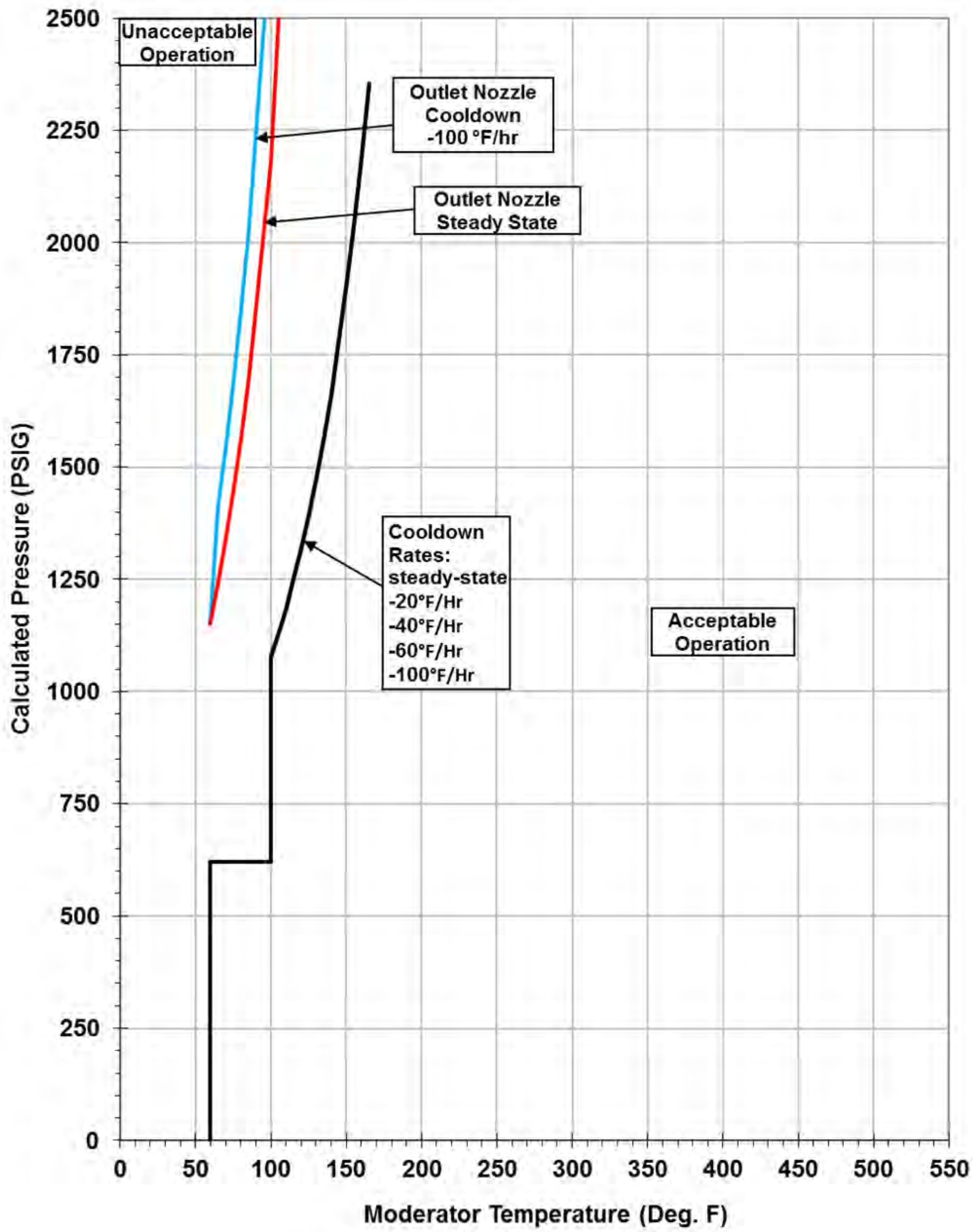

Figure B-2 Comparison of Watts Bar Unit 2 32 EFPY Beltline P-T Limits to 32 EFPY Outlet Nozzle P-T Limits, Without Margins for Instrumentation Errors

B.3 REFERENCES

B-1 NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[ADAMS Accession Number ML14149A165]*

B-2 Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

B-3 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

B-4 NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 5 LWR Edition, Branch Technical Position 5-3, "Fracture Toughness Requirements," Revision 2, U.S. Nuclear Regulatory Commission, March 2007.

B-5 ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, "Class 1 Components."

B-6 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.

B-7 Oak Ridge National Laboratory Report, ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1," June 2012. *[ADAMS Accession Number ML110060164]*

B-8 ASME PVP2011-57015, "Additional Improvements to Appendix G of ASME Section XI Code for Nozzles," G. Stevens, H. Mehta, T. Griesbach, D. Sommerville, July 2011.

APPENDIX C OTHER RCPB FERRITIC COMPONENTS

10 CFR Part 50, Appendix G [Ref. C-1], requires that all Reactor Coolant Pressure Boundary (RCPB) components meet the requirements of Section III of the ASME Code. The lowest service temperature requirement (LST) for all RCPB components, which is specified in NB-3211 and NB-2332(b) of the Section III ASME Code, is the relevant requirement that would affect the pressure-temperature (P-T) limits. This requirement is applicable to ferritic materials outside of the reactor vessel with a nominal wall thickness greater than 2 ½ inches, such as piping, pumps and valves [Ref. C-2]. The Watts Bar Unit 2 reactor coolant system components do not contain ferritic materials in the Class 1 piping, pumps and valves. Therefore, the LST requirements of NB-2332(b) and NB-3211 are not applicable to the Watts Bar Unit 2 P-T limits.

The other ferritic RCPB components that are not part of the reactor vessel beltline or extended beltline consist of the reactor vessel closure head, pressurizer, steam generators, and future replacement steam generators.

The reactor vessel closure head materials have been considered in the development of P-T limits, see Section 6.3 of this report for further detail. Furthermore, the reactor vessel closure head was constructed to the 1971 Edition through 1971 Winter Addenda Section III ASME Code and has met all applicable requirements at the time of construction.

The original steam generators and pressurizer were constructed to the 1971 Edition through 1971 Summer Addenda Section III ASME Code and have met all applicable requirements at the time of construction. Furthermore, the steam generators and pressurizer have not undergone neutron embrittlement that would affect P-T limits. Therefore, no further consideration is necessary for this component with regards to P-T limits.

The future replacement steam generators have been designed to the 1989 Edition Section III ASME Code and should meet all applicable requirements at the time of construction. No further consideration is necessary for these components with regards to P-T limits.

C.1 REFERENCES

C-1 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

C-2 ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NB, "Class 1 Components."

APPENDIX D UPPER-SHELF ENERGY EVALUATION

Charpy upper-shelf energy (USE) is associated with the determination of acceptable reactor pressure vessel (RPV) toughness during the licensed operating period.

The requirements on USE are included in 10 CFR 50, Appendix G [Ref. D-1]. 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 Regulatory Guide 1.99, Revision 2 [Ref. D-2]. For vessel 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 Regulatory Guide 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 Regulatory Guide to predict the change in USE (Position 2.2) of the RPV material due to irradiation.

The 32 EFPY (EOL) 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 Regulatory Guide 1.99, Revision 2.

The predicted Position 2.2 USE values are determined for the reactor vessel materials that are contained in the surveillance program by using the reduced plant surveillance data along with the corresponding 1/4T fluence projection. However, since no Watts Bar Unit 2 capsules have been tested, no Watts Bar Unit 2 surveillance data yet exists. Therefore, no percent USE reductions are calculated based on Regulatory Guide 1.99, Revision 2, Position 2.2.

The projected USE values were calculated to determine if the Watts Bar Unit 2 beltline and extended beltline materials remain above the 50 ft-lb criterion at 32 EFPY (EOL). These calculations are summarized in Table D-1.

USE Conclusion

For Watts Bar Unit 2, the limiting USE value at 32 EFPY is 72.0 ft-lb (see Table D-1); this value corresponds to Intermediate Shell Forging 05. Therefore, all of the beltline and extended beltline materials in the Watts Bar Unit 2 reactor vessel are projected to remain above the USE screening criterion value of 50 ft-lb (per 10 CFR 50, Appendix G) through EOL (32 EFPY).

Table D-1 Predicted Position 1.2 USE Values at 32 EFPY (EOL) for Watts Bar Unit 2

Reactor Vessel Material and Identification Number	Wt % Cu[(a)]	EOL 1/4T Fluence[(b)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Initial USE[(a)] (ft-lb)	Projected USE Decrease[(c)] (%)	Projected EOL USE (ft-lb)
Reactor Vessel Beltline Materials					
Intermediate Shell Forging 05	0.05	1.120	90	20.0	72.0
Lower Shell Forging 04	0.05	1.138	105	20.0	84.0
Intermediate to Lower Shell Circumferential Weld Seam W05 (Heat # 895075)	0.05	1.120	127	20.0	101.6
Reactor Vessel Extended Beltline Materials					
Upper Shell Forging 06	0.07	0.0660	94	10.0	84.6
Upper to Intermediate Shell Circumferential Weld Seam W06 (Heat # 899680)	0.03	0.0660	92	10.0	82.8
Lower Shell to Bottom Head Ring Weld Seam W04 (Heat # 899680)	0.03	0.1477	92	12.0	81.0
Bottom Head Ring 03	0.06	0.1477	105	12.0	92.4

Notes:

(a) Data taken from Table 3-1 of this report.

(b) The 1/4T fluence was calculated using the Regulatory Guide 1.99, Revision 2 [Ref. D-2] correlation, and the Watts Bar Unit 2 reactor vessel beltline wall thickness of 8.465 inches.

(c) Percentage USE decrease values are based on Position 1.2 of Regulatory Guide 1.99, Revision 2 [Ref. D-2] and were calculated by plotting the 1/4T fluence values on Figure 2 of the Regulatory Guide and using the material-specific Cu wt. % values. In calculating Position 1.2 percent USE decreases, the base metal and weld Cu weight percentages were conservatively rounded up to the nearest line in Regulatory Guide 1.99, Revision 2, Figure 2.

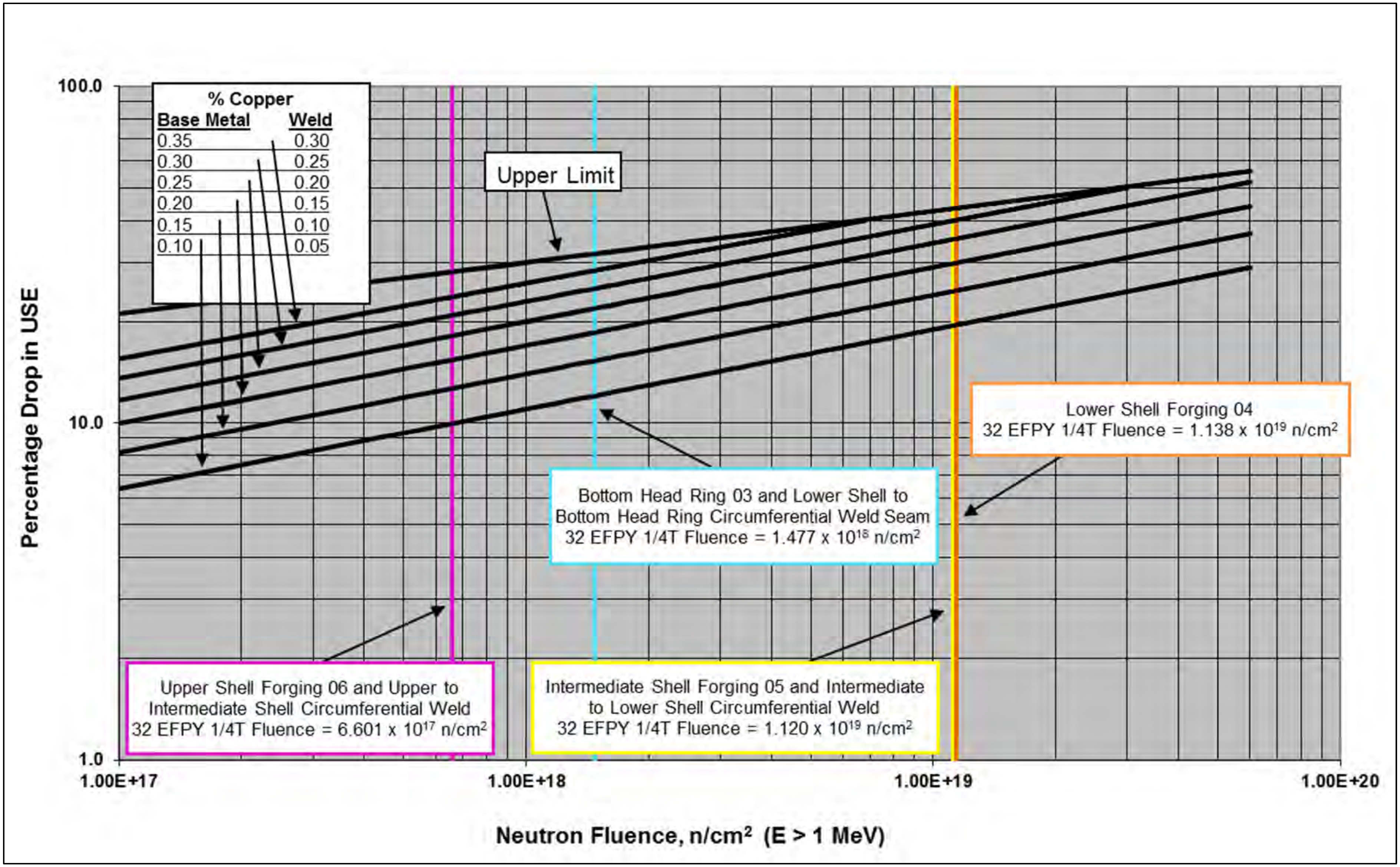

Figure D-1 Regulatory Guide 1.99, Revision 2 Predicted Decrease in Upper-Shelf Energy as a Function of Copper and Fluence

D.1 REFERENCES

D-1 Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

D-2 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

APPENDIX E PRESSURIZED THERMAL SHOCK AND EMERGENCY RESPONSE GUIDELINE LIMITS EVALUATION

E.1 PRESSURIZED THERMAL SHOCK

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 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. E-1]) 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 Regulatory Guide 1.99, Revision 2 [Ref. E-2].

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 Unit 2 RPV materials at 32 EFPY (EOL). The EOL RT_{PTS} calculations are summarized in Table E-1.

The following two conclusions from Section 4 of TLR-RES/DE/CIB-2013-01 [Ref. E-3] were utilized, as appropriate:

1. *The beltline is defined as the region of the RPV adjacent to the reactor core that is projected to receive a neutron fluence level of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) or higher at the end of the licensed operating period.*
2. *Embrittlement effects may be neglected for any region of the RPV if either of the following conditions are met: (1) neutron fluence is less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL, or (2) the mean value of ΔT_{30} estimated using an ETC acceptable to the staff is less than 25°F at EOL. The estimate of ΔT_{30} at EOL shall be made using best-estimate chemistry values.*

Therefore, embrittlement of reactor vessel materials with ΔT_{30} (which is equivalent to ΔRT_{NDT}) values less than 25°F need not be considered in the subsequent RT_{PTS} calculations documented in Table E-1.

Table E-1 RT_{PTS} Calculations for the Watts Bar Unit 2 Reactor Vessel Materials at 32 EFPY

Reactor Vessel Material and Identification Number	Heat Number	CF[(a)] (°F)	EOL Fluence[(a)] (x 10^{19} n/cm^2, E > 1.0 MeV)	FF	$RT_{NDT(U)}$[(a)] (°F)	ΔRT_{NDT}[(b)] (°F)	σ_U[(c)] (°F)	σ_Δ[(d)] (°F)	Margin (°F)	RT_{PTS}[(e)] (°F)
Reactor Vessel Beltline Materials										
Intermediate Shell Forging 05	527828	31.0	1.861	1.170	14	36.3	0.0	17.0	34.0	84.3
Lower Shell Forging 04	528658	31.0	1.891	1.174	5	36.4	0.0	17.0	34.0	75.4
Intermediate to Lower Shell Circumferential Weld Seam W05	895075	68.0	1.861	1.170	-50	79.6	0.0	28.0	56.0	85.6
Using credible Catawba Unit 1, Watts Bar Unit 1, and McGuire Unit 2 surveillance data	895075	32.3	1.861	1.170	-50	37.8	0.0	14.0	28.0	15.8
Reactor Vessel Extended Beltline Materials										
Upper Shell Forging 06	411572	44.0	0.1097	0.4356	-14	0.0 (19.2)	0.0	0.0	0.0	-14
Upper to Intermediate Shell Circumferential Weld Seam W06	899680	41.0	0.1097	0.4356	10	0.0 (17.9)	0.0	0.0	0.0	10
Lower Shell to Bottom Head Ring Weld Seam W04	899680	41.0	0.2454	0.6194	10	25.4	0.0	12.7	25.4	60.8
Bottom Head Ring 03	5329	37.0	0.2454	0.6194	-40	0.0 (22.9)	0.0	0.0	0.0	-40

Notes:

(a) CF values were taken from Table 5-2, fluence values were taken from Tables 2-7 and 2-10, and $RT_{NDT(U)}$ values were taken from Table 3-1.

(b) Calculated ΔRT_{NDT} values less than 25°F have been reduced to zero per TLR-RES/DE/CIB-2013-01 [Ref. E-3]. Actual calculated ΔRT_{NDT} values are listed in parentheses for these materials.

(c) The initial RT_{NDT} values are measured values; therefore, σ_U = 0°F.

(d) The credibility conclusion for the surveillance material is discussed in Section 4. The sister-plant weld Heat # 895075 data was deemed credible. Per the guidance of 10 CFR 50.61 [Ref. E-1], the base metal σ_Δ = 17°F for Position 1.1. Also per 10 CFR 50.61 [Ref. E-1], the weld metal σ_Δ = 28°F for Position 1.1, and the weld metal σ_Δ = 14°F for Position 2.1 with credible surveillance data (Heat # 895075). However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.

(e) The 10 CFR 50.61 [Ref. E-1] methodology was utilized in the calculation of the PTS values.

PTS Conclusion

The Watts Bar Unit 2 limiting RT_{PTS} value for base metal or longitudinal weld materials at 32 EFPY is 84.3°F (see Table E-1), which corresponds to Intermediate Shell Forging 05 (using Position 1.1). The Watts Bar Unit 2 limiting RT_{PTS} value for circumferentially oriented welds at 32 EFPY is 85.6°F (see Table E-1), which corresponds to the Intermediate to Lower Shell Circumferential Weld W05 (Heat # 895075, using Position 1.1). However, after applying credible sister-plant surveillance data, the RT_{PTS} value of this material is reduced to 15.8°F. Thus, the actual limiting RT_{PTS} value for circumferentially oriented welds at 32 EFPY is 60.8°F (see Table E-1), which corresponds to the Lower Shell to Bottom Head Ring Circumferential Weld W04 (Heat # 899680, using Position 1.1).

Therefore, all of the beltline and extended beltline materials in the Watts Bar Unit 2 reactor vessel are below the RT_{PTS} screening criteria of 270°F for base metal and/or longitudinal welds, and 300°F for circumferentially oriented welds through EOL (32 EFPY).

The Alternate PTS Rule (10 CFR 50.61a [Ref. E-4]) was published in the Federal Register by the NRC in 2010. This alternate rule is less restrictive than the PTS Rule (10 CFR 50.61) and is intended to be used for situations in which the 10 CFR 50.61 criteria cannot be met. Watts Bar Unit 2 currently meets the criteria for the PTS Rule through EOL and therefore does not need to utilize the Alternate PTS Rule at this time.

E.2 EMERGENCY RESPONSE GUIDELINE LIMITS

The Emergency Response Guideline (ERG) limits were developed to establish guidance for operator action in the event of an emergency situation, such as a PTS event [Ref. E-5]. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. 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, which is calculated the same way as the RT_{PTS} values are calculated in Section E.1 of this report. The material with the highest RT_{NDT} defines the limiting material, which for Watts Bar Unit 2 is the Intermediate Shell Forging 05 (using Position 1.1). Table E-2 identifies ERG category limits and the limiting material RT_{NDT} values at 32 EFPY for Watts Bar Unit 2.

Table E-2 Evaluation of Watts Bar Unit 2 ERG Limit Category

ERG Pressure-Temperature Limits [Ref. E-5]	
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 III b
Limiting RT_{NDT} Value[(b)]	
Reactor Vessel Material	**RT_{NDT} Value @ 32 EFPY**
Intermediate Shell Forging 05 (Position 1.1)	84.3°F

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 E-1.

Per the ERG limit guidance document [Ref. E-5], 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.

Per Table E-2, the limiting material for Watts Bar Unit 2 (Intermediate Shell Forging 05 [Position 1.1]) has an RT_{NDT} less than 200°F through 32 EFPY. Therefore, Watts Bar Unit 2 remains in ERG Category I through EOL (32 EFPY).

Conclusion of ERG P-T Limit Categorization

As summarized above, Watts Bar Unit 2 is currently in ERG Category I and will remain in ERG Category Unit I through EOL (32 EFPY).

E.3 REFERENCES

E-1 Code of Federal Regulations 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, dated December 19, 1995, effective January 18, 1996.

E-2 U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

E-3 U.S. NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Office of Nuclear Regulatory Research [RES], November 2014. *[ADAMS Accession Number ML14318A177]*

E-4 Code of Federal Regulations, 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 75, No. 1, dated January 4, 2010, with corrections dated February 3, 2010 (No. 22), March 8, 2010 (No. 44), and November 26, 2010 (No. 227).

E-5 Westinghouse Owners Group Document HF04BG, "Background Information for Westinghouse Owners Group Emergency Response Guidelines, Critical Safety Function Status Tree, F-0.4 Integrity, HP/LP-Rev. 2," April 2005.

APPENDIX F SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE

The following surveillance capsule removal schedule (Table F-1) meets the recommendations of ASTM E185-82 [Ref. F-1] as required by 10 CFR 50, Appendix H [Ref. F-2]. Note that it is recommended for future capsules to be removed from the Watts Bar Unit 2 reactor vessel.

Table F-1 Surveillance Capsule Withdrawal Schedule

Capsule ID	Capsule Location	Capsule Lead Factor[(a)]	Withdrawal EFPY[(b)]	Capsule Fluence (n/cm^2, E > 1.0 MeV)
U	Dual 34°	4.80	2.61 (EOC 2)	0.7714×10^{19}
W	Single 34°	4.87	6.91 (EOC 5)	1.901×10^{19}
X	Dual 34°	4.80	Note (c)	Note (c)
Z	Single 34°	4.87	Note (d)	Note (d)
V	Dual 31.5°	4.15	Note (d)	Note (d)
Y	Dual 31.5°	4.15	Note (d)	Note (d)

Notes:

(a) Projected capsule fluence values are documented in Table 2-3. The lead factor is the proportional constant by which the capsule fluence leads the vessel's maximum inner wall fluence. This value was calculated using the maximum EOL vessel fluence value and the capsule EOL fluence values documented in Table 2-5 and Table 2-3, respectively.

(b) EFPY from plant startup.

(c) Capsule X must be withdrawn between EOC 6 and 13.5 EFPY in order satisfy the recommendations of the third capsule for EOL per ASTM E185-82 [Ref. F-1]. However, if the capsule is removed after 11.6 EFPY (but still before 13.5 EFPY), this capsule will satisfy the requirements of the third capsule for both EOL (40 years) and end of license extension (60 years) per ASTM E185-82 [Ref. F-1] and NUREG-1801, Revision 2 [Ref. F-3]. Thus, if possible, the capsule should be pulled between 11.6 EFPY and 13.5 EFPY, but the capsule must be pulled between EOC 6 and 13.5 EFPY. The removal EFPY of the third capsule should be revisited at a later date, such as after Capsules U and W are removed.

(d) Capsules Z, V, and Y should remain in the reactor. If additional metallurgical data is needed for Watts Bar Unit 2, such as in support of a second license renewal to 80 total years of operation, withdrawal and testing of these capsules should be considered.

F.1 REFERENCES

F-1 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, July 1982.

F-2 Code of Federal Regulations 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, No. 243, dated December 19, 1995.

F-3 U.S. Nuclear Regulatory Commission, "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Revision 2, December 2010. *[ADAMS Accession Number ML103490041]*

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

Proprietary Holtec Report No: 2177876, "Licensing Report for the Criticality Safety Analysis of the Watts Bar Nuclear Plant Spent Fuel Pool"

This report contains information that Holtec International considers to be proprietary in nature.

Subject: 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)

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

Holtec Report No: 2177950, “Non-Proprietary Licensing Report for the Criticality Safety Analysis of the Watts Bar Nuclear Plant Spent Fuel Pool”

This report does not contain proprietary information.

Subject: 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)

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NON-PROPRIETARY LICENSING REPORT FOR THE CRITICALITY SAFETY ANALYSIS OF THE WATTS BAR NUCLEAR PLANT SPENT FUEL POOL

FOR

TVA

Holtec Report No: HI-2177950

Holtec Project No: 2732

Sponsoring Holtec Division: NPD

Report Class : SAFETY RELATED

HOLTEC INTERNATIONAL

DOCUMENT ISSUANCE AND REVISION STATUS[1]

DOCUMENT NAME:	NON-PROPRIETARY LICENSING REPORT FOR THE CRITICALITY SAFETY ANALYSIS OF THE WATTS BAR NUCLEAR PLANT SPENT FUEL POOL			
DOCUMENT NO.:	HI-2177950	CATEGORY: ☐ GENERIC		
PROJECT NO.:	2732	☑ PROJECT SPECIFIC		

Rev. No.[2]	Date Approved	Author's Initials	VIR #
0	10/20/2017	T.He	971559

DOCUMENT CATEGORIZATION

In accordance with the Holtec Quality Assurance Manual and associated Holtec Quality Procedures (HQPs), this document is categorized as a:

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Summary of Revisions:

Revision 0: Original Issue.

Table of Contents

1. INTRODUCTION

The criticality safety analysis in this report documents the criticality calculations performed for Tennessee Valley Authority (TVA) for the Watts Bar Nuclear (WBN) Plant spent fuel pool (SFP). The WBN SFP contains a single type of Boral racks with flux traps designed for storage of PWR fuel. The criticality safety analysis of record for the WBN SFP is documented in [1], which was developed in 2001 and relied in part on analyses performed in 1996 [2]. The purpose of this report is to provide an up-to-date criticality safety evaluation for the WBN SFP based on the latest methodologies consistent with current NRC guidance [12, 13, 14]. This report replaces the analysis of record [1].

Criticality control in the BORAL™ storage racks DOES rely on the following:

- Fixed neutron absorbers: BORAL™ panels.
- Storage cell spacing, i.e., flux traps between storage cells.
- Soluble boron.

Criticality control in the BORAL™ storage racks DOES NOT rely on:

- Radial neutron leakage, i.e. arrangements are considered radially infinite.
- Non-integral absorbers in the fuel assembly.
- Integral fuel burnable absorbers (IFBA) in the fuel assembly.
- Burnup of fuel assemblies.

The criticality calculations qualify the BORAL™ storage racks uniformly loaded with fresh fuel assemblies (initial enrichment up to 5 wt.%).

2. METHODOLOGY

2.1 General Approach

The analysis is performed in a manner such that the results are below the regulatory limit with a 95% probability at a 95% confidence level. The calculations are performed using either the worst case bounding approach or the statistical analysis approach with respect to the various calculation parameters. The approach considered for each parameter is discussed below. These calculations are used to determine the final k_{eff} used to show compliance with the regulatory limits for both normal and accident conditions. The accident calculations are essentially modifications of the design basis cases for the normal conditions but do not introduce a new fuel or rack design change. Therefore, the uncertainty and bias calculations for the normal conditions are applicable and do not need to be repeated for the accident calculations.

2.2 Computer Codes and Cross Section Libraries

2.2.1 MCNP5-1.51

MCNP5-1.51 [3] is used for the criticality analyses. MCNP5-1.51 is a three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. MCNP was selected because it has a long history of successful use in fuel storage criticality analyses and has all the necessary features for the analysis to be performed for WBN SFP. MCNP5-1.51 calculations use continuous energy cross-section data predominantly based on ENDF/B-VII [4]. The default ENDF/B-VII cross sections are adjusted for temperature dependence using the appropriate continuous energy cross-section data processed with NJOY 99.396 code using ENDF/B-VII library [4, 5, 6].

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. All MCNP5-1.51 calculations are performed with a minimum of 12,000 histories per cycle, a minimum of 400 skipped cycles before averaging, and a minimum of 800 cycles that are accumulated. The initial source is specified as the fueled regions (assemblies) and confirmed to converge. It is a well-known fact [7] that k_{eff} (eigenvalue), which is an integral quantity, converges much faster than the fission source spatial distribution (eigenfunction). However, a convergence of the spatial source distribution is important for estimating local quantities, such as pin power. To assist users in assessing the convergence of the fission source spatial distribution, MCNP5 computes a quantity called the Shannon entropy of the fission source distribution, H_{src} [3]. The Shannon entropy [7] is a well-known concept from information theory that has been shown to be an effective diagnostic measure for characterizing convergence and provides a single number for each cycle to help characterize convergence of the fission source distribution. It has been found that the Shannon entropy converges to a single steady-state value as the source distribution approaches stationarity. Therefore, the convergence of the power iteration process is ensured using the Shannon entropy, as implemented in MCNP5 [3]. Since the eigenvalue (k_{eff}) converges faster than the fission source distribution, the convergence of the k_{eff} is assured by the convergence of the source distribution. The Shannon entropy convergence has been checked for each calculation.

2.2.1.1 MCNP5-1.51 Validation

Benchmarking of MCNP5-1.51 for criticality calculations is documented in [8]. The benchmarking is based on the guidance in [9, 12], and includes calculations for a total of [redacted] critical experiments with fresh UO_2 fuel, fresh MOX fuel, and fuel with simulated actinide composition of spent fuel (HTC experiments [8]). The benchmarking area of applicability is presented in Table 2.1. The results of the benchmarking calculations for the full set of all [redacted] experiments are presented in Table 2.2 along with trending analysis. Following the guidance in [12], the statistical treatment used to determine those values considered the variance of the population about the mean and used appropriate confidence factors and trend analysis. This is also consistent with the requirement in [14].

Trend analyses are also performed in [8], and the significant trends determined for various subsets and parameters are presented in Table 2.2. In order to determine the maximum bias and bias uncertainty that is applicable to the calculations in this report, the trend equations from [8] are

evaluated for the specific parameters of the current analyses in Table 2.3. The results presented in Tables 2.2 and 2.3 show the maximum bias and bias uncertainty associated with the benchmark subsets. The maximum bias and bias uncertainty are applied to all analysis calculations to determine the maximum k_{eff}.

2.3 Analysis Methods

The calculations are performed using either the worst case bounding approach or the statistical analysis approach with respect to the various calculation parameters. These bounding inputs and assumptions for the fuel and storage rack models are summarized below:

Bounding Fuel Designs and Fuel Assembly Parameters:

- Each design basis analysis calculation considers fresh fuel with a uniform enrichment. The same bounding enrichment is considered along the entire active length for each fuel pin. Lower enriched blankets are neglected. Therefore, there is no axial or radial variation in fuel along the entire active length. See Section 2.3.1.
- A bounding fuel density of all types of fuel assemblies is considered. This bounding approach provides analysis simplicity and margin. See Section 2.3.2.

Bounding Storage Rack Parameters:

- The bounding BORAL™ panel parameters are considered. This bounding approach provides analysis simplicity and margin since the reactivity effects of BORAL™ panel parameters are treated as a bias, not an uncertainty. See Section 2.3.3.
- Since it is not known if actual BORAL™ blistering has occurred, water with thickness of 0.09" (the same thickness of BORAL™ panel) in the gap between the BORAL™ and the steel sheathing is conservatively replaced by void over the entire length and width of all panels in the pool to account the reactivity effect of BORAL™ blistering. This bounding approach provides analysis simplicity and margin since the BORAL™ blisters would only be expected locally. See Section 2.3.6.

Bounding SFP Moderator Temperature:

- The bounding SFP moderator temperature and density are used for all design basis calculations. The calculations include NJOY corrected cross sections and $S(\alpha,\beta)$ cards. See Section 2.3.4.

Various fuel and rack models are used in the analysis. In addition to the conservative inputs and assumptions discussed above, the base model used for the analysis calculations in Section 2.3 considers the following MCNP5-1.51 rack model (with additional variations for the various evaluations described in each subsection):

- 2x2 array of storage cells with the BORAL™ panels.
- All assemblies are centered in their fuel storage cells.

- Periodic boundary conditions are considered along the water gap, thus creating a laterally infinite array.
- The storage rack design parameters are nominal values, except for the BORAL™ panel parameters.
- All materials in the model have the same temperature as the SFP moderator.
- The fuel design parameters are nominal values, except for the fuel density.
- Minor structural materials were neglected; i.e., spacer grids were assumed to be replaced by water.

The MCNP5 BORAL™ design basis model is shown in Figure 2.1. Additional details and analysis methodology discussions are provided in each section below.

2.3.1 Design Basis Fuel Design

The WBN SFP contains various Westinghouse 17x17 PWR fuel designs (i.e. V5H, V+/P+ and RFA-2), which are selected for potential storage in the BORAL™ racks.

In this analysis, the V+/P+ fuel design is the design basis assembly. The reactivity of the V+/P+ fuel design is more than or statistically equal to the V5H and RFA-2 fuel designs and is therefore used to bound those lower reactivity fuel designs and qualify all the Westinghouse 17x17 PWR fuel designs for storage in the BORAL™ racks.

The fuel design evaluation calculates the maximum reactivity of each fuel design using the 2x2 rack model discussed in Section 2.3. The bounding fuel density of all fuel assemblies is used for each fuel design. Each fuel design calculation considers fresh fuel with a uniform enrichment. The same bounding enrichment is considered along the entire active length for each fuel pin. Lower enriched blankets are neglected. Therefore, there is no axial or radial variation in fuel along the entire active length. This bounding approach provides analysis simplicity and additional margin.

The following cases are evaluated:

- Case 2.3.1.1: V5H.
- Case 2.3.1.2: V+/P+.
- Case 2.3.1.3: RFA-2.

2.3.2 Reactivity Effect of Fuel Design Parameters

The fuel design parameters are determined using the same 2x2 rack model as discussed in Section 2.3 with the exception for variations in fuel design parameters. For the fuel assembly parameter evaluations, each fuel assembly in the model has the tolerance applied at the same time.

Note that 97% of UO_2 theoretical density is used in the calculations, which bounds the maximum value of fuel density of all types of fuel design (See Table 5.1), thus the tolerance of fuel density is not considered here. In addition, the fuel enrichment tolerance cases are not considered since the value of maximum enrichment plus its tolerance (4.95 + 0.05 wt.% U-235) is used in the

calculations. Therefore, the following PWR fuel tolerances are considered for the bounding fuel (the V+/P+) design for nominal conditions:

- Case 2.3.2.1: Reference case. All tolerance parameters are nominal except that the fuel density is the maximum.
- Case 2.3.2.2: Minimum cladding thickness.
- Case 2.3.2.3: Maximum cladding thickness.
- Case 2.3.2.4: Minimum fuel rod pitch.
- Case 2.3.2.5: Maximum fuel rod pitch.
- Case 2.3.2.6: Minimum fuel pellet OD
- Case 2.3.2.7: Maximum fuel pellet OD
- Case 2.3.2.8: Minimum guide tube/instrument tube thickness.
- Case 2.3.2.9: Maximum guide tube/instrument tube thickness.

The reactivity effect of each tolerance is determined from:

delta-$k_{calc} = (k_{calc2} - k_{calc1}) \pm 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$

where $\pm 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$ is called the 95/95 uncertainty.

The maximum positive (if any) reactivity effect of the MCNP5-1.51 calculations for each tolerance is then selected and this maximum value is statistically combined with the other maximum values from every tolerance calculation to determine the fuel design manufacturing tolerance value used to determine the maximum k_{eff}.

2.3.3 Reactivity Effect of BORAL™ Rack Parameters

The tolerances of BORAL™ rack parameters are determined using the same 2x2 rack model as discussed in Section 2.3 with V+/P+ fuel with nominal fuel design parameters and with the exception for variations in BORAL™ rack parameters.

The cases of BORAL™ B-10 areal density, BORAL™ panel width and BORAL™ panel thickness increases/decreases are not considered since it is a well-known effect that minimum parameter values will increase reactivity. Instead, the minimum BORAL™ B-10 areal density, BORAL™ panel width and BORAL™ panel thickness are used in all the calculations.

The following tolerances are considered:

- Case 2.3.3.1: Reference case. All storage rack parameters nominal except that the minimum BORAL™ B-10 areal density, BORAL™ panel width and BORAL™ panel thickness are used.
- Case 2.3.3.2: Minimum storage cell inner diameter.
- Case 2.3.3.3: Maximum storage cell inner diameter.
- Case 2.3.3.4: Minimum storage cell wall thickness.

- Case 2.3.3.5: Maximum storage cell wall thickness.
- Case 2.3.3.6: Minimum storage cell sheathing thickness.
- Case 2.3.3.7: Maximum storage cell sheathing thickness.
- Case 2.3.3.8: Minimum storage cell poison gap thickness.
- Case 2.3.3.9: Maximum storage cell poison gap thickness.
- Case 2.3.3.10: Minimum storage cell flux trap. Note that the flux trap is a fixed value except for Cases 2.3.3.10 and 2.3.3.11. i.e., when the tolerance in each of the other case is applied.
- Case 2.3.3.11: Maximum storage cell flux trap.
- Case 2.3.3.12: Poison coupon measurement uncertainty. To account for the measurement uncertainty associated with the B-10 content in the BORAL™, an uncertainty of 5% (a typical value) on the minimum areal density of the B-10 content in the BORAL™ is included in the uncertainty analysis.

The reactivity effect of each tolerance is determined from:

delta-$k_{calc} = (k_{calc2} - k_{calc1}) \pm 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$

where $\pm 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$ is called the 95/95 uncertainty.

The maximum positive (if any) reactivity effect of the MCNP5-1.51 calculations for each tolerance is then selected and this maximum value is statistically combined with the other maximum values from every tolerance calculation to determine the BORAL™ rack manufacturing tolerance value used to determine the maximum k_{eff}.

2.3.4 Reactivity Effect of Spent Fuel Pool Water Temperature

The criticality analysis should be performed at the most reactive temperature and density [12]. Additionally, there may be temperature-dependent cross section effects in MCNP5-1.51 that need to be considered. In general, both density and cross section effects are not necessarily the same for all storage rack scenarios, since configurations with strong neutron absorbers typically show a higher reactivity at lower water temperature, while configurations without such neutron absorbers typically show a higher reactivity at a higher water temperature. For the WBN SFP BORAL™ storage racks, the maximum reactivity condition therefore is the minimum SFP water temperature and maximum density.

Additionally, the standard cross section temperature in MCNP5-1.51 is 300 K. Cross sections are also available at other temperatures, however not usually at the desired temperature for SFP criticality analysis. MCNP5-1.51 has the ability to automatically adjust the cross sections to the specified temperature when using the TMP card. Additionally, MCNP5-1.51 has the ability to make a molecular energy adjustment for select materials (such as water) by using the S (α, β) card. The S (α, β) card is provided for certain fixed temperatures which are not always applicable to SFP criticality analysis. Rather, there are limited temperature options, i.e. 300 K and 350 K, etc. Additionally, MCNP5-1.51 does not have the ability to adjust the S (α, β) card for temperatures as it does for the TMP card discussed above. Therefore, the cross sections and S (α, β) card are adjusted using NJOY [5, 6], and these adjusted cross sections use a temperature that is reasonably

close to the SFP specific values. This approach is acceptable because the system is poisoned and the reactivity trend is well known.

Studies are performed to demonstrate the reactivity effect of the moderator temperature and density over the range 4 °C through 120 ℃ [1] using temperature adjusted cross sections and S (α, β) cards. The bounding temperature is determined using the same 2x2 rack model as discussed in Section 2.3 with V+/P+ fuel (V+/P+ with fuel enrichment at 5 wt.% U-235 and nominal fuel design parameters). The following studies are performed:

- Case 2.3.4.1: Reference case. Temperature of 4 ℃ (NJOY cross sections are 4 ℃). Same model as discussed in Section 2.3.1.
- Case 2.3.4.2. Minimum nominal temperature 20 °C (NJOY cross sections are 21 °C).
- Case 2.3.4.3. Maximum possible temperature 120 °C (NJOY cross sections are 124 °C).
- Case 2.3.4.4. Maximum possible temperature 120 °C with 10% void, i.e., a reduction in the water density by 10% (boiling conditions, NJOY cross sections are 124 ℃).

All design basis calculations consider the bounding moderator temperature and density using temperature adjusted cross sections and S (α, β) cards.

2.3.5 Reactivity Effect of Fuel Radial Positioning

The fuel assembly can be in the following main radial locations: cell centered, eccentrically positioned towards the center of the racks, and moved towards the same corner of the cells. Significant reactivity effect from the radial position of the fuel assembly is not expected [12]. The radial location for fuel is evaluated with V+/P+ fuel (at 5 wt.% U-235 and nominal fuel design parameters) using the same 2x2 rack model as discussed in Section 2.3 which evaluate more local effects, as well as larger arrays that represent an entire rack, and therefore captures global positioning effects. Exceptions for the variations are described below:

- Case 2.3.5.1: Reference case. All assemblies are centered in their fuel storage cell of the 2x2 rack model;
- Case 2.3.5.2: All assemblies in the rack are moved as closely to the center of the 2x2 rack model as permitted by the rack geometry; this creates a laterally infinite arrangement of 2x2 arrays where the assemblies are close together in each 2x2 array. Note that a configuration with assemblies moved away from the center in each 2x2 array would be equivalent due to the boundary condition and is therefore not separately considered;
- Case 2.3.5.3: All assemblies in the 2x2 arrays of rack are moved towards the same corner of the cell. This creates a laterally infinite configuration with assemblies moved towards the same corner of each cell.
- Case 2.3.5.4: 8x8 array where all assemblies are moved closest to the center of that array. This essentially represents entire racks and therefore captures global positioning effects, where the 2x2 arrays evaluate more local effects. A periodic boundary condition is also used on the periphery of the model. This neglects the gap between adjacent racks, and is therefore conservative and simplifies model generation.

The reactivity effect of each tolerance is determined from:

delta-$k_{calc} = (k_{calc2} - k_{calc1}) \pm 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$

where $\pm 2 * \sqrt{(\sigma_1^2 + \sigma_2^2)}$ is called the 95/95 uncertainty.

The result of the MCNP5-1.51 calculation for the fuel eccentric positioning is considered as bias and bias uncertainty, rather than the uncertainty only, to determine the maximum k_{eff}. Therefore, the delta-k is treated as bias and the 95/95 uncertainty of the bias is statistically combined with the other uncertainty.

2.3.6 Reactivity Effect of BORAL™ Blistering

The effect of BORAL™ blistering on the reactivity of the spent fuel racks installed at the WBN is also evaluated. It is to be expected that formation of BORAL™ blisters wouldn't result in loss of neutron absorber. However, the blisters may displace the water considered in the gap. Therefore, water in the gap between the BORAL™ and the steel is replaced by void. While blisters would only be expected locally, they are modeled over the entire length and width of all panels in the pool. This is conservative, since the total amount of the void space is much larger than that expected from actual blisters. The details of MCNP5 model for BORAL™ blistering in the design basis case is shown in Figure 2.2.

The following calculations are considered:

- Case 2.3.6.1: reference case, with void with thickness of 0.09", i.e. the same value as the minimum BORAL™ panel thickness, in the gap between the BORAL™ and the steel in the storage racks.
- Case 2.3.6.2: the thickness of void in the gap is 0.08".
- Case 2.3.6.3: the thickness of void in the gap is 0.07".
- Case 2.3.6.4: the thickness of void in the gap is 0.06".
- Case 2.3.6.5: the thickness of void in the gap is 0.05".
- Case 2.3.6.6: the thickness of void in the gap is 0.04".
- Case 2.3.6.7: the thickness of void in the gap is 0.03".
- Case 2.3.6.8: the thickness of void in the gap is 0.02".
- Case 2.3.6.9: the thickness of void in the gap is 0.01".
- Case 2.3.6.10: the thickness of void in the gap is 0", i.e., the void in the gap is completely replaced by water.

The maximum reactivity case is used for all design basis calculations.

2.3.7 Model Simplifications

While the fuel and rack models used in the analyses are very detailed, to assure the true reactivity will always be less than the calculated reactivity, a number of conservative design criteria and assumptions were still necessary to be employed. The following is a list of all those simplifications,

together with a brief discussion of the possible effect, and an identification of those simplifications where additional studies may be necessary (and performed in Section 7) to show that they are acceptable.

- Moderator is borated or unborated water at a temperature in the operating range that results in the highest reactivity, as determined by the results discussed in Section 2.3.4.

- Dishing and chamfering of the fuel pellets is neglected, i.e. the fuel is always modeled as solid cylinder inside the cladding. This is acceptable since the amount of fuel is maintained, and the water-to-fuel ratio and the principal location of the fuel remain unchanged.

- 97% (10.631 g/cm^3) of UO_2 theoretical density is used in the analysis, which bounds the maximum as-built UO_2 density of all fuel assemblies in WBN SFP, as discussed in Section 2.3.

- Minor parts of the fuel and rack construction are neglected and replaced by water. Those include grid straps and minor structural rack components.

- All fuel and rack structures above and below the active region of the fuel are neglected and replaced by unborated water. While it may neglect some reflection from steel structures in those areas, it also neglects the absorption in those steel structures, and maximizes the axial water reflection. In addition, the maximum reactivity is expected at the center for the fresh fuel. The assumption is therefore considered appropriate and conservative.

- All fuel cladding material is modeled as pure zirconium, while the actual fuel cladding consists of one of several zirconium alloys. This is acceptable since the model neglects the trace elements in the alloy which may provide additional neutron absorption.

- The neutron absorber (BORAL™) length in the racks is larger than the active length of the fuel. In the calculations, it is assumed that the neutron absorber length is as the same as the active fuel length.

- Each calculation considers fresh fuel with a uniform enrichment. Lower enriched blankets are neglected. Therefore, there is no axial or radial variation in fuel along the entire active length. This bounding approach provides analysis simplicity and additional margin.

- The value of maximum enrichment plus its tolerance (4.95 + 0.05 wt.% U-235) is considered along the entire active length for each fuel pin, which bounds the enrichment of all fuel assemblies in WBN SFP, as discussed in Section 2.3.

- While actual fresh fuel assembly may contain some number of IFBA rods and other fuel will have some burnup, no attempt is made here to quantify the negative reactivity associated with the effects of IFBA rods or fuel burnup.

2.3.8 Calculations to Determine Maximum k_{eff} Values

2.3.8.1 Calculation of a Maximum k_{eff} Value

Applying all the considerations from the previous sections, the calculation of a single k_{eff} value for the design basis calculations consists of the following steps:

- An infinite array is used in the MCNP design basis model, which precludes leakage.
- The calculated k (k_{calc}) is determined including the following conservative values within the MCNP design basis model:
    - Bounding fuel density, as discussed in Section 2.3.2.
    - Bounding fuel enrichment, as discussed in Section 2.3.2.
    - Bounding BORAL™ panel thickness, BORAL™ panel width and BORAL™ B-10 areal density, as discussed in Section 2.3.3.
    - Water temperature and density identified which results in the maximum reactivity are used which covers all normal and accident conditions, as discussed in Section 2.3.4.
    - BORAL™ blistering, as discussed in Section 2.3.6.
- The maximum k_{eff} value is then calculated by adding all biases, and a statistical combination of all remaining uncertainties using the following equation:

  $$k_{eff} = k_{calc} + \text{uncertainty} + \text{bias}$$

  where uncertainty includes:
    - MCNP5-1.51 bias uncertainty from criticality benchmarking validation (95% probability at a 95% confidence level), as discussed in Section 2.2.1.1.
    - MCNP5-1.51 calculations statistics (95% probability at a 95% confidence level, 2σ).
    - Fuel tolerance uncertainty. As discussed in Section 2.3.2, to determine this, MCNP studies are performed to determine the reactivity effects of the fuel manufacturing tolerance.
    - Rack tolerance uncertainty. As discussed in Section 2.3.3, to determine this, MCNP studies are performed to determine the reactivity effects of the rack manufacturing tolerance.
    - Fuel eccentric positioning bias uncertainty. As discussed in Section 2.3.5, to determine this, MCNP studies are performed to determine the reactivity effects of the eccentric positioning of fuel assemblies.

  and the biases include:
    - MCNP5-1.51 bias from criticality benchmarking validation, as discussed in Section 2.2.1.1.
    - Fuel eccentric positioning bias. As discussed in Section 2.3.5, to determine this, MCNP studies are performed to determine the reactivity effects of the eccentric positioning of fuel assemblies.

Note that each uncertainty is statistically combined with other uncertainties, while biases are added together in order to determine k_{eff}. The summation of the statistical summation of the uncertainties

and the summation of the biases, is also defined as The Total Correction Factor (TCF). The TCF is added to k_{calc} to determine the maximum k_{eff} to show compliance with the regulatory limit.

2.3.8.2 Maximum k_{eff} Calculation for Borated Water

The approach used here takes credit for soluble boron under normal conditions (see Section 3). The soluble boron concentration limit in the WBN SFP is presented in the boron dilution analysis [16]. To ensure that the effective neutron multiplication factor (k_{eff}) is less than the regulatory limit with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity, the calculation with soluble boron content of 500 ppm for the maximum enrichment of 5.0 wt.% is performed. Soluble boron content of 500 ppm is bounded by the soluble boron concentration limit in [16].

2.4 Storage Rack Interfaces

The storage cell racks are separated by at least 1 inch gap [10] that is determined by the rack baseplate extensions and bumper bars at the top. This gap represents the minimum possible separation between racks. Since the design basis model is an infinite reflection of the 2x2 array with uniform loading of fresh fuel described above, the design basis model is bounding with respect to any rack interface configurations and therefore no further rack interface calculations are necessary.

2.5 Other Normal Conditions

2.5.1 Fuel Movement Operations under Normal Conditions

The movement of fuel in the SFP is a normal operation. The following case is analyzed:

- Case 2.5.1.1: a single fresh fuel assembly in water. No storage rack is modelled. Unborated Water
- Case 2.5.1.2: a single fresh fuel assembly in water. No storage rack is modelled. 500 ppm borated Water.

This condition bounds all situations during normal fuel movement in the pool, since only a single assembly can be moved at a time. As discussed previously in Section 2.3.1, the V+/P+ fuel design is bounding thus it is considered in the calculation.

Fuel activities related to the PWR fuel are also addressed considering the accident conditions (see Section 2.6).

2.5.2 Replacing Fuel with Empty Cell or Non-Fuel Hardware

Insertion of nonfuel hardware in the cells allowed to contain fresh fuel is permitted. This is because each rack storage cell is surrounded by the cell walls containing neutron absorber material (BORAL™), where one or more cell locations are not occupied with fuel, the amount of fissile material is obviously reduced. Also, if the cells are filled with all water with full density, the increased amount of water will result in an increased moderation of neutrons in the empty cell locations. This increased moderation will increase the effectiveness of the surrounding thermal neutron absorber. However, even if all the water in the cell is replaced by non-fissile materials, though the reactivity of the rack may increase because of decreased moderation (although this is not physically realistic since plenty of full density water will actually remain along the edge of the cell near the poison), it is still well bounded by the design basis case with fuel in the cell since the effect of removal of fissile material (fuel bundle) is still dominant. Therefore, there is no limit on the amount of non-fuel hardware that can be placed in the cell. No further evaluations of this condition are necessary.

Also, storage of inserts or control rods in these cells is considered acceptable, since those devices only replace a very small amount of water, while at the same time adding a substantial amount of neutron absorber.

2.6 Accident Conditions

There are various potential accident conditions that must be evaluated. The credible accidents to be evaluated are:

- The effect of SFP temperature exceeding the normal range.
- A dropped fuel assembly.
- A misloaded fuel assembly (a fuel assembly in the wrong location within the storage rack).
- A mislocated fuel assembly (a fuel assembly in the wrong location outside the storage rack).
- Rack movement due to seismic activity.

For each accident condition considered, a calculation is performed without soluble boron and an additional calculation is performed with a soluble boron concentration of 500 ppm.

The calculation of the maximum k_{eff} for accident conditions for the BORAL™ storage racks in includes the same conservative biases as those within the design basis model:

- Bounding fuel density.
- Bounding fuel enrichment.
- Bounding BORAL™ panel parameters.
- Water density and temperature identified to result in higher reactivities at normal conditions,
- BORAL™ blistering.

The maximum k_{eff} value is then calculated by adding the total correction factor (TCF), i.e. the statistical summation of the uncertainties and the summation of the biases, to k_{calc} to show compliance with the regulatory limit.

Note that each uncertainty is statistically combined with other uncertainties, while biases are added together in order to determine k_{eff}. For the cases with credit for soluble boron, where the regulatory limit is 0.95, no specific target k_{eff} is defined, rather a soluble boron concentration is selected and the k_{eff} is shown to meet the regulatory limit.

2.6.1 Temperature and Water Density Effects

When the SFP water temperature exceeds the nominal SFP temperature range, it is considered as the SFP water temperature accident condition. The reactivity effect of the SFP temperature over the range 4 °C through 120 °C is evaluated in Section 2.3.4. All design basis calculations consider the bounding SFP water temperature and density. Therefore, no further calculations are necessary.

2.6.2 Dropped Assembly – Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of a rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more than 12 inches [10], which is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). Consequently, the horizontal fuel assembly drop accident will not result in an increase in reactivity.

2.6.3 Dropped Assembly – Vertical into a Storage Cell

It is also possible to vertically drop an assembly into a location that might be occupied by another assembly. Such a vertical impact would at most cause a small compression of the stored assembly, if present. However, the design basis model is an infinite reflection of the 2x2 array with uniform loading of fresh fuel at the maximum enrichment. The vertical drop is therefore bounded by the design basis calculation and no separate calculation is performed.

2.6.4 Misloaded Fresh Fuel Assembly

It is possible that a fresh fuel assembly could be accidently placed in the BORAL™ racks in the SFP. However, the design basis model is an infinite reflection of the 2x2 array with uniform loading of fresh fuel at the maximum enrichment of 5.0 wt.%, therefore, no further calculations are necessary.

2.6.5 Mislocated Fresh Fuel Assembly

The possibility exists that a fuel assembly could be accidently mislocated outside the periphery of the BORAL™ racks. The possible locations for a mislocated fuel would be in location where the mislocated fuel assembly would be face adjacent to a fresh fuel assembly in the storage rack. Note that all storage racks have boundary poison requirements [10]. This location bounds all other possible configurations. The mislocated fuel assembly accident evaluation considers an 8x8 rack model with a mislocated fresh 5.0 wt.% U-235 fuel assembly face adjacent to the rack at its closest proximity, as

shown in Figure 2.3. Reflective boundary conditions are considered along the centerline of the rack to rack gaps, and 30 cm of water are considered beyond the mislocated fuel assembly. The following calculations are performed:

- Case 2.6.5.1: mislocated fresh 5.0 wt.% fuel assembly. The fuel in the storage racks is centered in the cell. The soluble boron concentration is 0 ppm.
- Case 2.6.5.2: mislocated fresh 5.0 wt.% fuel assembly. The fuel in the storage racks is eccentrically positioned as close to the mislocated fuel assembly as possible. The soluble boron concentration is 0 ppm.
- Case 2.6.5.3: mislocated fresh 5.0 wt.% fuel assembly. The fuel in the storage racks is centered in the cell. The soluble boron concentration is 500 ppm.
- Case 2.6.5.4: mislocated fresh 5.0 wt.% fuel assembly. The fuel in the storage racks is eccentrically positioned as close to the mislocated fuel assembly as possible. The soluble boron concentration is 500 ppm.

The TCF from the normal condition results is applied to determine the maximum k_{eff}. The soluble boron requirement is interpolated between the two calculations.

2.6.6 Rack Movement

In the event of seismic activity, there is a hypothetical possibility that the storage rack arrays may move and come closer to each other. In the worst case scenario, two racks may touch each other at the baseplate or bumper bar at the top and reducing the physical separation of the fuel assemblies along the rack interface. In the SFP, the worst case rack movement scenario is for the racks to all move together, and the gap between racks to also close. This accident condition is bounded by the evaluations of the design basis cases discussed in Section 2.3 because the design basis cases consider all the racks at their closest approach (a laterally infinite arrangement of 2x2 arrays where the assemblies are close together in each 2x2 array so that the gap between the racks is neglected). Therefore, no further evaluations are needed.

2.7 Margin Evaluation

The criticality analysis methodology conservatively includes biases in the design basis model for simplicity and margin. The following calculations are performed to estimate the overall margins of the analyses:

- Case 2.7.1: reference case, as the same as the design basis case.
- Case 2.7.2: same as the reference case, but fuel density is changed to be the actual maximum percent of UO_2 theoretical density, [redacted] (See Table 5.1)
- Case 2.7.3: same as the reference case, but fuel enrichment is changed to be the actual maximum enrichment, i.e., 4.95 wt.% U-235.
- Case 2.7.4: same as the reference case, but the void in the gap between the BORAL™ and the steel is replaced by water, i.e., there is no BORAL™ blistering.

Note that the analysis is performed with a uniform pattern of fresh fuel. While actual fresh fuel may contain some number of IFBA rods and other fuel will have some burnup, no attempt is made here to quantify the negative reactivity associated with those effects. Furthermore, only 500 ppm soluble boron is being credited.

3. ACCEPTANCE CRITERIA

Codes, standard, and regulations or pertinent sections thereof that are applicable to these analyses include the following:

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling." (10CFR 50.62).
- Code of Federal Regulations, Title 10, Part 50, Section 68, "Criticality Accident Requirements" (10CFR 50.68).
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.1, Criticality Safety of Fresh and Spent Fuel Storage and Handling, Rev. 3 – March 2007.
- L. Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.
- ANSI ANS-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- USNRC, NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology, January 2001.
- DSS-ISG-2010-01, Revision 0, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools.
- "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water-Reactor Power Plants", NEI 12-16, latest revision, Nuclear Energy Institute.

The objective of this analysis is to ensure that the effective neutron multiplication factor (k_{eff}) of the SFP loaded with fuel of the highest anticipated reactivity, at a temperature corresponding to the highest reactivity, is less than 1.0 for the pool flooded with un-borated water, and does not exceed 0.95 for the pool flooded with borated water, all for 95% probability at a 95% confidence level.

4. ASSUMPTIONS

A number of assumptions, either for conservatism or to simplify the calculation approach are applied in the analyses. Each assumption is appropriately discussed and justified in the text. Important aspects of applying those assumptions are as follows:

- Bounding or sufficiently conservative inputs and assumptions are used essentially throughout the entire analyses, and in most cases, studies are presented to show that the selected inputs and parameters are in fact conservative or bounding. See Section 2.3.7.
- An evaluation is performed to estimate the overall margins of the analyses. See Section 2.7.

5. INPUT DATA

5.1 Fuel Assembly Designs

The spent fuel storage racks are designed to accommodate three types of Westinghouse PWR 17x17 fuel assemblies: V5H, V+/P+ and RFA-2. V5H fuel has original standard fuel rods in a VANTAGE 5 cage, its mechanical design includes Zirc-4 grids, fuel cladding, guide thimbles and instrument tubes. V+/P+ combines features of the V+ and the P+ designs with mechanical design features of ZIRLO fuel rod cladding, guide thimbles and instrument tubes, a protective bottom grid, and ZIRLO mid grids. RFA-2 fuel is the same as V+/P+ but with thicker ZIRLO guide thimbles and instrument tube, mid-grid design changes and the addition of ZIRLO Intermediate Flow Mixer grids located in the three uppermost spans between the mixing vane grids. The PWR fuel assembly data used in the analysis is presented in Table 5.1.

For all fuel designs, the fuel assembly is explicitly modeled in terms of fuel pin, cladding, instrument tube and guide tubes. The instrument tube and guide tubes are all considered to be the same length as the fuel.

5.2 Storage Rack Designs

The storage cells are composed of stainless steel boxes separated by a gap with BORAL™ neutron absorber panels (attached by stainless steel sheathing), centered on each side of the storage cells. The steel walls define the storage cells and the stainless steel sheathing supports the BORAL™ panel and defines the boundary of the flux-trap water-gap. Stainless steel channels connect the storage cells in a rigid structure and define the flux-trap between the sheathing of the neutron absorber panels. Figure 5.1 provides a sketch of the BORAL™ storage racks along with the critical dimensions.

The design basis calculational models consist of 2x2 configuration of storage cells with periodic boundary conditions through the centerline of the water gap on the outer boundary of the cells, thus simulating an infinite array of storage cells. The storage rack designs data used in the analysis is presented in Table 5.2.

5.3 Material Composition

The material compositions for the various MCNP models are presented in Table 5.3.

6. COMPUTER CODES

The following computer code was used in this analysis.

- MCNP5-1.51 [3] is a three-dimensional continuous energy Monte Carlo code developed at Los Alamos National Laboratory. This code offers the capability of performing full three-dimensional calculations for the loaded storage racks. MCNP5-1.51 was run on the workstations at Holtec.

7. ANALYSIS

As discussed in Section 2.0, the analysis is performed using a combination of bounding analysis parameters and statistical uncertainties. The analysis results and discussions are provided in each section below.

7.1 Bounding Fuel Design

As discussed in Section 2.3.1, the following PWR fuel designs are permitted for storage in the WBN SFP BORAL™ storage racks: V5H, V+/P+ and RFA-2. Evaluations are performed for the three fuel designs and the results are presented in Table A.1. The reactivities of the V5H and V+/P+ are statistically equivalent and higher than the reactivity of the RFA-2. For simplicity and analysis margin, the V+/P+ is selected as the bounding fuel design with a conservative enrichment of 5.0 wt.% U-235. Thus, further evaluations of the other fuel designs are not required.

7.2 Reactivity Effect of Fuel Design Parameters

As discussed in Section 2.3.2, evaluations are performed for the bounding fuel design, V+/P+ to determine the reactivity effect of the fuel assembly manufacturing tolerances. The results of the evaluations are presented in Table A.2. The reactivity effect of the fuel assembly parameters is treated as an analysis uncertainty.

7.3 Reactivity Effect of BORAL™ Storage Rack Parameters

As discussed in Section 2.3.3, the BORAL™ storage rack parameters were evaluated to determine the reactivity effect of the storage rack manufacturing tolerances. The results of the evaluations are presented in Table A.3. The reactivity effect of the storage rack parameters is treated as an analysis uncertainty.

7.4 Reactivity Effect of SFP Water Temperature

As discussed in Section 2.3.4, the reactivity effect of SFP water temperature and density is evaluated so that the bounding values can be used in the design basis mode. The results of the evaluations are presented in Table A.4. The results show that the bounding temperature (and corresponding density) are the minimum temperature and maximum density. Therefore, these values are used in the design basis model.

7.5 Reactivity Effect of Radial Positioning of Fuel Assembly

As discussed in Section 2.3.5, the reactivity effect of the fuel radial location is evaluated. The results of the evaluations are presented in Table A.5. The reactivity effect of the radial position of the fuel is treated as a bias and bias uncertainty rather than an analysis uncertainty only.

7.6 Reactivity Effect of BORAL™ Blistering

As discussed in Section 2.3.6, the reactivity effect of the BORAL™ blistering is evaluated to determine the bounding configuration for use in the design basis model. The results of the evaluations are presented in Table A.6. The results show that the bounding configurations are the reference case with the 0.09” thickness void in the gap between the BORAL™ and the steel sheathing. Therefore, the bounding BORAL™ blistering configurations is included for all design basis calculations.

7.7 Calculations to Determine Maximum k_{eff} Values

As discussed in Section 2.3, various evaluations have been performed to determine the bounding set of parameters, biases and bias uncertainties for the design basis model. The results of these evaluations have been discussed in the previous sections. Based on the results of those evaluations, the design basis calculations for normal conditions have been performed and the maximum k_{eff} value is determined as discussed in Section 2.3.8.1. The results of the maximum k_{eff} value for normal conditions are presented in Table 7.1.

7.8 Storage Rack Interfaces

As discussed in Section 2.4, the WBN storage rack interfaces are bounded by the design basis model. No additional calculation is needed for the rack interface.

7.9 Fuel Movement at Normal Condition

As discussed in Section 2.5, a single fresh fuel assembly in water was analyzed. This bounds any conditions during movements of a single assembly in the pool. The result is presented in Table A.7 and show that the reactivity for this condition is well below the regulatory limit.

7.10 Accident Conditions

As discussed in Section 2.6, the following accident conditions have been evaluated:

- The effect of SFP temperature exceeding the normal range.
- A dropped fuel assembly.
- A misloaded fuel assembly (a fuel assembly in the wrong location within the storage rack).
- A mislocated fuel assembly (a fuel assembly in the wrong location outside the storage rack).
- Rack movement due to seismic activity.

Additional calculations are only needed for the mislocated accident condition. The results of the mislocated accident condition evaluations are presented in Table A.8. The results of the maximum k_{eff} value for accident conditions are presented in Table 7.2. Results show that a soluble boron concentration of 500 ppm is sufficient to ensure that the maximum k_{eff} is below regulatory limit.

7.11 Margin Evaluation

The margin evaluation that is used in this report is described in detail in Section 2.7. The results of the overall margins of the analyses are presented in Table 7.3.

8. CONCLUSION

The criticality calculations for the WBN SFP have been performed for the BORAL™ storage racks. The objective of the analysis is to demonstrate that the effective neutron multiplication factor (k_{eff}) is less than 1.0 with the pool flooded with un-borated water, and will not exceed the regulatory limit of 0.95 with credit for 500 ppm soluble boron [1]. The maximum k_{eff} includes a margin for uncertainty in reactivity calculations including manufacturing tolerances and is shown to be less than the regulatory limit with a 95% probability at a 95% confidence level, as presented in Table 7.1 for normal condition, and Table 7.2 for accident condition. The analyses also show that normal fuel movement in the SFP and replacing any cells that contain fuel assemblies with empty water cells or non-fuel hardware are acceptable.

Additionally, the criticality safety analysis provides additional margin to the regulatory limit, the results of margin evaluations are presented in Table 7.3.

[1] An additional 50 ppm of soluble boron has been added per Section 5.1.1 of [14].

9. REFERENCES

[1] HI-2012620, “Evaluation of the Effect of the Use of Tritium Producing Burnable Absorber Rods (TPBARS) on Fuel Storage Requirements”, Revision 2, Holtec International.

[2] HI-961513, “Evaluation of the Spent Fuel Storage Racks for the Watts Bar Nuclear Plant”, Revision 2, Holtec International.

[3] “MCNP - A General Monte Carlo N-Particle Transport Code, Version 5,” Los Alamos National Laboratory, LA-UR-03-1987 (2003, Revised 2/1/2008).

[4] ENDF/B-VII.0 Evaluated Nuclear Data Library, Released December 15, 2006.

[5] NJOY99.0, Code System for Producing Pointwise and Multigroup Neutron and Photon Cross Section from ENDF/B Data, PSR-480, March 2000, ORNL.

[6] HI-2156450, "MCNP5 Temperature Dependent Library Construction with NJOY", latest revision, Holtec International.

[7] F. Brown, A Review of Monte Carlo Criticality Calculations – Convergence, Bias, Statistics, 2009 International Conference on Mathematics, Computational Methods and Reactor Physics, Saratoga Springs, NY, 2009.

[8] "Nuclear Group Computer Code Benchmark Calculations", Holtec Report HI-2104790 Revision 3.

[9] “Guide for Validation of Nuclear Criticality Safety Calculational Methodology”, USNRC, NUREG/CR-6698, January 2001.

[10] Transmittal of TVA Watts Bar (WBN) Parameters for Spent Fuel Pool (SFP) Criticality Analysis, Dated December 19, 2016, from Z. I. Martin (TVA) to Jordan Landis (Holtec International).

[11] Email from Zita I. Martin (TVA) to Jordan Landis (Holtec International), dated May 3, 2017, subject “RE: Watts Bar Spent Fuel Pool Criticality Analysis: Draft Summary of Results per PO# 2538594”.

[12] DSG-ISG-2010-01, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools, Revision 0.

[13] L. Kopp, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.

[14] “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water-Reactor Power Plants”, NEI 12-16, latest revision, Nuclear Energy Institute.

[15] Email from Mike Pamula (Holtec International) to Tao He (Holtec International), dated May 3, 2017, subject "RE: Watts Bar Spent Fuel Pool Criticality Analysis: Draft Summary of Results per PO# 2538594".

[16] "Spent Fuel Pool Boron Concentration Analysis During DCS Activities at Watts Bar", Holtec Report HI-2156794, Revision 2.

Table 2.1
Summary of the Area of Applicability of the MCNP5-1.51 Benchmark

Parameter	Design Application	Benchmarks	Validated
Fissionable Material	[redacted]	[redacted]	[redacted]
Isotopic Composition			
$^{235}U/U$	[redacted]	[redacted]	[redacted]
Pu/(U+Pu)	[redacted]	[redacted]	[redacted]
Physical Form	[redacted]	[redacted]	[redacted]
Fuel Density (g/cm^3)	[redacted]	[redacted]	[redacted]
Moderator Material (coolant)	[redacted]	[redacted]	[redacted]
Physical Form	[redacted]	[redacted]	
Density (g/cm^3)	[redacted]	[redacted]	[redacted]
Reflector Material	[redacted]	[redacted]	[redacted]
Physical Form	[redacted]	[redacted]	[redacted]
Density (g/cm^3)	[redacted]	[redacted]	[redacted]
Interstitial Reflector Material			
Plate	[redacted]	[redacted]	[redacted]
Absorber Material			
Soluble	[redacted]	[redacted]	[redacted]
Rods	[redacted]	[redacted]	[redacted]
Separating Material			
Plate	[redacted]	[redacted]	[redacted]
Geometry			
Lattice type	[redacted]	[redacted]	[redacted]
Lattice Pitch (cm)	[redacted]	[redacted]	[redacted]
Neutron Energy	[redacted]	[redacted]	[redacted]

[redacted]

Table 2.2
MCNP5-1.51 Benchmarking Bias and Bias Uncertainty and Trending Analysis [8]

Experiment Description	No. of exp.	Bias [3]	Bias Uncertainty[4]	Normality χ^2 ($P_d(\chi^2;d)$)	Linear Correlation	Residuals Normality, ($P_d(\chi^2;d)$)
Fresh UO_2 Fuel with Fresh Water	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Fresh UO_2 Fuel with Borated Water [5]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[3] The same definition of the bias as in [8] is used in Tables 2.2 and 2.3, i.e. Bias = k_{eff} -1. A negative value therefore indicates that the corresponding positive value would need to be added to the calculated results before it is compared to the applicable limit.
[4] The distribution free bias uncertainty is provided in parentheses in Tables 2.2 and 2.3 for the subsets with the rejected normality assumption. See [8].
[redacted]

Table 2.3
Summary of MCNP5-1.51 Benchmarking Bias and Bias Uncertainty Significant Trending Analysis

Experiment Description	Linear Correlation	Analysis Parameter Value[6]	Analysis Parameter Trend Bias	Analysis Parameter Trend Bias Uncertainty	[8] Trend Table
Fresh UO_2 Fuel with Fresh Water	[redacted]	[redacted]	[redacted]	[redacted]	D.3-16

[6] The maximum or minimum parameter value that provides the maximum bias is used.

Table 5.1
Specification of the Fuel Assembly Parameters [10]

Fuel Assembly Design	V5H	V+/P+	RFA-2
Fuel Assembly Data			
Fuel Rod Array	17x17	17x17	17x17
Number of fuel rods	264	264	264
Fuel Assembly Overall Length, Inches	[redacted]	[redacted]	[redacted]
Distance from Bottom of Fuel Assembly to Beginning of Active Length, Inches	[redacted]	[redacted]	[redacted]
Active Fuel Length, Inches	[redacted]	[redacted]	[redacted]
Fuel Rod Pitch, Inches	[redacted]	[redacted]	[redacted]
Fuel Rod Data			
Clad Outer Diameter, Inches	[redacted]	[redacted]	[redacted]
Clad Inner Diameter, Inches	[redacted]	[redacted]	[redacted]
Clad Material	Zirc-4	ZIRLO	ZIRLO
Pellet Diameter, Inches	[redacted]	[redacted]	[redacted]
Max % Theoretical Density [7], %	[redacted]	[redacted]	[redacted]
Guide/Instrument Tube Data			
Number of Guide Tube	24	24	24
Guide Tube Outer Diameter, Inches	[redacted]	[redacted]	[redacted]
Guide Tube Inner Diameter, Inches	[redacted]	[redacted]	[redacted]
Guide Tube Material	Zirc-4	ZIRLO	ZIRLO
Number of Instrument Tube	1	1	1
Instrument Tube Outer Diameter, Inches	[redacted]	[redacted]	[redacted]
Instrument Tube Inner Diameter, Inches	[redacted]	[redacted]	[redacted]
Instrument Tube Material	Zirc-4	ZIRLO	ZIRLO

[7] 97% (10.631 g/cc) of UO_2 theoretical density is bounding and used in the analysis.

Table 5.2
Specification of the WBN BORAL™ Storage Racks [1, 2, 10, 11]

Parameter	Value
Rack Type	PWR BORAL™
Number of Racks	24
Storage Rack Material	304 SS
Rack Height (top of baseplate to top of rack), Inches	168.5
Distance from Rack Baseplate to Bottom of Neutron Absorber, Inches	2.75
Length of Neutron Absorber, Inches	147 ± 0.156
Neutron Absorber Thickness, Inches	0.100 (max), 0.090 (min[8])
Neutron Absorber Width, Inches	8.625 ± 0.120
Storage Cell Inner Diameter, Inches	8.75 ± 0.10
Storage Cell Pitch, Inches	10.375
Storage Cell Box Wall Thickness, Inches	0.09 ± 0.005
Storage Cell Neutron Absorber Pocket Thickness, Inches	0.200
Storage Cell Neutron Absorber Sheathing Thickness, Inches	0.036 ± 0.002
Storage Flux Trap, Inches	0.973 ± 0.107 [9]
Neutron Absorber Type	BORAL™
Neutron Absorber areal density (g/sqcm)[10]	0.0233 (min)
Neutron Absorber Density, g/cc	2.6
Rack to Rack Gap, Inches	1 (min)

[8] The minimum neutron absorber thickness is assumed and accepted by TVA per [15].
[9] Tolerance is assumed as the sum of tolerances of cell ID, cell wall thickness and cell sheathing thickness.
[10] For the BORAL™ racks, the minimum certified B-10 areal density is used per [11].

Table 5.3 (1 of 3)
Material Compositions

Element	MCNP ZAID[11] [3, 6]	Weight Fraction
Steel (density 7.92 g/cc)		
Cr	24050.21c	0.0079050
	24052.21c	0.1585266
	24053.21c	0.0183218
	24054.21c	0.0046467
Mn	25055.21c	0.0200100
Fe	26054.21c	0.0389826
	26056.21c	0.6345800
	26057.21c	0.0149174
	26058.21c	0.0020200
Ni	28058.21c	0.0671977
	28060.21c	0.0267760
	28061.21c	0.0011834
	28062.21c	0.0038348
	28064.21c	0.0010082
Zr (density = 6.55 g/cc)		
Zr	40090.21c	0.5070612
	40091.21c	0.1118009
	40092.21c	0.1727810
	40094.21c	0.1789110
	40096.21c	0.0294379

[11] The MCNP ZAID used is customized for ENDF/B-VII NJOY adjusted cross sections.

Table 5.3 (2 of 3)
Material Compositions

Element	MCNP ZAID[12] [3, 6]	Weight Fraction
BORAL (density = 2.6 g/cc, B-10 loading = 0.0233 g/sqcm)		
B	5010.21c	0.0385129
	5011.21c	0.1704448
C	6000.21c	0.0580379
Al	13027.21c	0.7330044
Pure water (density = 1.0 g/cc)		
H	1001.21c	0.1118854
	1002.21c	0.0000257
O	8016.21c	0.8857957
	8017.21c	0.0022932
500 ppm Borated Water (density = 1.0 g/cc)		
H	1001.21c	0.1118300
	1002.21c	0.0000257
O	8016.21c	0.8853540
	8017.21c	0.0022921
B	5010.21c	0.0000922
	5011.21c	0.0004078

[12]The MCNP ZAID used is customized for ENDF/B-VII NJOY adjusted cross sections.

Table 5.3 (3 of 3)
Material Compositions

Element	MCNP ZAID[13] [3, 6]	Weight Fraction
Fresh 5 wt.% U-235 Fuel (density = 10.631 g/cc)[14]		
U	92235.21c	0.0440800
	92238.21c	0.8374200
O	8016.21c	0.1185000

[13] The MCNP ZAID used is customized for ENDF/B-VII NJOY adjusted cross sections.
[14] Other fresh fuel compositions may be used; the design basis case is provided as an example.

Table 7.1

Summary of the Analysis Results for Normal Condition [15]

Parameter	Value
Uncertainties	
Fuel Tolerance Uncertainty, from Table A.2	0.0053
Rack Tolerance Uncertainty, from Table A.3	0.0167
Eccentric Positioning Uncertainty, from Table A.5	0
MCNP5-1.51 Calculation Statistics (95%/95%, 2σ) [16]	0.0008
MCNP5-1.51 Code Bias Uncertainty, from Table 2.2	0.0078
Statistical Combination of Uncertainties	0.0192
Biases	
Fuel Eccentricity Bias, from Table A.5	0
MCNP5-1.51 Code Bias, from Table 2.2	0.0007
Sum of Biases	0.0007
Total Correction Factor	
Total Correction Factor	0.0199
Determination of k_{eff}, Unborated Water	
Calculated MCNP5-1.51 k_{calc}	0.9759
Maximum k_{eff}	**0.9958**
Regulatory Limit	1.0000
Margin to the Limit	0.0042
Determination of k_{eff}, 500 ppm Borated Water [17]	
Calculated MCNP5-1.51 k_{calc}	0.9237
Maximum k_{eff}	**0.9436**
Regulatory Limit	0.9500

[15] As discussed in Section 7.6, the bounding BORAL™ blistering configurations is included for all design basis calculations in Tables 7.1 and 7.2. This is conservative and provides analysis margin.

[16] The standard deviation (σ) of the MCNP calculations is about 0.0004.

[17] An additional 50 ppm of soluble boron has been added per Section 5.1.1 of [14].

Table 7.2

Summary of the Analysis Results for Accident Conditions

Parameter	Value
Total Correction Factor	
Total Correction Factor (see Table 7.1)	0.0199
Determination of k_{eff}, 500 ppm Borated Water [18]	
Calculated MCNP5-1.51 k_{calc}, see Table A.8	0.9229
Maximum k_{eff}	**0.9428**
Regulatory Limit	0.9500

[18] An additional 50 ppm of soluble boron has been added per Section 5.1.1 of [14].

Table 7.3

Summary of the Margin Evaluation

Case	Description	k_{calc}	σ	Delta k_{calc}
2.7.1	Reference case, Design Basis Case	0.9759	0.0004	n/a
2.7.2	Actual maximum fuel density	0.9750	0.0004	-0.0009
2.7.3	Actual maximum fuel enrichment	0.9744	0.0004	-0.0015
2.7.4	No BORAL™ blisters	0.9569	0.0004	-0.0190
Total Margin				-0.0214

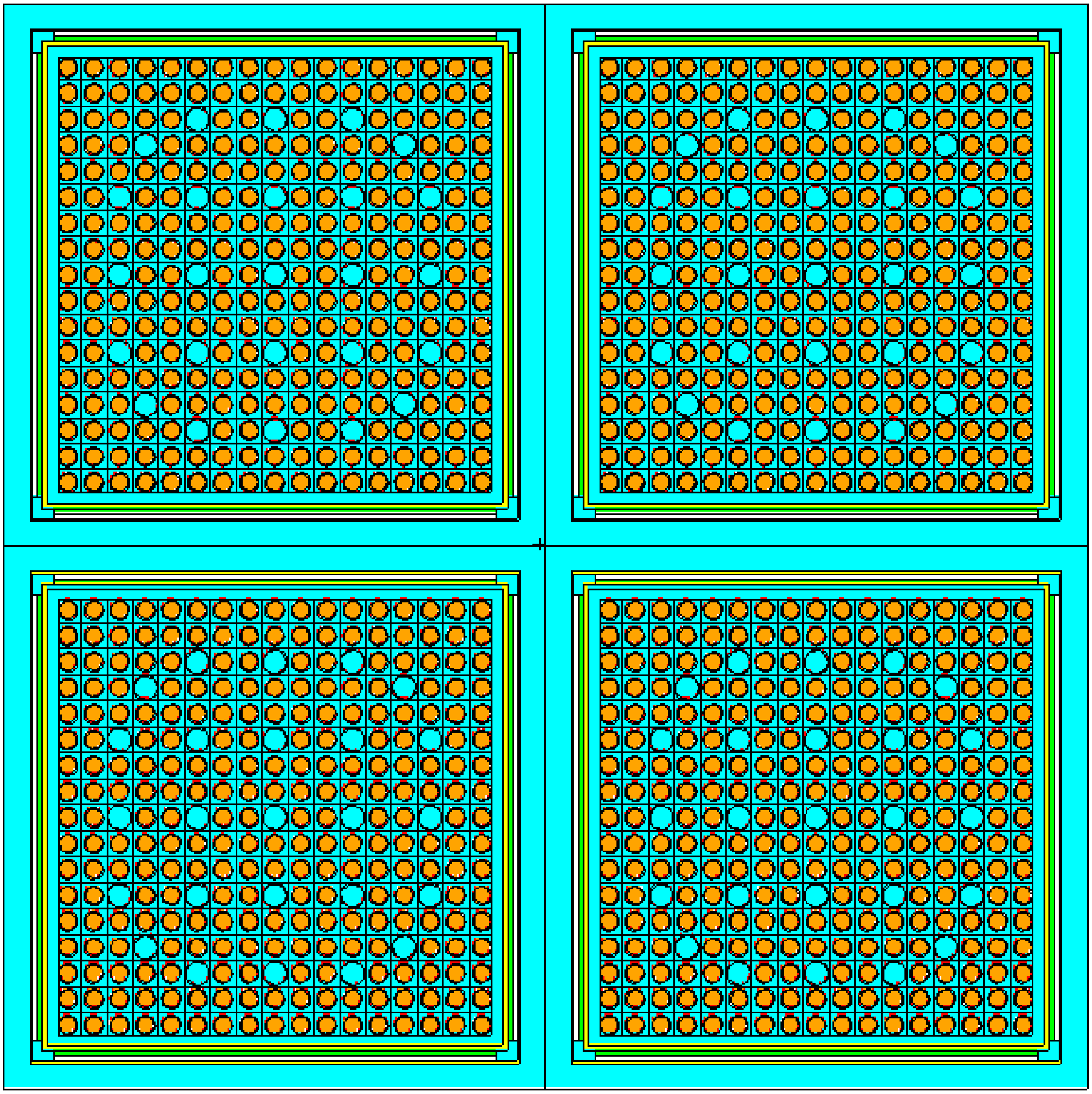


Figure 2.1
MCNP5-1.51 BORAL™ Design Basis Model

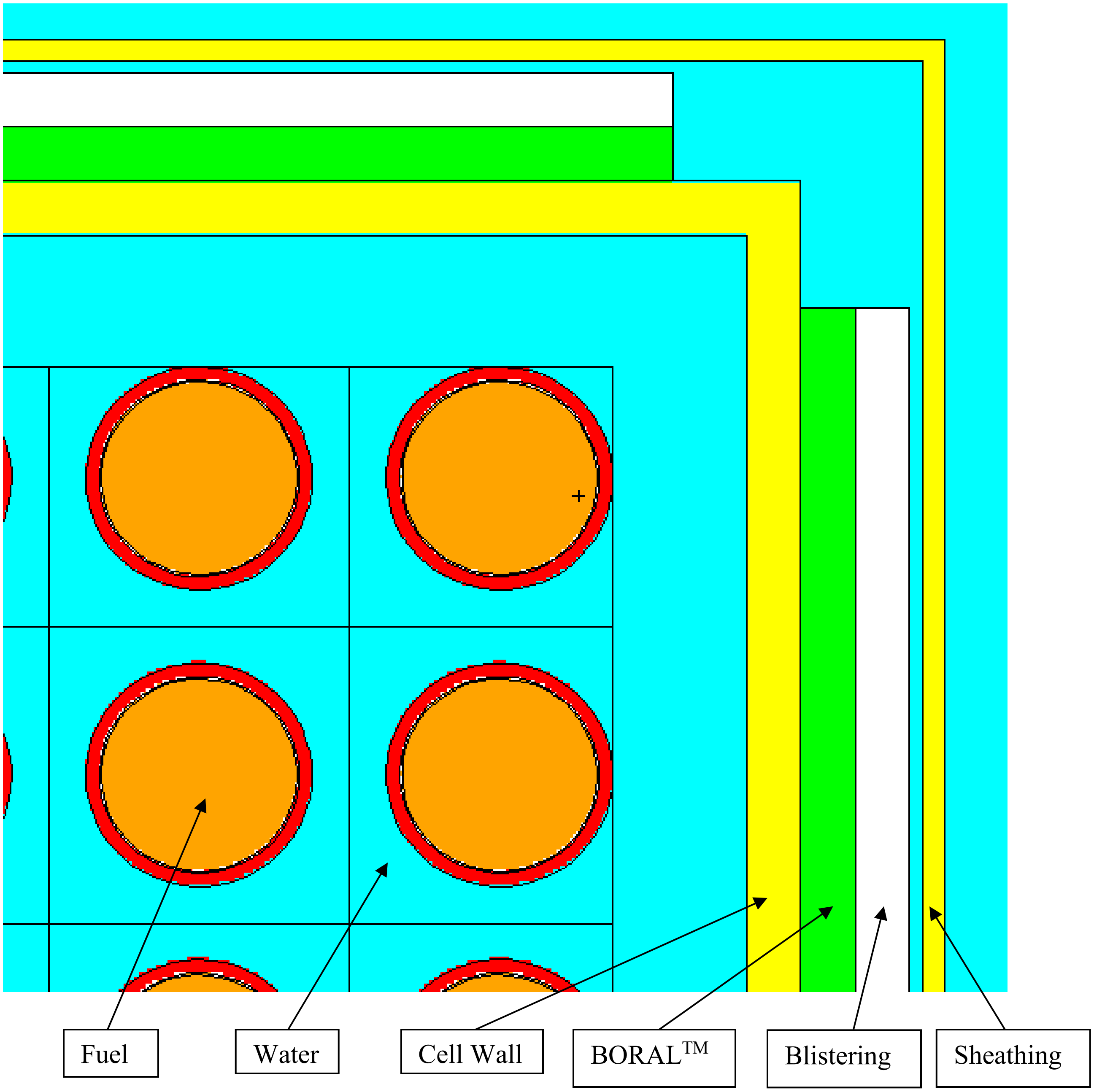


Figure 2.2
Details of MCNP5-1.51 Design Basis Model for BORAL™ Blistering

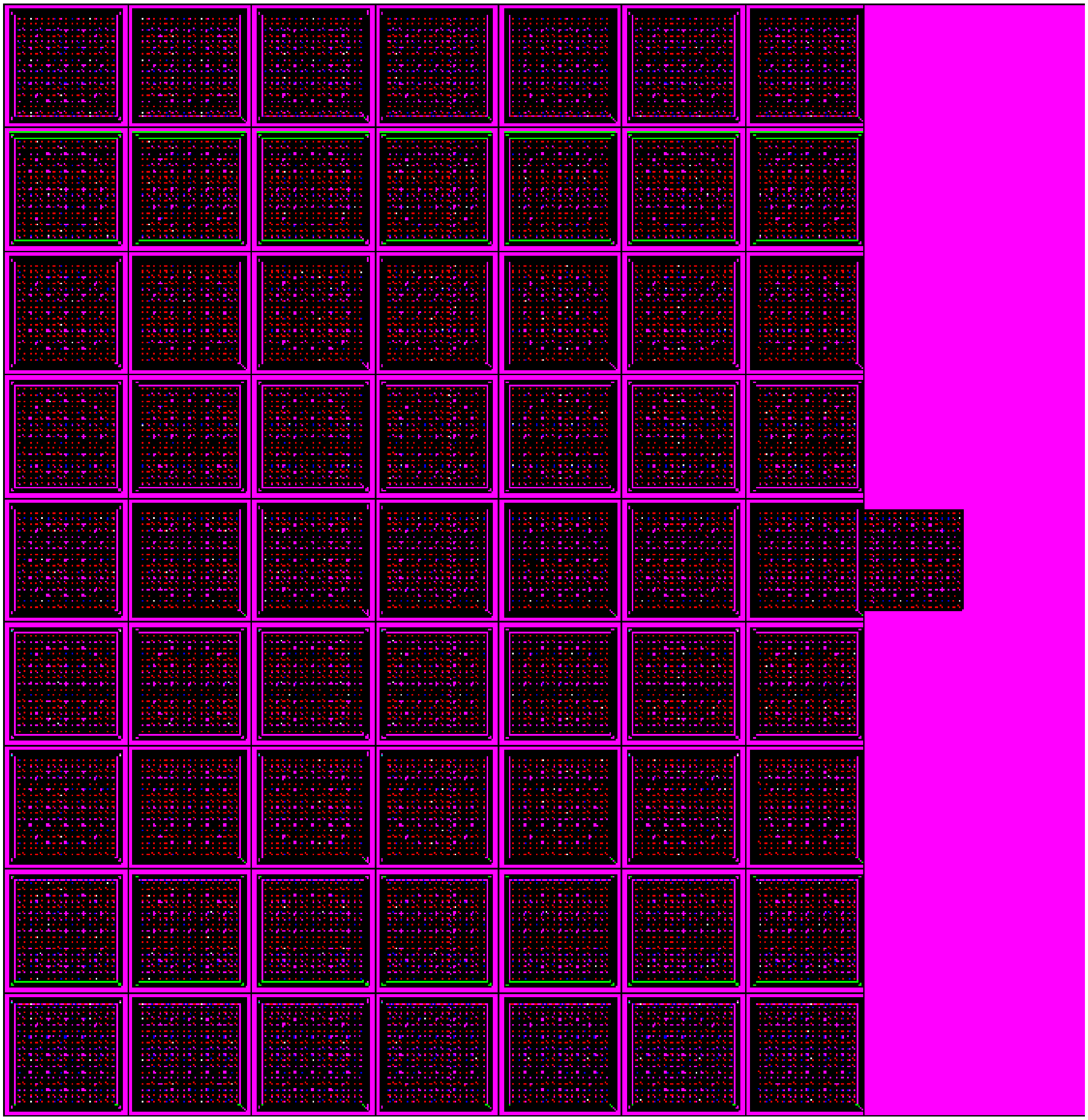
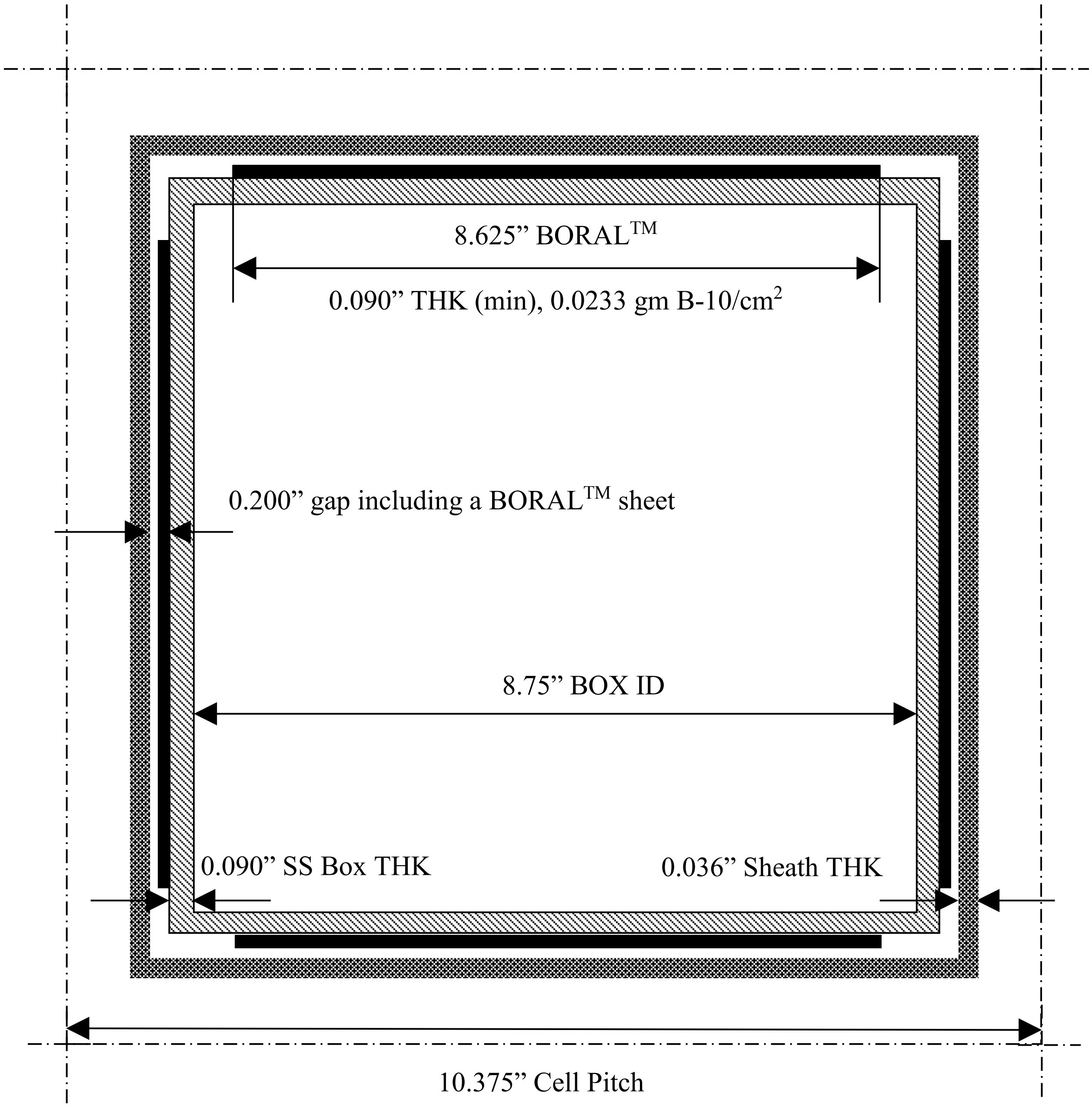


Figure 2.3
Mislocated Accident Configuration



NOT TO SCALE

Figure 5.1
Sketch of the BORAL™ Storage Racks

Appendix A

Analysis Results
(number of pages: 9)

Table A.1 Reactivity of Different Fuel Designs

Case	Fuel Type	k_{calc}	σ
2.3.1.1	V5H	0.9760	0.0004
2.3.1.2	V+/P+	0.9759	0.0004
2.3.1.3	RFA-2	0.9742	0.0004

Table A.2 Reactivity Effect of Fuel Design Parameters

Case	Description	k_{calc}	σ	Delta k_{calc}
2.3.2.1	reference case	0.9759	0.0004	n/a
2.3.2.2	Minimum cladding thickness	0.9790	0.0004	**0.0042**
2.3.2.3	Maximum cladding thickness.	0.9738	0.0004	-0.0010
2.3.2.4	Minimum fuel rod pitch.	0.9750	0.0004	0.0002
2.3.2.5	Maximum fuel rod pitch.	0.9772	0.0004	**0.0024**
2.3.2.6	Minimum fuel pellet OD	0.9761	0.0004	**0.0013**
2.3.2.7	Maximum fuel pellet OD	0.9756	0.0004	0.0008
2.3.2.8	Minimum guide tube/instrument tube thickness	0.9764	0.0004	**0.0016**
2.3.2.9	Maximum guide tube/instrument tube thickness.	0.9748	0.0004	0.0000
Statistical Combination				0.0053

Table A.3 Reactivity Effect of PWR BORAL^{TM} Storage Rack Parameters

Case	Description	k_{calc}	σ	Delta k_{calc}
2.3.3.1	reference case	0.9759	0.0004	n/a
2.3.3.2	Minimum storage cell inner diameter.	0.9735	0.0004	-0.0013
2.3.3.3	Maximum storage cell inner diameter.	0.9782	0.0004	**0.0034**
2.3.3.4	Minimum storage cell wall thickness	0.9758	0.0004	0.0010
2.3.3.5	Maximum storage cell wall thickness	0.9766	0.0004	**0.0018**
2.3.3.6	Minimum storage cell sheathing thickness	0.9760	0.0004	**0.0012**
2.3.3.7	Maximum storage cell sheathing thickness	0.9758	0.0004	0.0010
2.3.3.8	Minimum storage cell poison gap thickness	0.9769	0.0004	**0.0021**
2.3.3.9	Maximum storage cell poison gap thickness	0.9743	0.0004	-0.0005
2.3.3.10	Minimum storage cell flux trap	0.9906	0.0004	**0.0158**
2.3.3.11	Maximum storage cell flux trap	0.9625	0.0004	-0.0123
2.3.3.12	Poison coupon measurement uncertainty [1]	0.9773	0.0004	0.0025
			Statistical Combination	0.0167

[1] Poison coupon measurement uncertainty is explained in Section 2.3.3.

Table A.4 Reactivity Effect of Spent Fuel Pool Water Temperature

Case	Description	k_{calc}	σ
2.3.4.1	Reference case, Temperature of 4 °C	0.9759	0.0004
2.3.4.2	Minimum nominal temperature 20 °C	0.9758	0.0004
2.3.4.3	Maximum possible temperature 120 °C	0.9588	0.0004
2.3.4.4	Maximum possible temperature 120 °C with 10% void	0.9304	0.0004

Table A.5 Reactivity Effect of Fuel Radial Positioning

Case	Description	k_{calc}	σ	Delta k_{calc}
2.3.5.1	Reference case, all assemblies centered in the cell	0.9759	0.0004	n/a
2.3.5.2	eccentric to rack center of 2x2 array	0.9731	0.0004	-0.0017
2.3.5.3	eccentric to rack corner of 2x2 array	0.9726	0.0004	-0.0022
2.3.5.4	eccentric to rack center of 8x8 array	0.9727	0.0004	-0.0021

Table A.6 Reactivity Effect of BORAL™ Blistering

Case	Blistering Void Thickness, inches	k_{calc}	σ	Delta k_{calc}
2.3.6.1	**Reference case, 0.09**	**0.9759**	**0.0004**	**n/a**
2.3.6.2	0.08	0.9737	0.0004	-0.0011
2.3.6.3	0.07	0.9717	0.0004	-0.0031
2.3.6.4	0.06	0.9689	0.0004	-0.0059
2.3.6.5	0.05	0.9675	0.0004	-0.0073
2.3.6.6	0.04	0.9659	0.0004	-0.0089
2.3.6.7	0.03	0.9635	0.0004	-0.0113
2.3.6.8	0.02	0.9615	0.0004	-0.0133
2.3.6.9	0.01	0.9591	0.0004	-0.0157
2.3.6.10	0	0.9569	0.0004	-0.0179

Table A.7 A Single Fresh Fuel Assembly in Water

Case	Description	k_{calc}	σ
2.5.1.1	single fuel assembly in water, 0 ppm water	0.9367	0.0004
2.5.1.2	single fuel assembly in water, 500 ppm water	0.8441	0.0004

Table A.8 Reactivity of Mislocated Fuel Accident

Case	Description	k_{calc}	σ
2.6.5.1	Reference case, fuel centered in the cell, 0 ppm water	0.9791	0.0004
2.6.5.2	Fuel eccentrically positioned to the mislocated fuel, 0 ppm water	0.9802	0.0004
2.6.5.3	Fuel centered in the cell, 500 ppm water	0.9229	0.0004
2.6.5.4	Fuel eccentrically positioned to the mislocated fuel, 500 ppm water	0.9198	0.0004

Appendix B:

NEI 12.16 Criticality Analysis Checklist [14]

The criticality analysis checklist is completed by the applicant prior to submittal to the NRC. It provides a useful guide to the applicant to ensure that all the applicable subject areas are addressed in the application, or to provide justification/identification of alternative approaches.

The checklist also assists the NRC reviewer in identifying areas of the analysis that conform or do not conform to the guidance in NEI 12-16 [14]. Subsequently, the NRC review can then be more efficiently focused on those areas that deviate from NEI 12-16 and the justification for those deviations.

Subject	Included	Notes / Explanation
1.0 Introduction and Overview		
Purpose of submittal	Provide a complete up-to-date criticality safety evaluation for the WBN SFP BORAL™ racks designed for storage of PWR fuel.	
Changes requested		
Summary of physical changes	none	
Summary of Tech Spec changes	changes are required for inclusion of soluble boron credit	This needs to be updated by TVA.
Summary of analytical scope	Criticality analysis to ensure that the effective neutron multiplication factor (k_{eff}) is less than 1.0 for the pool flooded with un-borated water, and 0.95 for the pool flooded with borated water.	
2.0 Acceptance Criteria and Regulatory Guidance		
Summary of requirements and guidance	**YES**/NO	
Requirements documents referenced	**YES**/NO	
Guidance documents referenced	**YES**/NO	
Acceptance criteria described	**YES**/NO	
3.0 Reactor and Fuel Design Description		
Describe reactor operating parameters	YES/**NO**	Fresh fuel is used in the analysis
Describe all fuel in pool	**YES**/NO	
Geometric dimensions (Nominal and Tolerances)	**YES**/NO	
Schematic of guide tube patterns	**YES**/NO	
Material compositions	**YES**/NO	
Describe future fuel to be covered	**YES**/NO	Current Tech Spec limits govern future fuel.

Subject	Included	Notes / Explanation
Geometric dimensions (Nominal and Tolerances)	**YES**/NO	
Schematic of guide tube patterns	YES/**NO**	
Material compositions	YES/**NO**	
Describe all fuel inserts	YES/**NO**	Fresh fuel is used in the analysis
Geometric Dimensions (Nominal and Tolerances)		
Schematic (axial/cross-section)		
Material compositions		
Describe non-standard fuel	YES/**NO**	
Geometric dimensions		
Describe non-fuel items in fuel cells	**YES**/NO	Bounded by design basis calculation
Nominal and tolerance dimensions	YES/**NO**	
4.0 Spent Fuel Pool/Storage Rack Description		
New fuel vault & Storage rack description	YES/**NO**	Not applicable
Nominal and tolerance dimensions		
Schematic (axial/cross-section)		
Material compositions		
Spent fuel pool, Storage rack description	**YES**/NO	a single type of BORAL™ racks
Nominal and tolerance dimensions	**YES**/NO	
Schematic (axial/cross-section)	**YES**/NO	
Material compositions	**YES**/NO	
Other Reactivity Control Devices (Inserts)	YES/**NO**	Fresh fuel is used in the analysis
Nominal and tolerance dimensions		
Schematic (axial/cross-section)		
Material compositions		
5.0 Overview of the Method of Analysis		
New fuel rack analysis description	YES/**NO**	Not applicable
Storage geometries		
Bounding assembly design(s)		
Integral absorber credit		
Accident analysis		
Spent fuel storage rack analysis description	**YES**/NO	
Storage geometries	**YES**/NO	
Bounding assembly design(s)	**YES**/NO	
Soluble boron credit	**YES**/NO	
Boron dilution analysis	**YES**/NO	Supported by Reference [16]
Burnup credit	YES/**NO**	Fresh fuel is used in the analysis
Decay/Cooling time credit	YES/**NO**	Fresh fuel is used in the analysis
Integral absorber credit	YES/**NO**	Fresh fuel is used in the analysis
Other credit	YES/**NO**	Fresh fuel is used in the analysis
Fixed neutron absorbers	**YES**/NO	
Aging management program	YES/**NO**	This needs to be updated by TVA
Accident analysis	**YES**/NO	
Temperature increase	**YES**/NO	
Assembly drop	**YES**/NO	
Single assembly misload	**YES**/NO	Bounded by Design Basis Calculation
Multiple misload	**YES**/NO	Bounded by Design Basis Calculation
Boron dilution	**YES**/NO	Supported by Reference [16]
Other	**YES**/NO	seismic
Fuel out of rack analysis	YES/**NO**	

Subject	Included	Notes / Explanation
Handling		
Movement		
Inspection		
6.0 Computer Codes, Cross Sections and Validation Overview		
Code/Modules Used for Calculation of k_{eff}	**YES**/NO	
Cross section library	**YES**/NO	
Description of nuclides used	**YES**/NO	
Convergence checks	**YES**/NO	
Code/Module Used for Depletion Calculation	YES/**NO**	Fresh fuel is used in the analysis
Cross section library		
Description of nuclides used		
Convergence checks		
Validation of Code and Library	**YES**/NO	
Major Actinides and Structural Materials	**YES**/NO	
Minor Actinides and Fission Products	**YES**/NO	
Absorbers Credited	**YES**/NO	
7.0 Criticality Safety Analysis of the New Fuel Rack	**NO**	Not applicable
Rack model		
Boundary conditions		
Source distribution		
Geometry restrictions		
Limiting fuel design		
Fuel density		
Burnable Poisons		
Fuel dimensions		
Axial blankets		
Limiting rack model		
Storage vault dimensions and materials		
Temperature		
Multiple regions/configurations		
Flooded		
Low density moderator		
Eccentric fuel placement		
Tolerances		
Fuel geometry		
Fuel pin pitch		
Fuel pellet OD		
Fuel clad OD		
Fuel content		
Enrichment		
Density		
Integral absorber		
Rack geometry		
Rack pitch		
Cell wall thickness		
Storage vault dimensions/materials		
Code uncertainty		
Biases		
Temperature		
Code bias		

Subject	Included	Notes / Explanation
Moderator Conditions		
Fully flooded and optimum density moderator		
8.0 Depletion Analysis for Spent Fuel	**NO**	Fresh fuel is used in the analysis
Depletion Model Considerations		
Time step verification		
Convergence verification		
Simplifications		
Non-uniform enrichments		
Post Depletion Nuclide Adjustment		
Cooling Time		
Depletion Parameters		
Burnable Absorbers		
Integral Absorbers		
Soluble Boron		
Fuel and Moderator Temperature		
Power (		
Control rod insertion		
Atypical Cycle Operating History		
9.0 Criticality Safety Analysis of Spent Fuel Pool Storage Racks		
Rack model	**YES**/NO	
Boundary conditions	**YES**/NO	
Source distribution	**YES**/NO	
Geometry restrictions	YES/**NO**	
Design Basis Fuel Description	**YES**/NO	
Fuel density	**YES**/NO	
Burnable Poisons	**YES**/NO	
Fuel assembly inserts	YES/**NO**	
Fuel dimensions	**YES**/NO	
Axial blankets	YES/**NO**	
Configurations considered	**YES**/NO	
Borated	**YES**/NO	
Unborated	**YES**/NO	
Multiple rack designs	YES/**NO**	Single Rack Design in WBN SFP
Alternate storage geometry	**YES**/NO	
Reactivity Control Devices	YES/**NO**	Fresh fuel is used in the analysis
Fuel Assembly Inserts		
Storage Cell Inserts		
Storage Cell Blocking Devices		
Axial burnup shapes	YES/**NO**	Fresh fuel is used in the analysis
Uniform/Distributed		
Nodalization		
Blankets modeled		
Tolerances/Uncertainties	**YES**/NO	
Fuel geometry	**YES**/NO	
Fuel rod pin pitch	**YES**/NO	
Fuel pellet OD	**YES**/NO	
Cladding OD	**YES**/NO	
Axial fuel position	YES/**NO**	
Fuel content		

Subject	Included	Notes / Explanation
Enrichment	**YES**/NO	
Density	**YES**/NO	
Assembly insert dimensions and materials	**YES**/NO	
Rack geometry	**YES**/NO	
Flux-trap size (width)	**YES**/NO	
Rack cell pitch	**YES**/NO	
Rack wall thickness	**YES**/NO	
Neutron Absorber Dimensions	**YES**/NO	
Rack insert dimensions and materials	YES/**NO**	No rack insert
Code validation uncertainty	**YES**/NO	
Criticality case uncertainty	**YES**/NO	
Depletion Uncertainty	YES/**NO**	Fresh fuel is use in the analysis
Burnup Uncertainty	YES/**NO**	Fresh fuel is use in the analysis
Biases		
Design Basis Fuel design	**YES**/NO	
Minor actinides and fission product worth	YES/**NO**	Fresh fuel is use in the analysis
Code bias	**YES**/NO	
Temperature	**YES**/NO	
Eccentric fuel placement	**YES**/NO	
Incore thimble depletion effect	YES/**NO**	Fresh fuel is use in the analysis
NRC administrative margin	YES/**NO**	A margin evaluation is performed for this purpose.
Modeling simplifications		
Identified and described	**YES**/NO	
10.0 Interface Analysis		
Interface configurations analyzed	**YES**/NO	
Between dissimilar racks	YES/**NO**	Only one type of rack.
Between storage configurations within a rack	**YES**/NO	
Interface restrictions	YES/**NO**	
11.0 Normal Conditions		
Fuel handling equipment	YES/**NO**	
Administrative controls	**YES**/NO	
Fuel inspection equipment or processes	YES/**NO**	
Fuel reconstitution	YES/**NO**	
12.0 Accident Analysis		
Boron dilution	**YES**/NO	Supported by Reference [16]
Normal conditions	**YES**/NO	
Accident conditions	**YES**/NO	
Single assembly misload	**YES**/NO	
Fuel assembly misplacement	**YES**/NO	
Neutron Absorber Insert Misload	YES/**NO**	Not applicable
Multiple fuel misload	**YES**/NO	
Dropped assembly	**YES**/NO	
Temperature	**YES**/NO	
Seismic event/other natural phenomena	**YES**/NO	
13.0 Analysis Results and Conclusions		
Summary of results	**YES**/NO	
Burnup curve(s)	YES/**NO**	No burnup credit is used.

Subject	Included	Notes / Explanation
Intermediate Decay time treatment	YES/**NO**	
New administrative controls	YES/**NO**	
Technical Specification markups	YES/**NO**	
14.0 References	**YES**	
Appendix A: Computer Code Validation:		
Code validation methodology and biases	**YES**/NO	
New Fuel	**YES**/NO	
Depleted Fuel	**YES**/NO	
MOX	**YES**/NO	
HTC	**YES**/NO	
Convergence	**YES**/NO	
Trends	**YES**/NO	
Bias and uncertainty	**YES/**NO	
Range of applicability	**YES/**NO	
Analysis of Area of Applicability coverage	**YES/**NO	

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

Holtec International Application for Withholding Proprietary Information From Public Disclosure

Subject: 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)

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Kimberly Manzione, being duly sworn, depose and state as follows:

(1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld is information provided in HI-2177876, "Licensing Report for the Criticality Safety Analysis of the Watts Bar Nuclear Plant Spent Fuel Pool," which contains Holtec Proprietary information.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

AFFIDAVIT PURSUANT TO 10 CFR 2.390

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, and 4.e above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as

proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF CAMDEN)

Kimberly Manzione, being duly sworn, deposes and says:

That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of her knowledge, information, and belief.

Executed at Camden, New Jersey, this 2nd day of October, 2017.

Kimberly Manzione
Licensing Manager
Holtec International

Subscribed and sworn before me this 2nd day of October, 2017.

Maria C Massi

MARIA C. MASSI
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2020

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 2

List of Commitments

Subject: 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)

1. TVA will replace the containment isolation thermal relief check valves on the Unit 2 supply lines to the containment for the Component Cooling Water System and Essential Raw Cooling Water System with simple relief valves prior to loading TPBARs in the Unit 2 reactor core.