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1101 Market Street, Chattanooga, Tennessee 37402

CNL-16-038

March 31, 2016

10 CFR 50.90

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

Watts Bar Nuclear Plant, Unit 1
Facility Operating License Nos. NFP-90
NRC Docket No. 50-390

Subject: **Application to Revise Technical Specification 4.2.1, "Fuel Assemblies" (WBN-TS-15-03) (TAC No. MF6050) - Radioactive Waste System Design Basis Source Term Supplement to Response to NRC Request for Additional Information - Radiation Protection and Consequence Branch**

Reference:

1. Letter from TVA to NRC, CNL-15-001, "Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies,' (WBN-TS-15-03)," dated March 31, 2015 (ML15098A446)

2. Letter from TVA to NRC, CNL-15-077, "Correction to Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies' (WBN-TS-15-03)," dated April 28, 2015 (ML15124A334)

3. Letter from NRC to TVA, "Watts Bar Nuclear Plant, Unit 1 - Supplemental Information Needed for Acceptance of Requested Licensing Action Regarding Application to Increase Tritium Producing Absorbing Rods (TAC NO. MF6050)," dated May 14, 2015 (ML15127A250)

4. Letter from TVA 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)

5. Letter from TVA 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)

6. Letter from TVA to NRC, CNL-15-172, "Application to Revise Technical Specification 4.2.1, 'Fuel Assemblies' (WBN-TS-15-03) (TAC No. MF6050) - Response to NRC Request for Additional Information - Reactor Systems Branch," dated September 14, 2015 (ML15258A204)

7. Electronic Mail from Jeanne Dion (NRC) to Thomas A. Hess (TVA) and Clinton Szabo (TVA), "TPBAR RAIs Part 3b- ARCB," dated October 2, 2015

8. Letter from TVA 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 (ML15356A831)

9. Letter from TVA 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)

By letter dated March 31, 2015 (Reference 1), Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) to revise Watts Bar Nuclear Plant (WBN), Unit 1 Technical Specification (TS) 4.2.1, "Fuel Assemblies," to increase the maximum number of Tritium Producing Burnable Absorber Rods (TPBARs) that can be irradiated per cycle from 704 to 1,792. The proposed change also revises TS 3.5.1, "Accumulators," Surveillance Requirement (SR) 3.5.1.4 and TS 3.5.4, "Refueling Water Storage Tank (RWST)," SR 3.5.4.3 to delete outdated information related to the Tritium Production Program. TVA provided a correction letter on April 28, 2015 (Reference 2).

By letter dated May 14, 2015 (Reference 3), the Nuclear Regulatory Commission (NRC) requested that TVA provide additional information to supplement the LAR. TVA provided the requested supplemental information in TVA letters dated May 27, 2015, and June 15, 2015 (References 4 and 5, respectively).

By electronic mail dated October 2, 2015 (Reference 7), the NRC requested that TVA provide additional information to support the NRC review of the LAR. TVA provided a response to the NRC Request for Additional Information (RAI) by letter dated December 22, 2015 (Reference 8).

During the public meeting with the NRC on February 5, 2016, TVA informed the NRC that it would supplement its response to Radiation Protection and Consequences Branch (ARCB) RAI 6 submitted by the Reference 8 letter to address issues related to the Radioactive Waste (Radwaste) System Design Basis source term. In addition, TVA has determined that the responses to ARCB RAI 3 and ARCB RAI 4 submitted by the Reference 8 letter should also be revised for consistency. Enclosure 1 to this letter provides the revised responses to ARCB RAI 3, ARCB RAI 4, and ARCB RAI 6.

Enclosure 2 to this letter provides a revised Review of Radiological and Environmental Considerations for Production of Tritium at Watts Bar Nuclear Plant Unit 1 - 1,792 TPBAR Core previously provided in Enclosure 2 to the Reference 1 and Reference 5 letters. Enclosure 3 to this letter provides the revised TVA calculation WBNNAL3003, "Reactor Coolant and Secondary Side Activities In Accordance with ANSI/ANS-18.1-1984," previously provided as Attachment 2 to Enclosure 1 to the Reference 8 letter. Enclosure 4 to this letter provides the revised TVA calculation WBNTSR100, "Design Releases to Show Compliance with 10CFR20," previously provided as Attachment 3 to Enclosure 1 to the Reference 8 letter.

By letters dated March 31, 2015, September 14, 2015, December 22, 2015, and February 22, 2016 (References 1, 6, 8, and 9, respectively), TVA committed to replace containment isolation thermal relief check valves on the WBN, Unit 1 Component Cooling Water (CCW) System and Essential Raw Cooling Water (ERCW) System (Commitment 1). TVA has completed the replacement of the subject thermal relief check valves. Specifically, the existing check valves in the CCW System (1-CKV-070-0687, -0698, -0790) and the ERCW System (1-CKV-067-1054A, -1054B, -1054C, -1054D) were replaced with new pressure relief valves (1-RFV-070-0687, -0698, -0790, 1-RFV-067-1060A, -1060B, -1060C, and-1060D, respectively). Therefore, Commitment 1 is being deleted from the Enclosure 6 Commitment List.

During a telephone conversation with the NRC Project Manager for WBN on March 1, 2016, TVA agreed to propose an Operating License condition in lieu of Regulatory Commitment 2 requiring replacement of the WBN, Unit 1 upper compartment cooler cooling coils as provided in the References 1, 6, 8, and 9 letters. Enclosure 5 to this letter provides a markup of the WBN, Unit 1 Operating License page incorporating new Operating License Condition C(10) requiring replacement of the WBN, Unit 1 upper compartment cooler cooling coils. Enclosure 6 to this letter provides a complete updated commitment list that supersedes the previous commitment lists provided in the References 1, 6, 8, and 9 letters.

Consistent with the standards set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.92(c), TVA has determined that the additional information, as provided in this letter, does not affect the no significant hazards consideration associated with the proposed application previously provided in Reference 1.

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 are no new regulatory commitments associated with this submittal. 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 31st day of March 2016.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:
1. Tennessee Valley Authority, Watts Bar Nuclear Plant, Unit 1, Supplement to Response to NRC Request for Additional Information
2. Review of Radiological and Environmental Considerations for Production of Tritium at Watts Bar Nuclear Plant Unit 1 – 1,792 TPBAR Core, Revision 2
3. WBNNAL3003, Revision 6, "Reactor Coolant and Secondary Side Activities In Accordance with ANSI/ANS-18.1-1984"
4. WBNTSR100, Revision 13, "Design Releases to Show Compliance with 10CFR20"
5. Proposed Operating License Condition (Markup Page)
6. Watts Bar Nuclear Plant, Unit 1 Tritium Producing Burnable Absorber Rods License Amendment Request Regulatory Commitment List, Revision 4

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 1

Tennessee Valley Authority, Watts Bar Nuclear Plant, Unit 1,
Supplement to Response to NRC Request for Additional Information

By letter dated December 22, 2015, TVA responded to Radiation Protection and Consequence Branch (ARCB) Request for Additional Information (RAI) 6. During the public meeting with the NRC on February 5, 2016, TVA informed the NRC that it would revise the Radioactive Waste (Radwaste) System Design Basis source term to address additional NRC comments. As a result of the changes to the Radwaste System Design Basis source term, changes to the TVA responses to ARCB RAI 3, ARCB RAI 4, and ARCB RAI 6 are required. The below revised responses supersede the previous TVA responses to ARCB RAI 3, ARCB RAI 4, and ARCB RAI 6 in their entirety. The changes from the previous responses are shown with revision bars in the right margin.

ARCB RAI 3

Supplement dated June 2015, Enclosure 2, pages 9, 23 and 24, indicate that the amounts of H-3 released from Watts Bar in radwaste discharges are estimated to increase by a factor of about 14 (or up to 26,889 Ci per year). Since this license amendment request is to allow loading a maximum of 1792 TPBARs into the core, and the 26,889 Ci/yr value is based on a TPC of 2500 TPBARs, verify that these statements are incorrect and that there is no intentions of releasing 26,889 Ci/yr of H-3. Clarify and provide corrected text.

TVA Response

TVA has revised the Radioactive Waste (Radwaste) System Design Basis source term to consider 1,900 TPBARs and a permeation rate of 5 Ci/TPBAR/yr. The Radwaste System Design Basis source term is revised from a value of 26,889 Ci/yr of tritium to a value of 11,389 Ci/yr of tritium. The 11,389 Ci/yr of tritium value was used to demonstrate the adequacy of the Radwaste System.

The expected tritium release for operation at the licensed TPBAR loading limit would be 6,093 Ci/year based on 1,792 TPBARs; the expected permeation rate would be 3.4 Ci/yr for the Mark 9.2 TPBAR. Therefore, the tritium release from WBN, Unit 1 is not projected to exceed the established regulatory limits.

Enclosure 2 of this letter provides the revised "Review of Radiological and Environmental Considerations for Production of Tritium at Watts Bar Nuclear Plant Unit 1 – 1,792 TPBAR Core," previously provided as Enclosure 2 to the TVA letter dated June 15, 2015. The revised report incorporates changes required as a result of the changes to the Radwaste System Design Basis source term and also includes changes to pages 9, 23 and 24 identified in ARCB RAI 3.

ARCB RAI 4

Supplement dated June 2015, Enclosure 2, page 23, indicates that the assumed "design basis source term" (from a tritium production core (TPC) of 2500 TPBARs, with a H-3 permeability of 10 CI/TPBAR/yr) and "realistic source term" (from a TPC of 1900 TPBARs, with a H-3 permeability of 5 CI/TPBAR/yr), result in equilibrium H-3 reactor coolant concentrations of 29.8 Ci/gm and 7.0 Ci/gm respectively. Provide calculations, including all assumptions and parameters, demonstrating how these coolant concentrations were derived from their respective assumed source terms.

TVA Response

A copy of WBN calculation WBNNAL3003, Revision 6, "Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984," is provided as Enclosure 3 to this letter. The calculation describes the methodology and assumptions used to calculate the tritium concentrations in the RCS for the radwaste system design basis and routine effluent realistic source terms.

ARCB RAI 6

Supplement dated June 2015, Enclosure 2, pages 24 and 25, indicate that Enclosure 2 Tables 5 through 7 demonstrate that the Effluent Concentration Limits (ECL) in 10 CFR 20 Appendix B are met assuming a TPC of 2500 TPBARs, with a H-3 permeability of 10 CI/TPBAR/yr (i.e., the design source term). However, the TPC H-3 concentration entries in all three of these tables is set at the "maximum allowable tritium concentration" with no explanation of how this value was derived or any nexus to the assumed source term.

a. Provide the basis for this maximum allowable tritium concentration, and how is derived from the design basis source term.

   TVA Response

   A copy of WBN calculation WBNTSR100, Revision 13, "Design Releases to Show Compliance with 10CFR20," is provided as Enclosure 4 to this letter. This calculation no longer determines a maximum allowable tritium concentration. It demonstrates that the entire Radwaste System Design Basis source term can be released within regulatory limits. Tables 5 through 7 have been updated (See Enclosure 2 to this letter) to reflect the new source term.

b. How does TVA propose to administer this maximum allowable tritium concentration? Will it be incorporated into the plant Technical Specifications or some other on-site document?

   TVA Response

   TVA has revised the Radwaste System Design Basis source term to assume a permeation rate of 5 Ci/TPBAR/yr and 1,900 TPBARs. With this revision there is no longer a maximum allowable tritium concentration. WBNTSR100 demonstrates that the entire Radwaste System Design Basis source term can be released within regulatory limits.

c. Demonstrate by calculation that the Watts Bar radwaste processing systems are capable of annually releasing the design basis source term, while maintaining H-3 concentrations below the maximum allowable concentration.

TVA Response

A copy of WBN calculation WBNTSR100, Revision 13, "Design Releases to Show Compliance with 10CFR20," is provided as Enclosure 4 of this letter. TVA has revised this calculation to reflect a new design basis source term of 1,900 TPBARs and a permeation rate of 5 Ci/TPBAR/yr. With this revision there is no longer a maximum allowable tritium concentration. WBNTSR100 demonstrates that the entire Radwaste System Design Basis source term can be released within regulatory limits.

d. Tables 5 and 7 do not demonstrate that dual unit operation will result in annual liquid effluent discharges within the limits of 10 CFR 20, Appendix B. Correct the incorrect text on page 25 and provide a basis for why this is acceptable.

TVA Response

TVA has revised the Radwaste System Design Basis source term to assume 1,900 TPBARs and a permeation rate of 5 Ci/TPBAR/yr. With this revision, Table 7 has been revised as shown in Enclosure 2 of this letter to show that the aggregate ECL is ≤ 1, demonstrating discharges do not exceed the limits of 10 CFR Part 20, Appendix B. Table 5 shows the result without radwaste system processing. WBN, Unit 1 UFSAR Section 11.2.6.5.2 notes that UFSAR Tables 11.2-4a (same information as Table 5 except for revised tritium source term) and UFSAR Table 11.2-4b (same information as Table 6 except for revised tritium source term) describe liquid releases for both untreated and treated waste relative to the requirements of 10 CFR Part 20. The current licensing basis UFSAR Table 11.2-4a shows the sum of the concentrations/ECL for all isotopes is greater than unity for the case where all isotopes are at design values and the released liquid is not processed by the Mobile Demineralizers. The effects of the updated tritium source term shown in Table 5 leads to the same conclusion. Radwaste processing through the Mobile Demineralizers, described in WBN, Unit 1 Updated Final Safety Analysis Report (UFSAR) Chapter 11, demonstrates results that do not exceed 10 CFR Part 20 requirements, as shown in UFSAR Table 11.2-4b (current licensing basis) and Table 6 (for the revised tritium source term).

The results in Tables 6 and 7 represent a conservative (but not realistic) continuous release scenario. The radwaste system processing at WBN, Unit 1 is performed in batch mode. The releases are processed in accordance with the ODCM to ensure compliance with TS 5.7.2.7.b.

The information provided in Enclosure 2 of the June 2015 supplement on page 25 is revised as follows with deleted text lined out and inserted text underlined:

> Tables 6 and~~5 through~~ 7 demonstrate that the liquid releases do not exceed~~are below~~ the 10 CFR Part 20 Appendix B Table 2 limits.

e. Provide a basis for operating at (Table 7) or near (Table 6) the maximum ECLs is ALARA or acceptable.

TVA Response

Tables 6 and 7 are based on the Radwaste System Design Basis source term and are intended to demonstrate the adequacy of the Radwaste System. These tables are not intended to demonstrate that as low as reasonably achievable (ALARA) objectives are met with respect to dose commitment to a member of the public. The ALARA objectives for radioactive effluent releases are met by implementing the ODCM in accordance with WBN, Unit 1 TSs 5.7.2.3 and 5.7.2.7.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1

Review of Radiological and Environmental Considerations for Production of Tritium at Watts Bar Nuclear Plant Unit 1 – 1,792 TPBAR Core

Revision 2

(40 pages including this cover)

The changes from the previous revision provided by TVA letter dated June 15, 2015, are shown with revision bars in the right margin.

Review of Radiological and Environmental Considerations for Production of Tritium at Watts Bar Nuclear Plant Unit 1 – 1,792 TPBAR Core

Revision 2

TENNESSEE VALLEY AUTHORITY

March 14, 2016

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Background

The Department of Energy (DOE) and the Tennessee Valley Authority (TVA) have agreed to cooperate in a program to produce tritium for the National Security Stockpile by irradiating Tritium Producing Burnable Absorber Rods (TPBARs) at Watts Bar Nuclear (WBN) Unit 1.

The initial environmental impacts of producing tritium at WBN Unit 1 were assessed in a Final Environmental Impact Statement (EIS) for the Production of Tritium in a Commercial Light Water Reactor (DOE/EIS - 0288, March 1999) which was prepared by DOE. TVA was a cooperating agency in the preparation of this EIS, and adopted the EIS in accordance with 40 CFR 1506.30 of the Council on Environmental Quality regulations. TVA's *Record of Decision* (ROD*) and Adoption of the Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor* were published in the Federal Register at 65 Federal Register 26259 (May 5, 2000). In addition to the DOE EIS and TVA's ROD, a Tritium Production Core (TPC) Topical Report (NDP-98-181, Revision 1) was prepared by DOE to address the safety and licensing issues associated with incorporating TPBARs in a pressurized water reactor (PWR). The Nuclear Regulatory Commission's (NRC) Standard Review Plan (SRP) NUREG-0800 was used as the basis for evaluating the impact of the TPBARs on a reference plant. The NRC reviewed the TPC Topical Report and issued Safety Evaluation Report (SER) NUREG-1672 to support plant-specific licensing of TPBARs in a PWR.

TVA letter dated August 20, 2001[1], addressed the interface items described in NUREG-1672, Section 5.1, and requested authorization to irradiate not more than 2,304 TPBARs in WBN Unit 1. NRC issued a Safety Evaluation (SE) approving WBN Unit 1 License Amendment 40 on September 23, 2002[2], authorizing WBN Unit 1 to irradiate up to a maximum of 2,304 TPBARs in WBN Unit 1. TVA's application for that amendment provided radiological analyses based on 2,304 Curies (Ci)/year release attributable to TPBARs, based on the TVA functional requirement of one Ci/TPBAR/year for 2,304 TPBARs. The SE recognized that for the 2,304 TPC "licensee calculations demonstrated that doses to the public from effluents and the tritium release concentrations will remain below offsite dose calculation manual (ODCM) limits and 10 CFR Part 20 release limits." The ODCM reflects the plant-specific, applicable requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I. The NRC Environmental Assessment and Finding of No Significant Impact for the 2,304 TPC concluded that The proposed "*action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.*"

TVA notified NRC that TVA had imposed interim administrative limits on the number of TPBARs to be loaded in the WBN, Unit 1 reactor.[3] The interim controls limited the number of TPBARs to be irradiated in any cycle such that the total tritium released into the Reactor Coolant System (RCS) by permeation would remain below the 2,304 Ci value evaluated for WBN Unit 1 License Amendment 40. TVA has maintained the interim administrative limits while higher than expected tritium permeation was investigated.

TVA letter to NRC dated April 25, 2007, included an update on the tritium permeation investigation, including a discussion of the post irradiation examination (PIE) and the Mark 9.2 TPBAR design changes.[4] Based on the PIE and review of the mechanisms associated with tritium transport within the TPBAR, several design changes were made to the TPBARs inserted for Cycle 9. The changes were expected to decrease the tritium permeation and achieve the original tritium permeation goal of less than 1.0 Ci/TPBAR/year. TVA also stated that the

effectiveness of the TPBAR design changes would be determined through the monitoring of RCS tritium levels throughout Cycle 9 operation.

TVA Letter to NRC dated December 31, 2008, "Watts Bar Nuclear Plant (WBN) Unit 1 - Revised Technical Specifications Change WBN-TS-08-04 – Revision to the Maximum Number of TPBARs that Can Be Irradiated in the Reactor Core Per Cycle (TAC No. MD9396)," included an update on the TPBAR tritium release rate through Cycle 8, which showed consistent performance with that observed in Cycle 7.[5]

Radiological and Environmental Impact Considerations – 1,792 TPC[6]

TVA conducted this updated review of the environmental impacts with a particular focus on evaluating the radiological aspects associated with the irradiation of TPBARs at WBN for a 1,900 TPBAR TPC. This review utilizes the updated conservative TPBAR annual release (permeation) rate of 5 tritium Ci/TPBAR/year for the Realistic Basis (i.e., effluent dose calculations) and the Design Basis[7] (i.e., station occupational exposure and radwaste system capability review).

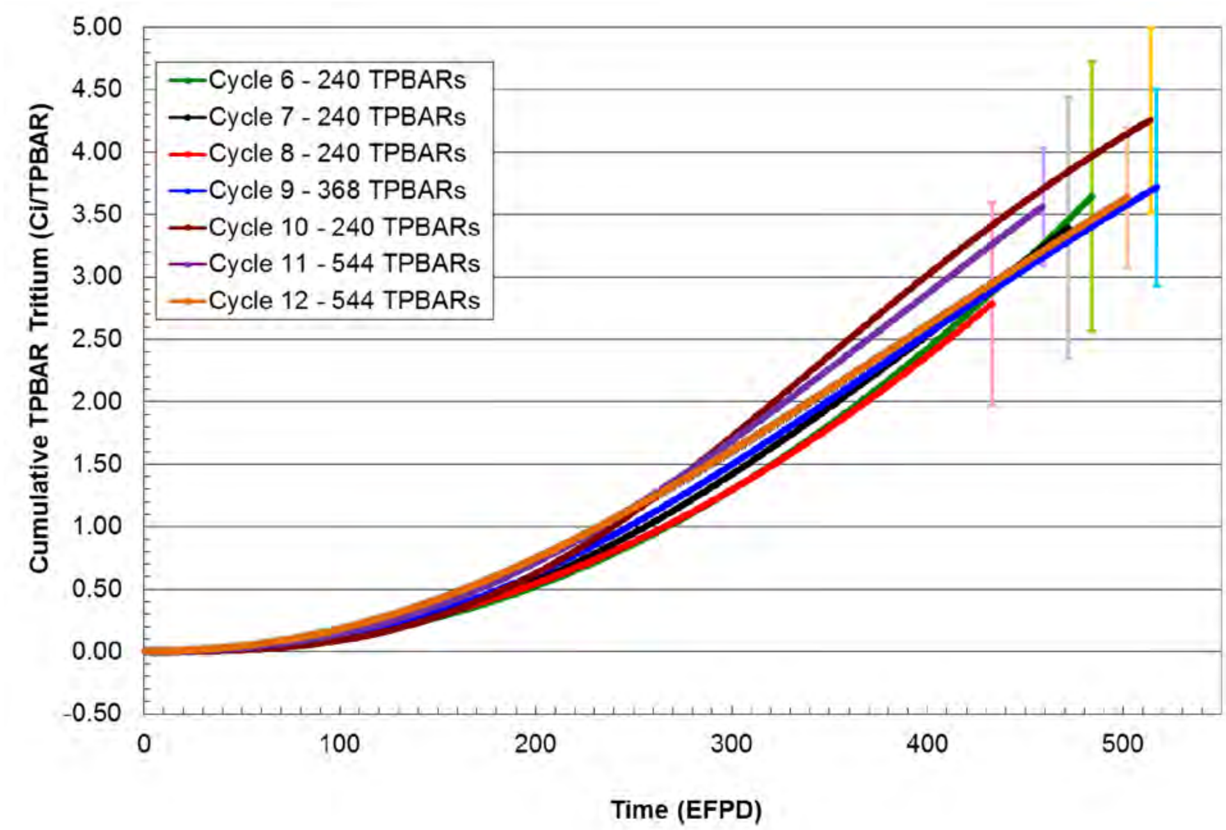
Technical justification: A permeation rate of 5 Ci/TPBAR/year is acceptable because it bounds the observed permeation rate.

Pacific Northwest National Laboratory (PNNL)[8] has estimated the permeation and uncertainty for cycle 12 TPBARs as end of cycle release for cycle 12 was calculated to be 3.6 ± 0.6 Ci/TPBAR. The bounding annual tritium release rate for cycle 12, averaged over the last year of the cycle, was calculated to be 3.2 ± 0.6 Ci/TPBAR/year.

The estimated cumulative tritium permeation per TPBAR for WBN Unit 1 Tritium Production Cycles 6 – 12 are shown on Figure 1.

The Mark 9.2 TPBAR design[9] included significant design changes from the multi-pencil Production TPBAR design of the prior TPC Cycles. However, the average annual release rate per Mark 9.2 TPBAR (3.4 ± 0.8 Ci/TPBAR/year) is similar to that estimated for the multi-pencil Production Design TPBARs of previous WBN Unit 1 cycles.

Figure 1: Estimated TPBAR Permeation for WBN Unit 1 Cycles 6 through 12 (Uncertainty bars represent 90% confidence interval)



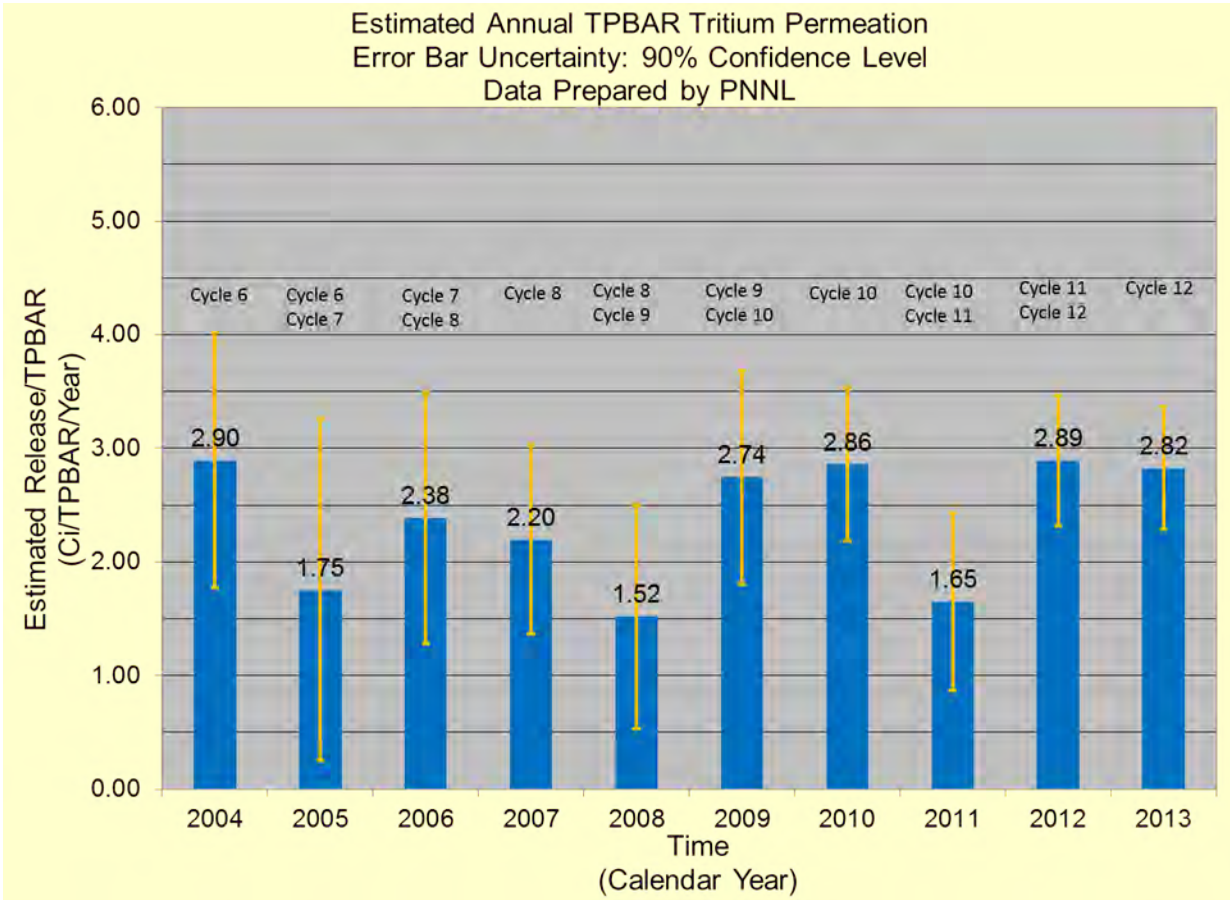
When the TPBAR permeation estimates are presented in a calendar year format (Figure 2), corresponding to the NRC monitoring and reporting requirements, the annualized per TPBAR permeation have consistently remained less than 3 Ci/year. With the approximate 18-month fuel cycles, portions of multiple (i.e., two) fuel cycles will occur periodically in the same calendar years.

Summary of Source Terms

The following source term changes have been made to address the changes in permeation.

Realistic Source Term: This source term is used for evaluation of normal effluent releases and estimating offsite dose. It was updated to reflect a new permeation rate of 5 Ci/TPBAR/year, which bounds the expected permeation rate. The source term was also updated to delete the contribution of two failed TPBARs, because such failures are not expected or realistic. Including the contribution of two failed TPBARs in the prior source term was done for calculation convenience (i.e., all of the sources terms used for different purposes were made to be the same) and represented excessive conservatism.

Figure 2: Estimated Annual TPBAR Permeation for WBN Unit 1 Cycles 6 through 12 (Uncertainty bars represent 90% confidence interval)



Radwaste System Design Basis Source Term: This source term is used for evaluation of radwaste system operation. It was updated to reflect a new permeation rate of 5 Ci/TPBAR/year, which bounds the expected permeation rate. The source term was also updated to delete the contribution of two failed TPBARs, because such failures are not the design basis case for radwaste system operation. It should be noted that the two failed TPBAR assumption previously used reflected an unrealistic and excessively conservative case, because the maximum TPBAR tritium Ci loading occurs at end of cycle life and the maximum RCS tritium concentration due to permeation occurs mid cycle. The mid cycle case is the proper design basis case for radwaste management to support continued operation for the remainder of the cycle. The modified source term provides sufficient margin to bound reasonable off-normal operational cases.

Non-LOCA Accident Source Term: This source term is used for evaluation of abnormal operational occurrences (AOOs) and postulated accidents (PAs). It was updated to reflect a new permeation rate of 10 Ci/TPBAR/year, which bounds the realistic permeation rate with 100% margin. It also has increased margin by assuming ~40% more TPBARs are loaded (i.e., 2,500 TPBARs). It continues to retain conservatism regarding the inclusion of two failed TPBARs. This bounding approach to source term assumptions for accidents is consistent with the accident evaluation methodology. The modified source term provides sufficient margin to bound AOO and PA cases.

Fuel Handling Accident Source Term: The source term used for the fuel handling accident (FHA) is the same as the one used for the License Amendment 40 evaluation. WBN Unit 1 License Amendment 40 and the current licensing basis consider that all of the tritium content of 24 TPBARs is released into the surrounding water. This is the maximum number of TPBARs that would be in one fuel assembly. Because the design inventory of the TPBARs and the maximum number of TPBARs in one fuel assembly remains the same, the source term released to the surrounding water is not affected. Since issuance of License Amendment 40, WBN Unit 1 has modified the licensing basis for an FHA. License Amendment 92 was approved on June 19, 2013 (ML13141A564) and implemented an Alternative Source Term for the FHA. Included in this amendment was also a change to the amount of tritium that was released to the environment. License Amendment 40 assumed a 100% release and License Amendment 92 assumed a 25% release. The amount of tritium released to the environment is not affected by any of the changes being requested and thus, the current licensing basis remains bounding.

LBLOCA Accident Source Term: The source term used for the large break loss of coolant accident (LBLOCA) evaluations remains unchanged from License Amendment 40. The LBLOCA accident source term assumes that all of the end-of-cycle accumulation of tritium is released from 2,304 TPBARs. This assumption is not affected by any of the changes being requested and remains bounding.

In addition, this review incorporates the experiences and lessons from the previous Tritium Production Fuel Cycles at WBN (Cycles 6 – 12). This review addresses both the onsite and offsite potential radiological impacts of tritium production with 1,792 TPBARs.

Updated plant-specific evaluations (and analyses if required) were performed for WBN using the equations and values given in the Watts Bar Updated Final Safety Analysis Report (UFSAR) and ODCM. The review includes the identification of any significant differences between the updated TPBAR permeation estimates with a 1,900 TPBAR Core and the WBN Unit 1 License Amendment 40 associated with the TPBARs and an assessment for potential impacts.

CONCLUSION

- Upon review of the documents and the analyses described above, this review determined that there were no substantial changes, that is, WBN Unit 1 will continue to demonstrate effluent release performance well within the regulatory As Low as Reasonably Achievable (ALARA) public dose guidelines of 10 CFR Part 50 Appendix I and occupational radiation exposure continues to be bounded by the station dose assessment of record[10], associated with the radiological impact analyses that were relevant to environmental concerns, nor were there any significant new circumstances or information relevant to environmental concerns which bore on the radiological impacts associated with the tritium production program. The impact of WBN Unit 1 operation with a TPC containing up to 1,900 TPBARs will have a minimal impact on the Radwaste System Design Basis and realistic fission and corrosion product sources and the treatment of these isotopes in liquid and gaseous waste[11].
- As indicated in Table 10, "Annual Projected Impact of TPC (1,900 TPBARs) on Effluent Dose to Maximally Exposed Members of the Public and Total Public Dose," the differences noted in the source terms for TPC operation would not affect the ability of the plant to remain within the applicable regulatory requirements relative to radioactivity in effluents to unrestricted areas (i.e., 10 CFR Part 20), the "as low as is reasonably achievable" criterion (i.e., 10 CFR Part 50 - Appendix I).
- Updated program controls provide further refinement to the application of the TPBAR permeation performance metric. The permeation performance metric refinements will facilitate the monitoring of TPBAR cycle-to-cycle performance that will allow TVA and DOE to monitor TPBAR permeation performance as a metric for tracking, trending, and evaluating effectiveness of future design changes and TPBAR performance.

Radiological Impacts of the Proposed Irradiations

Tritium

Tritium is a radioactive isotope of hydrogen with a half-life of 12.3 years. Tritium undergoes beta decay, with a maximum energy of 18.6 KeV. The average energy is 5.7 KeV. This low energy limits the maximum range of a tritium beta to about 6 millimeters in air and 0.0042 millimeters in soft tissue. Therefore, the primary radiological significance of exposure to tritium is in the form of internal exposure. Tritium occurs naturally due to cosmic rays interacting with atmospheric gases. In the most important reaction for natural production, a fast neutron interacts with atmospheric nitrogen.

Chemical Forms and Properties[12]

Tritium is almost chemically identical to the other hydrogen isotopes and can exist in several chemical forms including:

- tritiated water (HTO)
- tritiated gas (HT)
- organically bound tritium (OBT)

Tritiated Water

The most common form of tritium is HTO, where a tritium atom replaces a hydrogen atom in water (H_2O) to form HTO. HTO has the same chemical properties as water and is also odorless and

colorless. The majority of tritium in the atmosphere and Commercial Light Water Reactors (CLWR) is in the form of HTO which can be transferred to humans by inhalation, skin absorption (liquid and vapor), or ingestion of drinking water or food. HTO exposure is generally the most important consideration in assessing dose, because HTO quickly reaches equilibrium with water in the body and is distributed uniformly to all soft tissues. International Commission on Radiological Protection (ICRP) (1979[13]) recommended that internalized HTO be assumed to be completely and instantaneously absorbed and distributed uniformly with all body water. As a result, the concentration in sweat, sputum, urine, blood, perspiration, and exhaled water vapor is taken to be the same. HTO is excreted via urine, feces, sweat, and breath.

Tritiated Gas

HT is formed when a tritium atom replaces a hydrogen atom to form a tritium- hydrogen bond. In its elemental form, HT is an invisible, odorless gas chemically identical to hydrogen gas. HT is relatively inert in biological systems and has a very low uptake into body fluids and tissues. The main exposure pathways of HT include inhalation or skin contact with HT-contaminated surfaces. Releases from tritium processing facilities (such as self-luminous light manufactures, tritium recovery facilities, and nuclear fuel processing facilities) represent the primary source of exposure to HT. HT can be oxidized in the atmosphere to HTO.

Organically Bound Tritium

Following an intake of tritium (typically in the form of HTO) by plants or animals, a fraction of the tritium can become incorporated into organic molecules such as carbohydrates, fats, or proteins and is termed OBT. Within the body, OBT can become incorporated into various compounds such as amino acids, sugars, and structural materials. OBT can also enter the body directly by ingestion of tritiated food, by inhalation of volatile organic vapors or aerosols.

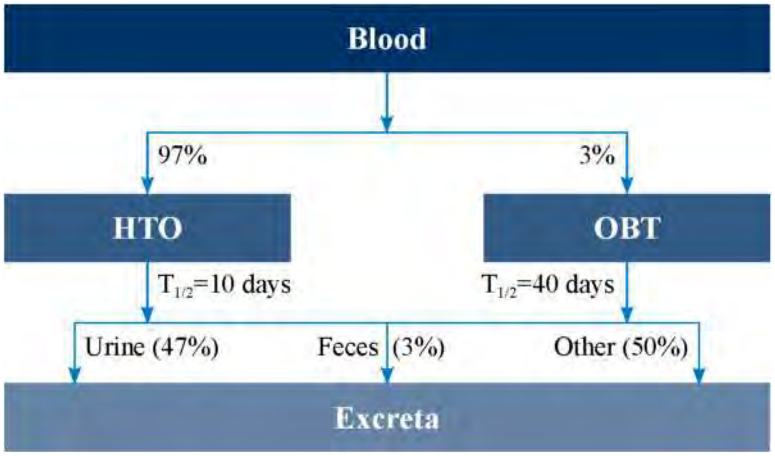
Tritium Summary

- Tritium is the only radioactive isotope of the element hydrogen.
- Tritium atoms can replace hydrogen in water molecules to form HTO, in organic molecules to form OBT and in air to form HT.
- Tritium is one of the lowest energy beta particle emitters. When it is incorporated into the body, more tritium is required than other radioisotopes to cause the same dose.
- Tritium occurs both naturally and as a by-product of the operation of nuclear power and research reactors. It can pose a health risk if it is ingested through drinking water or food, or if it is inhaled or absorbed through the skin in large quantities.

Dosimetric Considerations

Radiation doses from tritium cannot be measured directly and so are usually estimated by measuring the tritium in bioassay samples (such as urine) or through environmental monitoring (air sampling). Once an estimate of the quantity of tritium in the body is made, the dose can be calculated by using biological models that estimate the concentration of tritium in organs and tissues. For intakes as HTO, the current model assumes instantaneous translocation to blood. It is further assumed that HTO is transferred from the blood, with a biological half-life of 6 hours and then distributed uniformly throughout the body. The model assumes that 97% of tritium taken in remains as HTO once distributed, while 3% is converted to OBT. In adults, HTO is retained with a biological half-life of 10 days, and OBT is retained with the biological half-life of carbon, which is 40 days.

Figure 3: ICRP Model for the Biokinetics of Tritiated Water[14]



The committed effective dose-per-unit intake (dose coefficient) to adults resulting from the intake of HTO, as recommended by the ICRP (ICRP, 1993[15]; ICRP, 1995a[16]), is based on Figure 3. This model considers the ICRP's recommendations for radiation weighting factors as well as tissue weighting factors. The committed effective dose-per-unit intake is the computed effective dose received up to 50 years following a single intake for adults, and up to 70 years for intakes by infants and children. The value for intakes of HTO by adults computed by the ICRP is 1.8×10^{-11} Sv/Bq (0.066 mrem/µCi).

The Federal Guidance Report 11[17] value (Technical basis for Appendix B to Part 20—Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage[18] for adult committed effective dose-per-unit tritium intake is 1.73×10^{-11} Sv/Bq (0.064 mrem/µCi). Current NRC Regulations were developed prior to the latest ICRP recommendations and do not account for OBT.

Tritium Analysis

Because of the low beta energies, 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 radioactive sources.

Tritium Source Terms

Regarding tritium sources, in a non-Tritium Production Core (non-TPC), the production of tritium in the RCS is primarily the result of tritium production/release from:

- Fuel Rods (Ternary fission and Integral Fuel Burnable Absorbers (IFBAs)),
- Control Rods,
- Secondary neutron source rods,
- Wet Annular Burnable Absorbers (WABAs),

- RCS Deuterium (Heavy Water) activation,
- RCS Boron activation, and
- RCS Lithium activation.

A review of Westinghouse PWR benchmark tritium data[19] indicates a nominal production/release tritium value of about 870 Ci/year/unit. This nominal value is consistent with the 845 Ci/year/unit average non-TPC tritium effluent total observed over the four year period (1997 – 2000) at WBN and Sequoyah (SQN).

Tritium Source Term Definition and Discussion

Following the review guidance in Chapter 11, "Source Terms," in NUREG-800 Standard Review Plan[20] , 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.[21]

> "*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."

Radwaste System Design Basis Source Terms[22]

Gamma ray sources were considered in the plant design; they included fission and corrosion product sources, as well as activation sources such as the nitrogen-16 activity in the primary coolant. The changes in nuclide inventories were addressed in the tritium production NRC SE for WBN Unit 1 License Amendment 40[23], "*TVA has performed an analysis of the radioisotope inventory for a TPC using the ORIGEN2.1 computer code. A comparison of noble gas and iodine activities for a conventional core and a TPC core is provided in Table 2.11.2-1. The Iodine inventories are generally less, with the exception of Iodine 131. The analysis resulted in a minimal increase in this isotope of approximately 2 percent. This increase can be attributed to modeling differences and is not considered significant. This table shows that the isotopic concentrations of the more important noble gases are less for the TPC than for a conventional core."* The August 20, 2001 submittal addressed a TPC with 2,304 TPBARs and is considered bounding for the other than tritium nuclides.

Table 1: License Amendment 40 ORIGEN2.1 Radioisotope Non-TPC and TPC Comparison

Table 2.11.2-1 Comparison of Core Noble Gas and Iodine Activities for a Conventional Core to a Tritium Producing Core(Note 1)		
Isotope	**Total Core Inventory (Curies)**	
	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 cycle quantity of tritium produced and the Radwaste System Design Basis production/release in the RCS (Radwaste System Design Basis activity levels are considered in the process capacity design of plant systems and shielding) for the WBN reactor may be found in Table 11A-1 of the UFSAR. The annual Radwaste System Design Basis for a non-TPC is summarized in Table 2.

Table 2: Non-TPC Tritium Production/Radwaste System Design Basis Values (Annual per UFSAR Table 11A-2)

Tritium Source	Total Produced (Ci/year)	Design Release to RCS (Ci/year)
Indirect/reactor component	14,057	1,462
Direct/Soluble	427	427
Total	14,484	1,889

Notes: Power level = 3565 Megawatt (thermal)

Release fraction from fuel = 10% Design, 2% Expected

Release fraction from burnable poison rods = 10% Design, 2% expected

Operating time = 495 days

RCS Lithium concentration = 2.2 parts per million (ppm)

Initial RCS boron concentration = 1100 ppm.

The annual Radwaste System Design Basis tritium production/release with 1,900 TPBARs is summarized in Table 3.

Table 3: TPC Tritium Production/Radwaste System Design Basis Values (Annual per UFSAR Table 11A-1)

Tritium Source	Total Produced (Ci/year)	Design Release to RCS (Ci/year)
Non-TPC Tritium	14,484	1,889
TPBARs (1,900 with 5 Curies/TPBAR/Year permeation)	11,652,700	9,500
Total	11,667,184	11,389

Notes: At the end of the operating cycle, the maximum available tritium in a single TPBAR is calculated to be about 11,600 Ci (or 7,733 Ci/year). The average TPBAR will contain about 9,200 Ci (or 6,133 Ci/year) of tritium at the end of the operating cycle.

Realistic Source Terms

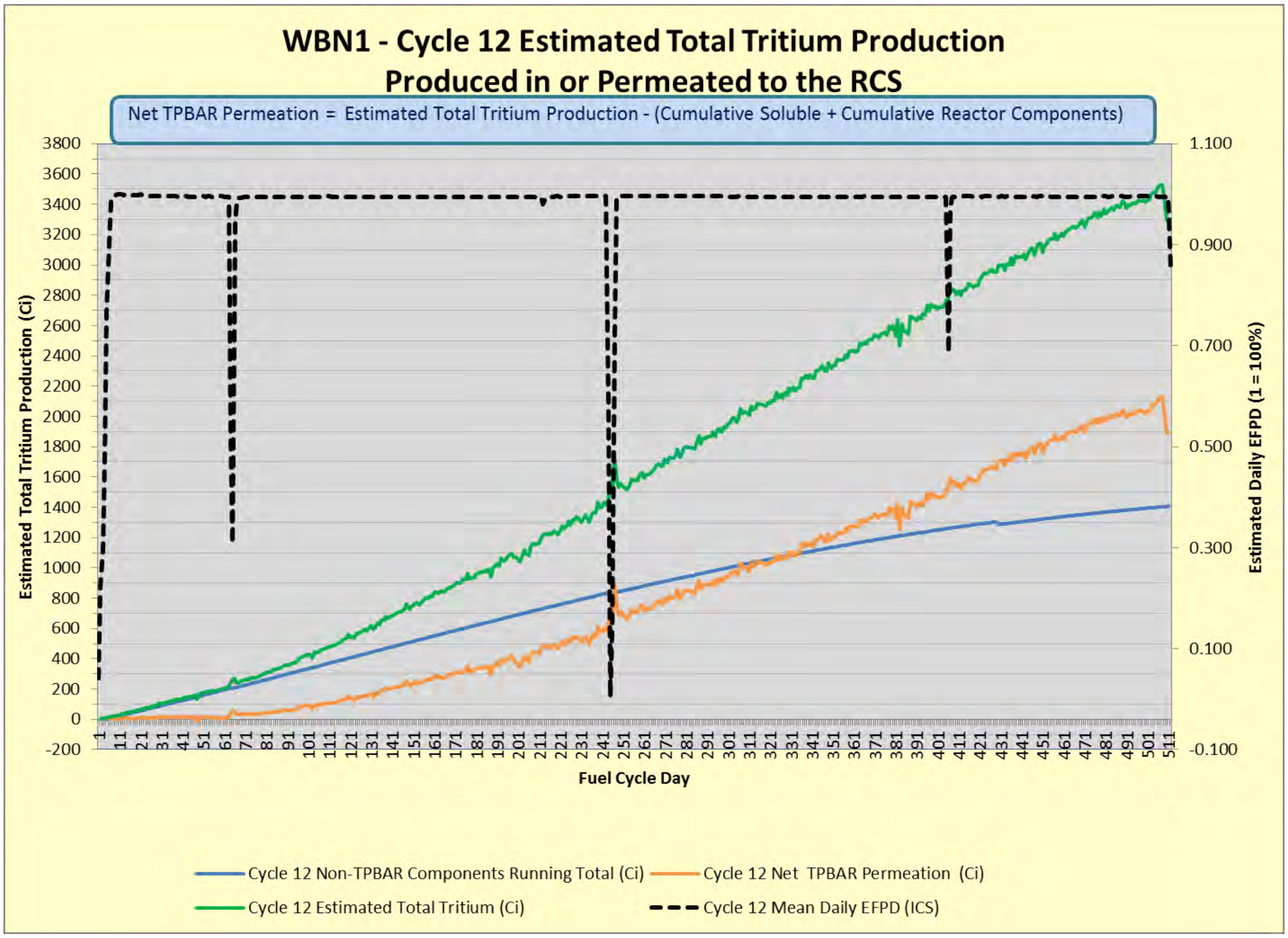
The NRC's regulatory guidance on WBN's nominal tritium production is found in NUREG-0017 R1[24]. The calculated realistic WBN 1 average annual tritium value from NUREG-0017 R1 is 1,392 Ci. The NRC tritium value with the addition of the 9,500 (1,900 TPBARs at 5 Ci/year) for a total average annual 10,892 tritium curies from the TPC was used by TVA to demonstrate continued compliance with the offsite ALARA dose objectives of 10 CFR Part 50 Appendix I.

The Realistic source term was updated from the License Amendment 40 source term to reflect a new permeation rate of 5 Ci/TPBAR/year, which bounds the expected permeation rate. The source term was also updated to delete the contribution of two failed TPBARs, because such failures are not expected or realistic. Including the contribution of two failed TPBARs in the prior source term was done for calculation convenience (i.e., all of the sources terms used for different purposes were made to be the same) and represents excessive conservatism.

WBN Operational Experience with Tritium Production Cores

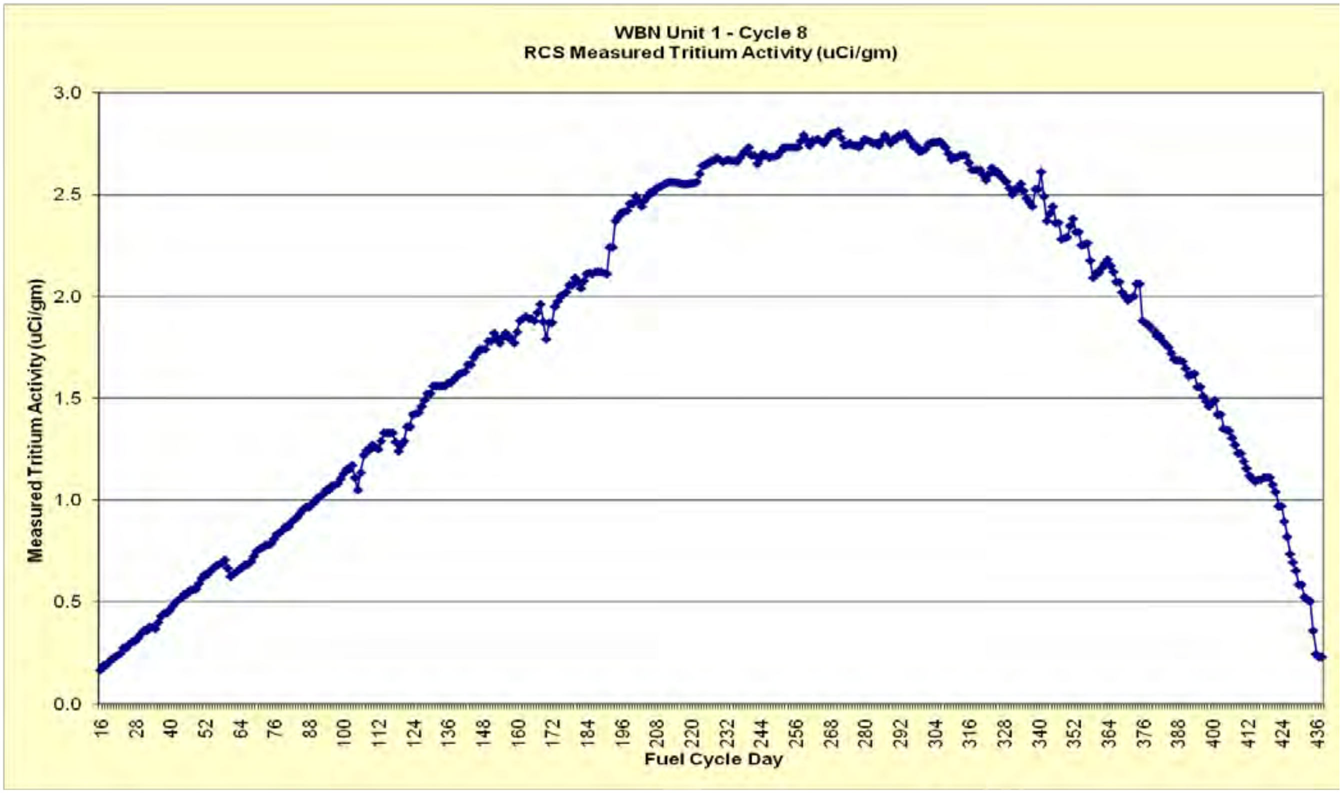
When reviewing station annual tritium effluents, it is important to recognize that plants such as WBN Unit 1 operate with 18-month fuel cycles which tend to generate more non-TPBAR tritium early in the core cycle, owing to higher initial boron concentrations and/or burnable poisons and IFBA rods that are required for reactivity control and more TPBAR-generated tritium later in core life as the tritium inventory within a TPBAR increases from 0 curies at the beginning of the cycle to an average of about 9,200 curies at the end of cycle. Figure 4 provides estimated Cycle 12 daily tritium RCS production/permeation rates for WBN Unit 1 with 540 Mark 9.2 TPBARs and four Lead Use Assemblies. The production by the soluble sources (i.e., Boron, and Lithium) decrease as the RCS is diluted to compensate for fuel burn up (i.e., peak RCS boron 1,615 ppm, End of Cycle RCS boron 62.4 ppm). Reactor component tritium production sources consist of fuel rods, control rods, secondary source rods, and WABAs. Reactor component tritium production tends to remain relatively flat (slight increase) for the cycle. The tritium is produced within the components and permeates through the cladding into the RCS. The TPBAR source term is a function of the number of installed TPBARs and may be related to their physical location within the core (neutron flux and temperature). Estimated TPBAR permeation continues to increase throughout the fuel cycle, with a constant power level. The overall combined tritium producing sources demonstrate increasing daily tritium releases to the RCS. Because of operational constraints and the time required to process RCS discharges for the non-tritium radioactive components, station tritium effluent releases may occur subsequent to the year of production and tritium release to the RCS.

Figure 4: Estimated Daily Tritium Releases to RCS with 540 Mark 9.2 TPBARs and 4 Lead Use Assemblies



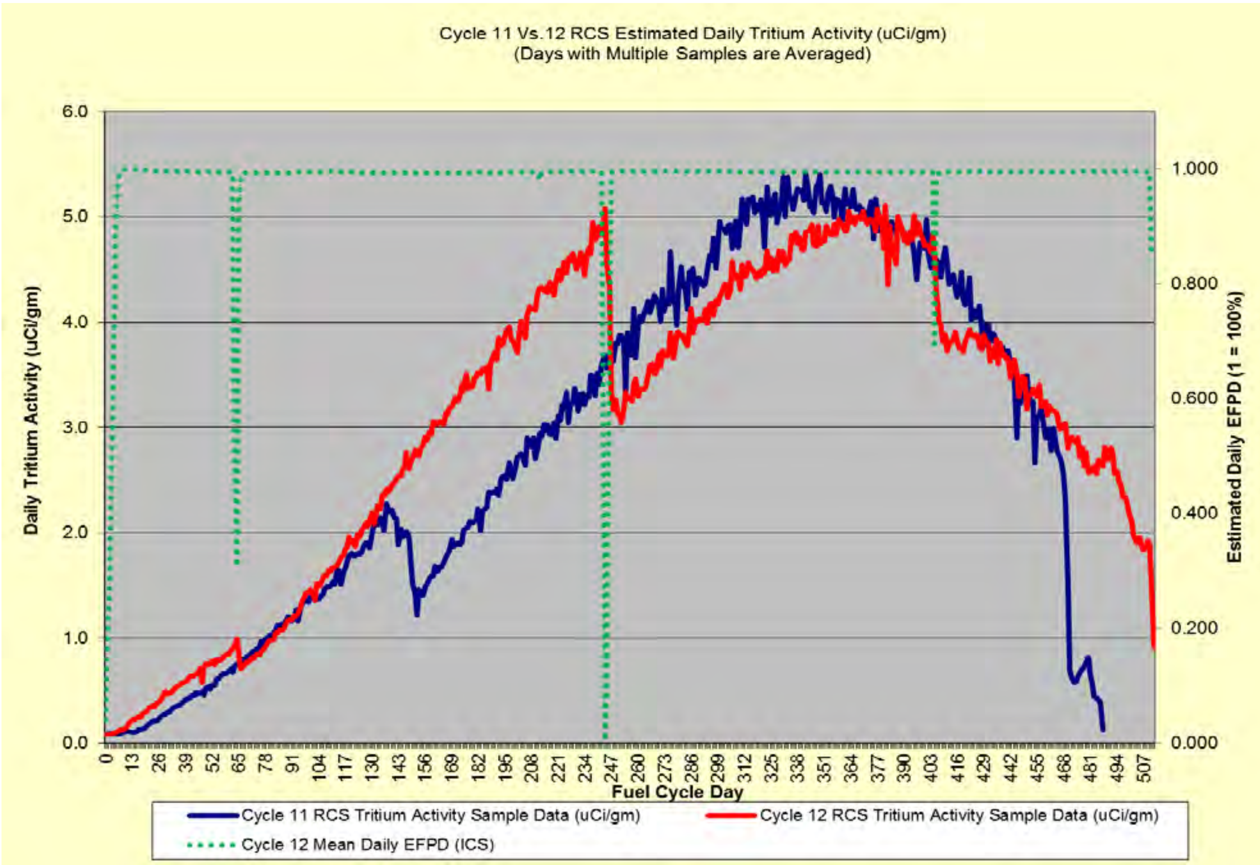
The typical RCS tritium concentrations patterns for breaker–to-breaker runs from WBN Unit 1 Cycle 8 are shown below in Figure 5 as an example. WBN Unit 1 non-tritium production Cycle 3 demonstrated a similar pattern. Cycles with breaker–to-breaker runs tend to demonstrate the highest peak RCS tritium concentrations.

Figure 5: RCS Tritium Concentrations (Breaker–To-Breaker Run) from WBN Unit 1 Cycle 8 (240 TPBARs)



WBN latest complete Tritium Production Cycles 11 and 12 were not breaker–to-breaker runs. The cycles experienced down powers and shutdowns, which resulted in large dilutions and subsequent RCS tritium reductions. A comparison of the daily RCS tritium activity for Cycles 11 (544 TPBARs) and 12 (544 TPBARs) is shown in Figure 6 and demonstrates the variability introduced and effect of down powers and outages on RCS tritium activity.

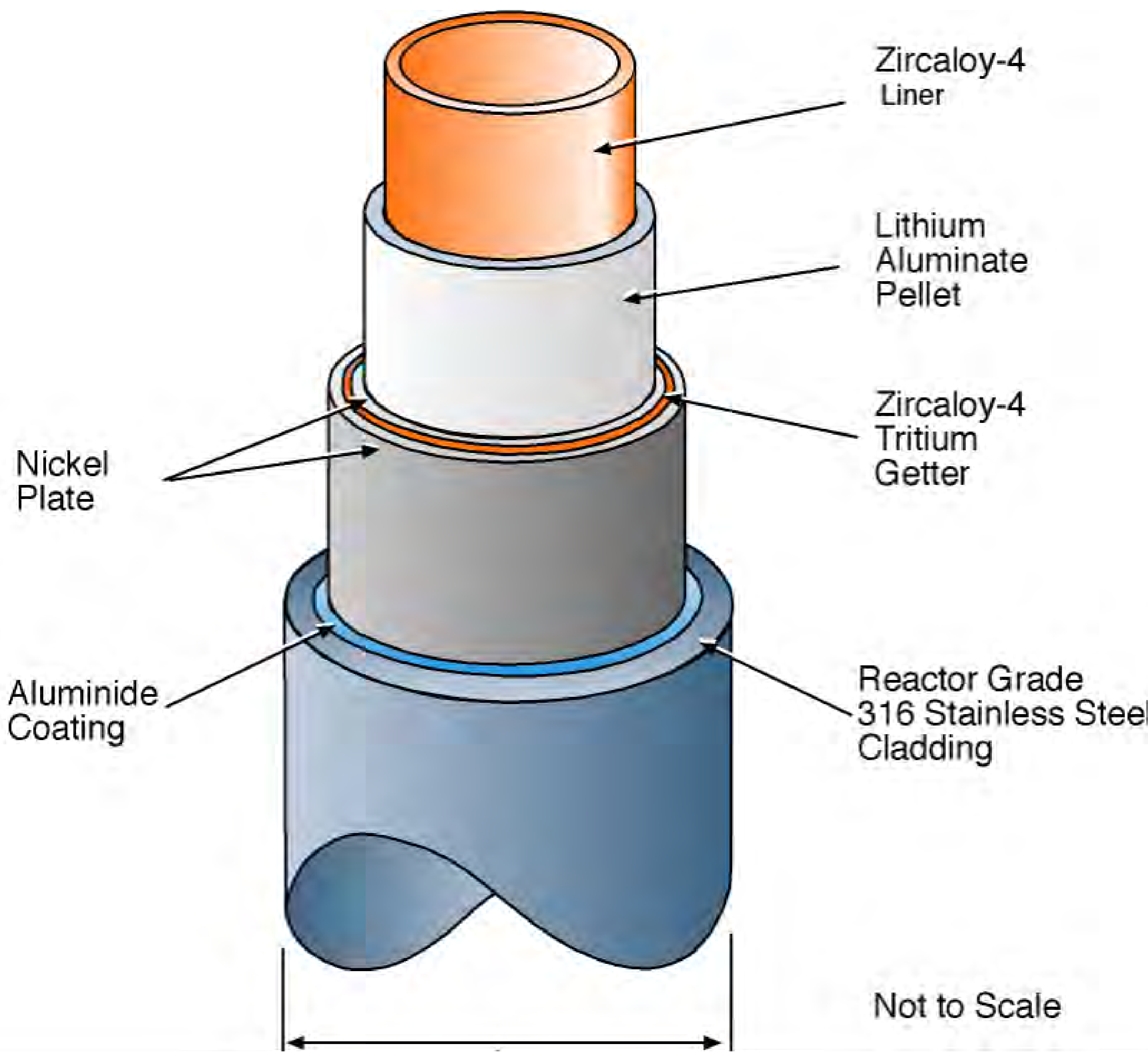
Figure 6: Comparison of the Daily RCS Tritium Activity for Cycles 11 and 12



TPBAR Tritium Permeation

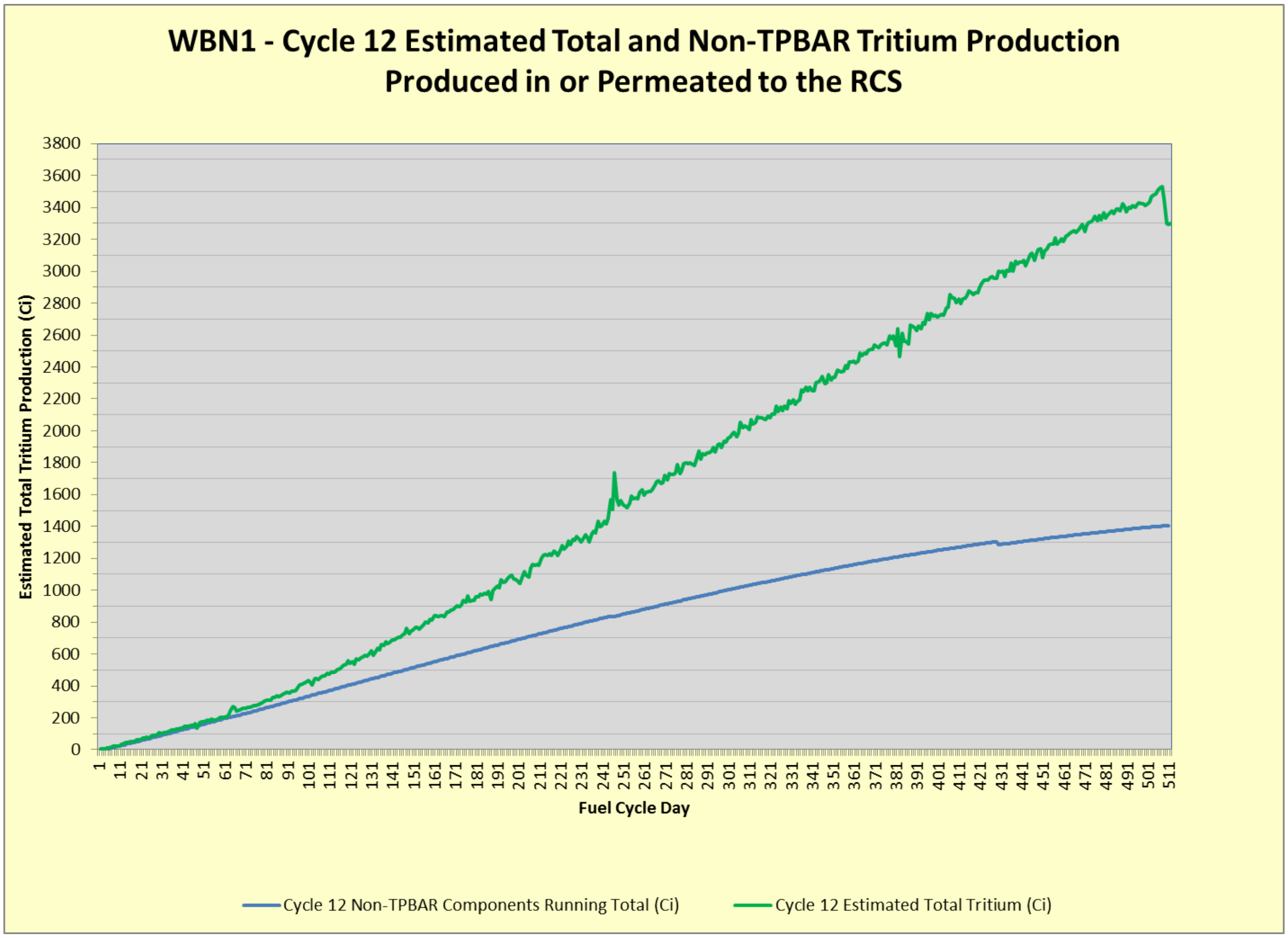
The TPBARs irradiated at WBN are designed with a stainless steel cladding and an aluminized internal coating. The tritium is produced by neutron irradiation of lithium aluminate pellets contained within the cladding and is gettered (collected and retained) by annular zirconium sleeves (getters) around the pellets. The aluminized coating and stainless steel cladding act as a barrier to tritium release (Figure 7). 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.

Figure 7: Concentric, Cylindrical, Internal Components of a TPBAR[25]



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 (Figure 8). 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. Figures 4 and 8 demonstrate this process by using the Cycle 12 estimated tritium production.

Figure 8: Cycle 12 Estimated Total Non-TPBAR and Total Tritium Production/Releases to the RCS



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,

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

$$T_{non\text{-}TPBAR}(t) = T_{Soluble\ Boron}(t) + T_{Lithium}(t) + T_{Fuel\ Rods}(t) + T_{Burnable\ Absorbers}(t) + T_{Control\ Rods}(t) + T_{Secondary\ Source\ Rods}(t)$$[26]

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 – 12 with a 90% uncertainty are shown on Table 4.

Table 4:[27] Estimated TPBAR Permeation for WBN Cycles 6 through 12 (Uncertainty represents 90% confidence interval)

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

Tritium Impacts on Station Operation

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. 10% due to flashing and Spent Fuel Pool (SFP) evaporative losses is the assumed gaseous effluent fraction for dose impact modeling (NUREG-0017, Revision 1), whereas WBN Unit 1 effluent history indicates an average of ≈5.0%. 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, 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 1,900 TPBARs at 5 Ci/TPBAR/year the average RCS tritium concentration is calculated to be 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 derived air concentration (DAC) values are a function of the RCS tritium activity, the transfer of tritium from the RCS to the containment atmosphere (leak rate), and the turnover/dilution of the containment atmosphere through periodic and continuous containment venting and purging. Consistent with 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 TPBARs releasing tritium at the radwaste system design basis maximum rate (i.e., Table 3) will result in about a factor of six increase over the Non-TPC tritium production rate, that is,

Ratio = (TPC) 11,389 Ci/year / (Nominal Core) 1,889 Ci/year = 6.0

The expected tritium release for operation at the licensed TPBAR loading limit would be 6,093 Ci/yr based on 1,792 TPBARs; the expected permeation rate would be 3.4 Ci/yr for the Mark 9.2 TPBAR.

TVA determined that with no modifications to the current boron-control feed and bleed methodologies (366,000 gallon cycle letdown), the design basis RCS average tritium value will be 12 µCi/gm. This mean value would indicate an estimated average containment tritium DAC-fraction of

0.08 DAC-fraction/1 µCi/gm H3 * 12.0 µCi/gm = 0.96 DAC-Fraction

The design basis 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/yr * 12 µCi/gm /1 µCi/gm H3 = 24 person-rem/yr

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[29]. The result was conservatively rounded up to 1 mrem/TPBAR. TVA estimates that when using Radwaste System Design Basis Tritium values for the 1,900 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 current three year collective TEDE per reactor year 2010 – 2012 is 39.998 rem[30]. 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[31] total.

Radwaste System Design Basis Operation

The radwaste system design basis source term was updated from the License Amendment 40 source term to reflect a new permeation rate of 5 Ci/TPBAR/year. The source term was also updated to delete the contribution of two failed TPBARs, because such failures are not the design basis case for radwaste system operation. It should be noted that the two failed TPBAR assumption previously used reflected an unrealistic and excessively conservative case, because the maximum TPBAR tritium Ci loading occurs at end of cycle life and the maximum RCS tritium concentration due to permeation occurs mid cycle. The mid cycle case is the proper design basis case for radwaste management to support continued operation for the remainder of the cycle. The modified source term provides sufficient margin to bound reasonable off-normal operational cases.

The effect of WBN Unit 1 operation with a TPC containing up to 1,900 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[32]. The Radwaste System Design Basis tritium sources are assumed to increase by a factor of about 6. The

analyzed gaseous and liquid releases are based on the radwaste system design basis tritium source term.

Effluent releases to the environment are controlled to meet 10 CFR Part 20 release limits by WBN Unit 1 Technical Specification 5.7.2.7, *Radioactive Effluent Controls Program*. The Radioactive Effluent Release Report is submitted to NRC as required by WBN Unit 1 Technical Specification 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 the 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. Once the fluids are sampled, they are 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 dilution flow is available for the discharge of radioactive liquids. The minimum dilution flow required for discharge of radioactivity into the CTB lines is 20,000 gpm.

WBN has three large tanks in the Liquid Radwaste System, including the new 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.

TVA has demonstrated that effluent releases to the environment can be controlled to meet the 10 CFR Part 20 release limits for an assumed TPBAR loading of 1,900 TPBARs each with an assumed permeation of 5 Ci/TPBAR/year. The assumed number of 1,900 TPBARs exceeds the requested value of 1,792 TPBARs to conservatively bound the projected environmental impacts. Likewise, the assumed permeation rate exceeds the historical average permeation rate of 3.4 Ci/TPBAR/year to conservatively bound the projected environmental impacts. This increase in tritium was added to the current licensing basis source term for a non-TPC to demonstrate conformance with the 10 CFR Part 20 effluent concentration limit values.

Tables 6 and 7 demonstrate that the liquid releases do not exceed the 10 CFR Part 20 Appendix B Table 2 limits. Tables 8 through 9 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 5: Liquid Releases with No Processing of Condensate Resin Regeneration Waste by Mobile Demineralizers

Nuclide	Exp. Rel. (Ci/year)	Des/Exp Ratio	Design (Ci/year)	Design (µCi/gm)	10CFR20 (ECL, µCi/cc)	Single Unit Operation Design C/ECL	Dual Unit Operation Design C/ECL
Br-84	6.94E-04	2.50	1.73E-03	4.36E-11	4.00E-04	1.09E-07	2.18E-07
I-131	1.06E+00	52.41	5.53E+01	1.39E-06	1.00E-06	1.39E+00	2.78E+00
I-132	1.23E-01	4.00	4.93E-01	1.24E-08	1.00E-04	1.24E-04	2.48E-04
I-133	8.40E-01	26.85	2.25E+01	5.67E-07	7.00E-06	8.10E-02	1.62E-01
I-134	3.23E-02	1.65	5.32E-02	1.34E-09	4.00E-04	3.35E-06	6.69E-06
I-135	4.43E-01	7.91	3.50E+00	8.80E-08	3.00E-05	2.93E-03	5.87E-03
Rb-88	1.03E-02	18.14	1.88E-01	4.72E-09	4.00E-04	1.18E-05	2.36E-05
Cs-134	2.38E-01	40.60	9.64E+00	2.42E-07	9.00E-07	2.69E-01	5.39E-01
Cs-136	2.39E-02	165.20	3.95E+00	9.94E-08	6.00E-06	1.66E-02	3.31E-02
Cs-137	3.18E-01	153.22	4.87E+01	1.22E-06	1.00E-06	1.22E+00	2.45E+00
Cr-51	1.21E-01	0.29	3.51E-02	8.82E-10	5.00E-04	1.76E-06	3.53E-06
Mn-54	6.70E-02	0.47	3.15E-02	7.91E-10	3.00E-05	2.64E-05	5.27E-05
Fe-59	1.40E-02	3.48	4.86E-02	1.22E-09	1.00E-05	1.22E-04	2.44E-04
Co-58	2.01E-01	5.37	1.08E+00	2.72E-08	2.00E-05	1.36E-03	2.72E-03
Co-60	4.01E-02	1.38	5.53E-02	1.39E-09	3.00E-06	4.63E-04	9.27E-04
Sr-89	5.36E-03	22.45	1.20E-01	3.02E-09	8.00E-06	3.78E-04	7.56E-04
Sr-90	4.87E-04	13.49	6.57E-03	1.65E-10	5.00E-07	3.30E-04	6.60E-04
Sr-91	2.98E-03	1.86	5.54E-03	1.39E-10	2.00E-05	6.97E-06	1.39E-05
Y-90	0	15.87	0	0	7.00E-06	0.00E+00	0.00E+00
Y-91	4.75E-04	1115.17	5.30E-01	1.33E-08	8.00E-06	1.67E-03	3.33E-03
Zr-95	1.62E-02	1.71	2.78E-02	6.98E-10	2.00E-05	3.49E-05	6.98E-05
Nb-95	1.34E-02	2.34	3.13E-02	7.88E-10	3.00E-05	2.63E-05	5.25E-05
Mo-99	1.26E-01	785.19	9.88E+01	2.48E-06	2.00E-05	1.24E-01	2.48E-01
Te-132	3.64E-02	145.25	5.28E+00	1.33E-07	9.00E-06	1.47E-02	2.95E-02
Ba-140	4.27E-01	0.31	1.32E-01	3.33E-09	8.00E-06	4.16E-04	8.31E-04
La-140	6.14E-01	0.06	3.69E-02	9.26E-10	9.00E-06	1.03E-04	2.06E-04
Ce-144	1.58E-01	0.08	1.26E-02	3.17E-10	3.00E-06	1.06E-04	2.11E-04
Pr-144	0	0.08	0	0.00E+00	6.00E-04	0.00E+00	0.00E+00
H-3	1252.8	1	1252.8	3.15E-05	1.00E-03	3.15E-02	6.30E-02
H-3 (TPC)	9.80E+03	1.0	9.80E+03	2.46E-04	1.00E-03	2.46E-01	2.78E-01
Total						3.16E+00	6.32E+00
Total (TPC)						3.37E+00	6.53E+00

Note: Dual unit operations consider only Unit 1 with TPC.

Table 6: Liquid Releases with Condensate Resin Regeneration Waste Processed by the Mobile Demineralizers

Nuclide	Exp. Rel. (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
Br-84	2.26E-04	2.50	5.65E-04	1.42E-11	4.00E-04	3.55E-08	7.11E-08
I-131	3.70E-02	52.41	1.94E+00	4.87E-08	1.00E-06	4.87E-02	9.74E-02
I-132	1.81E-02	4.00	7.23E-02	1.82E-09	1.00E-04	1.82E-05	3.63E-05
I-133	7.29E-02	26.85	1.96E+00	4.92E-08	7.00E-06	7.03E-03	1.41E-02
I-134	8.57E-03	1.65	1.41E-02	3.56E-10	4.00E-04	8.89E-07	1.78E-06
I-135	6.52E-02	7.91	5.16E-01	1.30E-08	3.00E-05	4.32E-04	8.65E-04
Rb-88	9.41E-03	18.14	1.71E-01	4.29E-09	4.00E-04	1.07E-05	2.15E-05
Cs-134	4.02E-02	40.60	1.63E+00	4.10E-08	9.00E-07	4.55E-02	9.11E-02
Cs-136	3.50E-03	165.20	5.79E-01	1.45E-08	6.00E-06	2.42E-03	4.85E-03
Cs-137	5.52E-02	153.22	8.46E+00	2.13E-07	1.00E-06	2.13E-01	4.25E-01
Cr-51	9.70E-03	0.29	2.81E-03	7.07E-11	5.00E-04	1.41E-07	2.83E-07
Mn-54	6.87E-03	0.47	3.23E-03	8.12E-11	3.00E-05	2.71E-06	5.41E-06
Fe-59	3.31E-03	3.48	1.15E-02	2.90E-10	1.00E-05	2.90E-05	5.80E-05
Co-58	3.18E-02	5.37	1.71E-01	4.29E-09	2.00E-05	2.14E-04	4.29E-04
Co-60	1.97E-02	1.38	2.72E-02	6.83E-10	3.00E-06	2.28E-04	4.55E-04
Sr-89	2.67E-04	22.45	5.98E-03	1.50E-10	8.00E-06	1.88E-05	3.76E-05
Sr-90	3.04E-05	13.49	4.10E-04	1.03E-11	5.00E-07	2.06E-05	4.13E-05
Sr-91	3.90E-04	1.86	7.26E-04	1.82E-11	2.00E-05	9.12E-07	1.82E-06
Y-90	0	15.87	0	0	7.00E-06	0	0
Y-91	1.23E-04	1115.17	1.37E-01	3.45E-09	8.00E-06	4.32E-04	8.63E-04
Zr-95	1.91E-03	1.71	3.27E-03	8.22E-11	2.00E-05	4.11E-06	8.22E-06
Nb-95	2.88E-03	2.34	6.75E-03	1.70E-10	3.00E-05	5.65E-06	1.13E-05
Mo-99	5.85E-03	785.19	4.59E+00	1.15E-07	2.00E-05	5.77E-03	1.15E-02
Te-132	1.55E-03	145.25	2.26E-01	5.67E-09	9.00E-06	6.30E-04	1.26E-03
Ba-140	1.44E-02	0.31	4.46E-03	1.12E-10	8.00E-06	1.40E-05	2.80E-05
La-140	2.28E-02	0.06	1.37E-03	3.43E-11	9.00E-06	3.81E-06	7.63E-06
Ce-144	9.49E-03	0.08	7.59E-04	1.91E-11	3.00E-06	6.36E-06	1.27E-05
Pr-144	0	0.08	0	0	6.00E-04	0	0
H-3	1252.80	1	1252.80	3.149E-05	1.00E-03	3.15E-02	6.30E-02
H-3 (TPC)	9.80E+03	1.0	9.80E+03	2.46E-04	1.00E-03	2.46E-01	2.78E-01
Total						3.56E-01	7.11E-01
Total (TPC)						5.70E-01	9.26E-01

Note: Dual unit operations consider only Unit 1 with TPC.

Table 7: Liquid Releases with No Condensate Resin Regeneration Waste Processed by the Mobile Demineralizers and SGBD at Maximum Allowable Concentration with 20,000 gpm Dilution

Nuclide	LRW (Ci/year)	SGB Ci/year Scaled to 4.40 Ci	Des/EXP Ratio	Des (Ci/year)	Liquid (µCi/cc)	ECL (µCi/cc)	Single Unit Operation C/ECL	Dual Unit Operation C/ECL
Br-84	2.26E-04	1.38E-02	2.50	1.44E-02	3.56E-10	4.00E-04	8.91E-07	1.78E-06
I-131	3.60E-02	2.03E-01	52.41	2.09E+00	5.18E-08	1.00E-06	5.18E-02	1.04E-01
I-132	1.80E-02	4.86E-01	4.00	5.58E-01	1.38E-08	1.00E-04	1.38E-04	2.77E-04
I-133	7.22E-02	5.82E-01	26.85	2.52E+00	6.25E-08	7.00E-06	8.93E-03	1.79E-02
I-134	8.55E-03	4.23E-01	1.65	4.37E-01	1.08E-08	4.00E-04	2.71E-05	5.42E-05
I-135	6.49E-02	8.93E-01	7.91	1.41E+00	3.49E-08	3.00E-05	1.16E-03	2.33E-03
Rb-88	9.41E-03	1.06E-01	18.14	2.77E-01	6.87E-09	4.00E-04	1.72E-05	3.44E-05
Cs-134	4.00E-02	6.61E-02	40.60	1.69E+00	4.19E-08	9.00E-07	4.66E-02	9.31E-02
Cs-136	3.48E-03	8.02E-03	165.20	5.83E-01	1.45E-08	6.00E-06	2.41E-03	4.83E-03
Cs-137	5.50E-02	8.82E-02	153.22	8.51E+00	2.11E-07	1.00E-06	2.11E-01	4.22E-01
Na-24	2.54E-02	2.68E-01	1.00	2.94E-01	7.29E-09	5.00E-05	1.46E-04	2.92E-04
Cr-51	9.59E-03	2.25E-02	0.29	2.53E-02	6.28E-10	5.00E-04	1.26E-06	2.51E-06
Mn-54	6.81E-03	1.13E-02	0.47	1.45E-02	3.59E-10	3.00E-05	1.20E-05	2.39E-05
Fe-55	1.11E-02	8.49E-03	1.00	1.95E-02	4.85E-10	1.00E-04	4.85E-06	9.70E-06
Fe-59	3.30E-03	2.08E-03	3.48	1.36E-02	3.37E-10	1.00E-05	3.37E-05	6.74E-05
Co-58	3.01E-02	3.29E-02	5.37	1.94E-01	4.82E-09	2.00E-05	2.41E-04	4.82E-04
Co-60	1.97E-02	3.81E-03	1.38	3.09E-02	7.68E-10	3.00E-06	2.56E-04	5.12E-04
Zn-65	5.22E-04	3.64E-03	1.00	4.16E-03	1.03E-10	5.00E-06	2.06E-05	4.13E-05
Sr-89	2.61E-04	9.87E-04	22.45	6.86E-03	1.70E-10	8.00E-06	2.13E-05	4.25E-05
Sr-90	3.00E-05	8.49E-05	13.49	4.89E-04	1.21E-11	5.00E-07	2.43E-05	4.85E-05
Sr-91	3.88E-04	5.08E-03	1.86	5.80E-03	1.44E-10	2.00E-05	7.20E-06	1.44E-05
Y-90	0	0	15.87	0	0	7.00E-06	0	0.00E+00
Y-91m	2.30E-04	6.26E-04	1.00	8.56E-04	2.12E-11	2.00E-03	1.06E-08	2.12E-08
Y-91	1.23E-04	3.64E-05	1115.17	1.37E-01	3.40E-09	8.00E-06	4.25E-04	8.50E-04
Y-93	1.73E-03	2.16E-02	1.00	2.34E-02	5.80E-10	2.00E-05	2.90E-05	5.80E-05
Zr-95	1.90E-03	2.77E-03	1.71	6.02E-03	1.49E-10	2.00E-05	7.46E-06	1.49E-05
Nb-95	2.87E-03	1.91E-03	2.34	8.63E-03	2.14E-10	3.00E-05	7.14E-06	1.43E-05
Mo-99	5.73E-03	4.37E-02	785.19	4.54E+00	1.13E-07	2.00E-05	5.63E-03	1.13E-02
Tc-99m	4.57E-03	2.02E-02	1.00	2.48E-02	6.15E-10	1.00E-03	6.15E-07	1.23E-06
Ru-103	8.03E-03	5.37E-02	1.00	6.17E-02	1.53E-09	3.00E-05	5.10E-05	1.02E-04
Ru-106	1.04E-01	6.41E-01	1.00	7.45E-01	1.85E-08	3.00E-06	6.16E-03	1.23E-02
Te-129m	1.92E-04	1.35E-03	1.00	1.54E-03	3.83E-11	7.00E-06	5.47E-06	1.09E-05
Te-129	9.97E-04	4.27E-02	1.00	4.37E-02	1.08E-09	4.00E-04	2.71E-06	5.42E-06
Te-131m	1.10E-03	9.53E-03	1.00	1.06E-02	2.64E-10	8.00E-06	3.29E-05	6.59E-05
Te-131	2.77E-04	5.73E-03	1.00	6.01E-03	1.49E-10	8.00E-05	1.86E-06	3.73E-06
Te-132	1.52E-03	1.15E-02	145.25	2.32E-01	5.76E-09	9.00E-06	6.40E-04	1.28E-03

Nuclide	LRW (Ci/year)	SGB Ci/year Scaled to 4.40 Ci	Des/EXP Ratio	Des (Ci/year)	Liquid (µCi/cc)	ECL (µCi/cc)	Single Unit Operation C/ECL	Dual Unit Operation C/ECL
Ba-140	1.40E-02	9.02E-02	0.31	9.45E-02	2.35E-09	8.00E-06	2.93E-04	5.86E-04
La-140	2.22E-02	1.63E-01	0.06	1.64E-01	4.08E-09	9.00E-06	4.53E-04	9.07E-04
Ce-141	4.65E-04	1.06E-03	1.00	1.52E-03	3.77E-11	3.00E-05	1.26E-06	2.52E-06
Ce-143	2.08E-03	1.76E-02	1.00	1.97E-02	4.89E-10	2.00E-05	2.44E-05	4.89E-05
Ce-144	9.34E-03	2.77E-02	0.08	2.85E-02	7.06E-10	3.00E-06	2.35E-04	4.71E-04
Pr-144	0	0	0.08	0	0	6.00E-04	0	0.00E+00
Np-239	1.88E-03	1.47E-02	1.00	1.66E-02	4.12E-10	2.00E-05	2.06E-05	4.12E-05
H-3	1252.80		1	1252.80	3.11E-05	1.00E-03	3.11E-02	6.22E-02
H-3(TPC)	9.80E+03		1.0	9.80E+03	2.43E-04	1.00E-03	2.43E-01	2.74E-01
Total							3.68E-01	7.36E-01
Total(TPC)							5.80E-01	9.48E-01

Note: Dual unit operations consider only Unit 1 with the TPC.

Table 8: Gaseous Release with Containment Purge

Nuclide	Exp. Rel. (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	2.58E+01	1.23E+01	3.17E+02	1.10E-10	1.00E-07	1.10E-03	2.19E-03
Kr-85	6.99E+02	3.31E+01	2.31E+04	7.99E-09	7.00E-07	1.14E-02	2.28E-02
Kr-87	1.62E+01	7.45E+00	1.21E+02	4.18E-11	2.00E-08	2.09E-03	4.18E-03
Kr-88	3.85E+01	1.23E+01	4.75E+02	1.64E-10	9.00E-09	1.82E-02	3.65E-02
Xe-131m	1.19E+03	2.91E+00	3.45E+03	1.19E-09	2.00E-06	5.97E-04	1.19E-03
Xe-133m	4.88E+01	4.32E+01	2.11E+03	7.29E-10	6.00E-07	1.21E-03	2.43E-03
Xe-133	3.20E+03	1.11E+02	3.55E+05	1.23E-07	5.00E-07	2.46E-01	4.91E-01
Xe-135m	8.52E+00	5.04E+00	4.29E+01	1.48E-11	4.00E-08	3.71E-04	7.42E-04
Xe-135	1.85E+02	6.97E+00	1.29E+03	4.46E-10	7.00E-08	6.38E-03	1.28E-02
Xe-138	7.66E+00	5.43E+00	4.16E+01	1.44E-11	2.00E-08	7.19E-04	1.44E-03
Br-84	5.07E-02	2.50E+00	1.27E-01	4.38E-14	8.00E-08	5.48E-07	1.10E-06
I-131	1.53E-01	5.24E+01	8.03E+00	2.77E-12	2.00E-10	1.39E-02	2.78E-02
I-132	6.75E-01	4.00E+00	2.70E+00	9.33E-13	2.00E-08	4.67E-05	9.34E-05
I-133	4.58E-01	2.69E+01	1.23E+01	4.25E-12	1.00E-09	4.25E-03	8.51E-03
I-134	1.08E+00	1.65E+00	1.78E+00	6.14E-13	6.00E-08	1.02E-05	2.05E-05
I-135	8.45E-01	7.91E+00	6.69E+00	2.31E-12	6.00E-09	3.85E-04	7.70E-04
Cs-134	2.27E-03	4.06E+01	9.20E-02	3.18E-14	2.00E-10	1.59E-04	3.18E-04
Cs-136	8.01E-05	1.65E+02	1.32E-02	4.57E-15	9.00E-10	5.08E-06	1.02E-05
Cs-137	3.48E-03	1.53E+02	5.33E-01	1.84E-13	2.00E-10	9.20E-04	1.84E-03
Cr-51	5.92E-04	2.90E-01	1.73E-04	5.96E-17	3.00E-08	1.99E-09	3.98E-09
Mn-54	4.31E-04	4.70E-01	2.03E-04	7.01E-17	1.00E-09	7.01E-08	1.40E-07
Fe-59	7.70E-05	3.48E+00	2.68E-04	9.27E-17	5.00E-10	1.85E-07	3.71E-07
Co-58	2.32E-02	5.37E+00	1.24E-01	4.30E-14	1.00E-09	4.30E-05	8.60E-05
Co-60	8.74E-03	1.38E+00	1.21E-02	4.17E-15	5.00E-11	8.33E-05	1.67E-04
Sr-89	2.98E-03	2.25E+01	6.69E-02	2.31E-14	1.00E-09	2.31E-05	4.63E-05
Sr-90	1.14E-03	1.35E+01	1.54E-02	5.33E-15	6.00E-12	8.88E-04	1.78E-03
Zr-95	1.00E-03	1.71E+00	1.71E-03	5.92E-16	4.00E-10	1.48E-06	2.96E-06
Nb-95	2.45E-03	2.34E+00	5.73E-03	1.98E-15	2.00E-09	9.90E-07	1.98E-06
Ba-140	4.00E-04	3.10E-01	1.26E-04	4.34E-17	2.00E-09	2.17E-08	4.34E-08
H-3	1.39E+02	1.00E+00	1.39E+02	4.80E-11	1.00E-07	4.81E-04	9.62E-04
H-3 (TPC)	1.09E+03	1.00E+00	1.09E+03	3.76E-10	1.00E-07	3.76E-03	4.24E-03
C-14	1.12E+01	1.00E+00	1.12E+01	3.87E-12	3.00E-09	1.29E-03	2.58E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.18E-11	1.00E-08	1.18E-03	2.35E-03
total						3.11E-01	6.23E-01
total (TPC)						3.15E-01	6.26E-01

Note: Dual unit operations consider only Unit 1 with the TPC.
Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

Table 9: Gaseous Release with Continuous Filtered Containment Vent

Nuclide	Exp. Rel. (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	9.48E+00	1.23E+01	1.16E+02	4.02E-11	1.00E-07	4.02E-04	8.05E-04
Kr-85	6.78E+02	3.31E+01	2.24E+04	7.75E-09	7.00E-07	1.11E-02	2.21E-02
Kr-87	5.81E+00	7.45E+00	4.33E+01	1.50E-11	2.00E-08	7.48E-04	1.50E-03
Kr-88	1.32E+01	1.23E+01	1.63E+02	5.63E-11	9.00E-09	6.25E-03	1.25E-02
Xe-131m	1.09E+03	2.91E+00	3.18E+03	1.10E-09	2.00E-06	5.49E-04	1.10E-03
Xe-133m	4.31E+01	4.32E+01	1.86E+03	6.44E-10	6.00E-07	1.07E-03	2.15E-03
Xe-133	2.90E+03	1.11E+02	3.22E+05	1.11E-07	5.00E-07	2.23E-01	4.45E-01
Xe-135m	4.68E+00	5.04E+00	2.36E+01	8.15E-12	4.00E-08	2.04E-04	4.08E-04
Xe-135	8.88E+01	6.97E+00	6.19E+02	2.14E-10	7.00E-08	3.06E-03	6.11E-03
Xe-138	4.34E+00	5.43E+00	2.36E+01	8.15E-12	2.00E-08	4.07E-04	8.15E-04
Br-84	5.07E-02	2.50E+00	1.27E-01	4.38E-14	8.00E-08	5.00E-07	1.00E-06
I-131	1.53E-01	5.24E+01	8.00E+00	2.77E-12	2.00E-10	1.38E-02	2.77E-02
I-132	6.73E-01	4.00E+00	2.69E+00	9.30E-13	2.00E-08	4.65E-05	9.30E-05
I-133	4.57E-01	2.69E+01	1.23E+01	4.24E-12	1.00E-09	4.24E-03	8.49E-03
I-134	1.07E+00	1.65E+00	1.77E+00	6.10E-13	6.00E-08	1.02E-05	2.04E-05
I-135	8.42E-01	7.91E+00	6.66E+00	2.30E-12	6.00E-09	3.84E-04	7.67E-04
Cs-134	2.27E-03	4.06E+01	9.20E-02	3.18E-14	2.00E-10	1.59E-04	3.18E-04
Cs-136	8.01E-05	1.65E+02	1.32E-02	4.57E-15	9.00E-10	5.10E-06	1.02E-05
Cs-137	3.48E-03	1.53E+02	5.33E-01	1.84E-13	2.00E-10	9.20E-04	1.84E-03
Cr-51	5.92E-04	2.90E-01	1.73E-04	5.96E-17	3.00E-08	0.00E+00	0.00E+00
Mn-54	4.31E-04	4.70E-01	2.03E-04	7.01E-17	1.00E-09	1.00E-07	2.00E-07
Fe-59	7.70E-05	3.48E+00	2.68E-04	9.27E-17	5.00E-10	2.00E-07	4.00E-07
Co-58	2.32E-02	5.37E+00	1.24E-01	4.30E-14	1.00E-09	4.30E-05	8.60E-05
Co-60	8.74E-03	1.38E+00	1.21E-02	4.17E-15	5.00E-11	8.33E-05	1.67E-04
Sr-89	2.98E-03	2.25E+01	6.69E-02	2.31E-14	1.00E-09	2.31E-05	4.62E-05
Sr-90	1.14E-03	1.35E+01	1.54E-02	5.33E-15	6.00E-12	8.88E-04	1.78E-03
Zr-95	1.00E-03	1.71E+00	1.71E-03	5.92E-16	4.00E-10	1.50E-06	3.00E-06
Nb-95	2.45E-03	2.34E+00	5.73E-03	1.98E-15	2.00E-09	1.00E-06	2.00E-06
Ba-140	4.00E-04	3.10E-01	1.26E-04	4.34E-17	2.00E-09	0.00E+00	0.00E+00
H-3	1.39E+02	1.00E+00	1.39E+02	4.80E-11	1.00E-07	4.81E-04	9.62E-04
H-3 (TPC)	1.09E+03	1.00E+00	1.09E+03	3.76E-10	1.00E-07	3.76E-03	4.24E-03
C-14	1.12E+01	1.00E+00	1.12E+01	3.87E-12	3.00E-09	1.29E-03	2.58E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.18E-11	1.00E-08	1.18E-03	2.35E-03
total						2.70E-01	5.40E-01
total (TPC)						2.73E-01	5.43E-01

Note: Dual unit operations consider only Unit 1 with the TPC.
Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

Real Time Performance Monitoring

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

Action level 1 is triggered when the net cumulative estimated TPBAR tritium exceeds 1.5 the TPC estimated value. Action level 2 is triggered when the net cumulative estimated TPBAR tritium exceeds triple the TPC estimated value. The Action level 1 value of 1.5 is approximately the 95% confidence level of the total uncertainty in the net estimated TPBAR tritium value. That is, if exceeded there is a 5% 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 Action Level Trigger follows:

$$AL_{Trigger} = (\text{Total RCS Inventory} - \text{non-TPBAR Sources}) / \text{TPC Estimated Value}$$

The use of the cycle specific TPC estimated value as the permeation performance metric compensates for RCS water balance (water makeup and letdown) and the cycle's reactor power history. The lower action level requires more frequent tritium system sampling to monitor, verify, track, and trend the tritium levels. In the unlikely event that the higher action level is exceeded, WBN 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 WBN Technical Instructions.

The WBN Unit 1 License Amendment 40 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 and without merit (Figure 6).

Tritium Impacts on Public Dose

Normal Operation

Using the revised realistic TPC source terms for 1,900 TPBARs, the offsite radiation incremental tritium doses calculated for releases of radionuclides in liquid and gaseous effluents during normal operation are summarized in Table 10. This table also lists WBN's regulatory established radioactive effluent guidelines and the estimated non-TPC values.

Table 10: Annual Projected Impact of TPC (1,900 TPBARs) on Effluent Dose to Maximally Exposed Members of the Public and Total Public Dose

	Non-TPC Realistic Dose	TPC Realistic Dose	Incremental Increase from TPC	NRC Annual Effluent Exposure Guideline
Annual Radioactive Gaseous Emissions				
Maximally Exposed Individual (mrem)	0.55	0.55	0	5.00 Whole Body
Maximally Exposed Individual (mrem)	8.8 (Bone) See Note 1	10.6 (Bone)	1.8	15.00 Organ
50-mile Population Dose (Rem)	7.01 (Bone)	10.7 (Bone)	3.69	NA
Annual Radioactive Liquid Emissions				
Maximally Exposed Individual (mrem)	0.40	0.43	0.03	3.00 Whole Body
Maximally Exposed Individual (mrem)	0.55 (Liver)	0.57 (Liver)	0.02	10.00 Organ
50-mile Population Dose (Rem)	3.6 (Thyroid)	6.7 (Thyroid)	3.1	NA

Note 1 - With the inclusion of C-14, as required by Revision 2 of RG 1.21, Bone became the critical organ.

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

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[33] (112 base plates and 384 thimble plugs) is 5,921 curies per cycle (180 day post irradiation decay) or an average of 3,527 curies per year when adjusted to reflect measured dose rate[34] for Base Plate with 24 Thimble Plugs following 113 day decay adjusted to 180 days (WBN Survey 010201 #2). This represented an increase from the current WBN Unit 1 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 Unit 1 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 assessed the environmental impact from the solid radioactive waste associated with the production of 2,304 TPBARs. The WBN revised license amendment establishes 1,792 as the maximum number of TPBARs per cycle.

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 no longer includes two TPBAR failures, so no additional feed and bleed operations are expected, and therefore, no additional resins are evaluated.

Thus, the tritium production solid radioactive waste environmental impact is bounded by the WBN Unit 1 License Amendment 40 impact assessment.

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 require approximately four additional fresh fuel bundles per cycle.

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

Tritium Impacts on Station Accident Analysis

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 category 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.

Radiological Consequences of Accidents

WBN Unit 1 License Amendment 40 assessed the station accident analyses affected by the production of 2,304 TPBARs. To appropriately account for the radiological consequences of the increased tritium in the TPC, TVA included calculated TEDE[35] and Federal Guidance Report Number 11[36] dose conversion values for thyroid in the accident analysis for informational purposes. TPBARs are designed to withstand the rigors associated with category I through IV events; therefore, no TPBAR failures are predicted to occur during the design-basis accidents except for the LBLOCA or the FHA.

Large Break Loss of Cooling Accident (LBLOCA)

WBN Unit 1 License Amendment 40 and the current licensing basis consider that all of the tritium content of 2,304 TPBARs is released to the containment atmosphere after an LBLOCA. This is based on a design inventory 1.2 gm of tritium/TPBAR and results in 2.68 x 10^7 Ci of tritium. The design inventory of tritium remains the same. Therefore, the current licensing basis LBLOCA radiological dose consequences analysis (Table 11) is bounding for the 1,792 TPBAR core.

Table 11: Dose Consequences from an LBLOCA

	Current Licensing Basis (CLB)		
Dose (rem)	Control Room (CR)	Exclusion Area Boundary (EAB)	Low Population Zone (LPZ)
Beta	8.97	1.29	2.57
TEDE	2.49	3.58	2.43

Fuel Handling Accident (FHA)

WBN Unit 1 License Amendment 40 and the current licensing basis consider that all of the tritium content of 24 TPBARs (84,490 Ci) is released into the surrounding water after an FHA. This is the maximum number of TPBARs that would be in one fuel assembly. Because the design inventory of the TPBARs and the maximum number of TPBARs in one fuel assembly remains the same, the source term released to the surrounding water is not affected. Since issuance of License Amendment 40, WBN Unit 1 has modified the licensing basis for an FHA. License Amendment 92 was approved on June 19, 2013 (ML13141A564) and implemented Alternative Source Term for the FHA. Included in this was also a change to the amount of tritium that was released to the environment. License Amendment 40 assumed a 100% release and License Amendment 92 assumed a 25% release. The amount of tritium assumed to go airborne after an FHA is not affected by any of the changes being requested. Therefore, the current licensing basis for the FHA remains bounding for the requested changes. Table 12 compares the dose consequences from License Amendment 40 and the current licensing basis.

Table 12: Dose Consequences from a FHA

	License Amendment 40			CLB		
Dose (rem)	CR*	EAB	LPZ	CR*	EAB	LPZ
Beta	4.281	1.377	0.3198	NA	NA	NA
TEDE	4.099	2.979	0.6921	2.869	2.834	0.79

* values represent the bounding case - FHA in the Auxiliary Building

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

These analyses do not assume any fuel damage. Therefore, the source term is the primary and secondary coolant activity. The primary and secondary coolant activities are in accordance with ANSI/ANS-18.1-1984, except for tritium. License Amendment 40 and the current licensing basis assume that the tritium concentration is based on the expected tritium concentration and two TPBAR failures. This source term was updated to reflect a new permeation rate of 10 Ci/TPBAR/year, which bounds the realistic permeation rate with 100% margin. It continues to retain conservatism regarding the inclusion of two failed TPBARs. This bounding approach to source term assumptions for accidents is consistent with accident evaluation methodology. The modified source term provides sufficient margin to bound AOO and PA cases.

WBN Unit 1 License Amendment 40 and the current licensing basis determined the expected tritium concentration assuming a permeation rate of 1 Ci/TPBAR/year, 2,304 total TPBARs, and two failed TPBARs. This results in a concentration of 98.4 µCi/gm. The assumed RCS tritium activity to support the requested changes is based on 2,500 TPBARs with an assumed permeation rate of 10 Ci/TPBAR/year and two failed TPBARs. This results in a concentration of 124.9 µCi/gm.

The radiological consequences of the MSLB and SGTR with a 2,500 TPBAR core remain well within 10 CFR Part 100 and General Design Criteria (GDC) 19 dose limits as shown in Table 13.

Table 13: Dose Consequences from MSLB and SGTR Accidents

Main Steam Line Break

Dose (rem)	CLB			New		
Pre-Accident Spike	CR	EAB	LPZ	CR	EAB	LPZ
Beta	6.30E-02	9.27E-03	4.34E-03	6.37E-02	9.28E-03	4.35E-03
TEDE	4.58E-01	1.92E-01	8.75E-02	4.66E-01	1.92E-01	8.76E-02
Accident Initiated Spike						
Beta	9.93E-02	2.55E-02	2.98E-02	9.98E-02	2.55E-02	2.98E-02
TEDE	6.29E-01	3.48E-01	4.69E-01	6.35E-01	3.49E-01	4.69E-01

Steam Generator Tube Rupture

	CLB			New		
Pre-Accident Spike	CR	EAB	LPZ	CR	EAB	LPZ
Beta	9.50E-01	2.03E-01	6.22E-02	9.62E-01	2.04E-01	6.25E-02
TEDE	1.15E+00	1.21E+00	3.49E-01	1.28E+00	1.22E+00	3.52E-01
Accident Initiated Spike						
Beta	9.34E-01	2.33E-01	7.15E-02	9.45E-01	2.35E-01	7.19E-02
TEDE	5.50E-01	1.06E+00	3.08E-01	6.50E-01	1.08E+00	3.14E-01

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[1] Watts Bar Nuclear Plant (WBN) - Unit 1 - Revision of Boron Concentration Limits and Reactor Core Limitations for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. TVA-WBN-TS-00-015, August 20, 2001 (ADAMS Accession No. ML012390106).

[2] Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 40 to Facility Operating License No. NPF-90 Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1 Docket No. 50-390, September 23, 2002 (ADAMS Accession No. ML022540925).

[3] TVA Letter to NRC dated March 22, 2005, "Watts Bar Nuclear Plant (WBN) Unit 1 - Tritium Production Program - Unit 1 Cycle 6 Operating Experience," (ADAMS Accession No. ML050870454).

[4] TVA Letter to NRC dated April 25, 2007, "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," (ADAMS Accession No. ML071210604).

[5] TVA Letter to NRC dated December 31, 2008, "Watts Bar Nuclear Plant (WBN) Unit 1 - Revised Technical Specifications Change WBN-TS-08-04 – Revision to the Maximum Number of TPBARs that Can Be Irradiated in the Reactor Core Per Cycle (TAC No. MD9396)," (ADAMS Accession No. ML090090044).

[6] TVA WBN Unit 1 Licensing Basis Post-LOCA subcriticality evaluation establishes a 1,792 TPBAR Core load as the maximum configuration. This evaluation uses a 1,900 TPBAR TPC Source Term to provide an additional margin when evaluating Realistic Effluent Releases and Design Basis Effluent Releases.

[7] This Design basis source term is used to assess station occupational exposure and radwaste system capability. It should not be confused with the UFSAR Chapter 15 Accident Design Basis source term used for offsite dose evaluations.

[8] TTP-1-3085, Revision 0, WBN-1 Cycle 12 TPBAR Tritium Release, Deduced From Analysis of RCS Data. July 18, 2014. Pacific Northwest National Laboratory, Richland, Washington

[9] TVA Letter to NRC dated April 25, 2007, "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. (ADAMS Accession No. ML071210604)

[10] Watts Bar Nuclear Plant, "Updated Final Safety Analysis Report (UFSAR)."

[11] WBNTSR-100, Revision 13 "Design Releases to Show Compliance with 10CFR20."

[12] *Health Effects, Dosimetry and Radiological Protection of Tritium.* Minister of Public Works and Government Services, Canada 2010. Catalogue number CC172-58/2010E-PDF ISBN 978-1-100-15583-8. Canadian Nuclear Safety Commission (CNSC) INFO-0799

[13] International Commission on Radiological Protection (ICRP), 1979-1982. Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Part 1 (and Supplement), Part 2 (and Supplement), Part 3 (and Supplements A and B), and Index, prepared by Committee 2, adopted by the Commission in July 1978, Annals of the ICRP, Pergamon Press, New York, N.Y

[14] ICRP, 1994b. Dose Coefficients for Intakes of Radionuclides by Workers. Publication 68, 24(4), Oxford, Pergamon Press.

[15] ICRP, 1993. Age-dependent doses to members of the public from intake of radionuclides: Part 2. ICRP Publication 67, Annals of the ICRP, 23(3/4), Oxford, Pergamon Press.

[16] ICRP, 1995a. Age-dependent Doses to Members of the Public from Intakes of Radionuclides: Part 4, Inhalation Dose Coefficients. Publication 71, 25(3-4) Oxford, Pergamon Press.

[17] EPA-520/1-86-020, Federal Guidance Report No. 11 1988 *Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhalation, Submersion, And Ingestion.* Washington, D.C.

[18] 10 CFR Part 20 Final Rule 56 FR 23391, May 21, 1991

[19] Westinghouse Electric Company, October 2000, Evaluation of Waste Management Issues for Operation with a Tritium Production Core (TPC).

[20] NUREG-800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, dated June 1987 and latest revision, by U.S. NRC.

[21] Regulatory Guide 1.206. U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research Division 1, June 2007

[22] This Design Basis source term is used to assess station occupational exposure and radwaste system capability. It should not be confused with the UFSAR Chapter 15 Accident Design Basis source term used for offsite dose evaluations.

[23] Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 40 to Facility Operating License No. NPF-90 Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1 Docket No. 50-390, September 23, 2002 (ADAMS Accession No. ML022540925).

[24] NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors"

[25] TTQP-1-015, Revision 19, “Description of the Tritium-Producing Burnable Absorber Rod for the Commercial Light Water Reactor. Pacific Northwest National Laboratory, Richland, Washington.

[26] TTP-1-3016, Revision 1, MARK 9.2 (FLG Design) TPBAR Tritium Release, Deduced from Analysis and Qualification of WBN Cycle 9 RCS Data. Pacific Northwest National Laboratory, Richland, Washington.

[27] TTP-1-3085, Revision 0, WBN-1 Cycle12 TPBAR Tritium Release, Deduced From Analysis Of RCS Data. Pacific Northwest National Laboratory, Richland, Washington.

[29] Operational experience, cycles 8 – 10, for all related activities (pre-work, TPBAR Handling fixture setup, Consolidation, fixture storage, production and Post Irradiation Examination shipping, waste hub disposal, cleanup and post-work activities) averaged 0.46 mrem/TPBAR. Rounded upward to 1 mrem/TPBAR to handle contingencies.

[30] NUREG-0713, Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 2012, Vol. 34, U.S. Nuclear Regulatory Commission, April 2014.

[31] Watts Bar Nuclear Plant, “Updated Final Safety Analysis Report (UFSAR).”

[32] WBNTSR-100, Revision 12, “Design Releases to Show Compliance with 10CFR20”. July 16, 2013

[33] Pacific Northwest National Laboratory, 1999, *Unclassified Bounding Source Term, Radionuclide Concentrations, Decay Heat, and Dose Rates for the Production TPBAR, TTQP-1-111, Revision 1.*

[34] BP-263, “*Low Level Radioactive Waste Liability Accrual*”

[35] 10 CFR Appendix B to Part 20--Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

[36] EPA-520/1-86-020, Federal Guidance Report No. 11 Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. Washington, D.C.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1

WBNNAL3003, Revision 6, "Reactor Coolant and Secondary Side Activities In Accordance with ANSIIANS-18.1-1984"

(26 pages including this cover)

TVA NUCLEAR CALCULATION COVERSHEET / CTS UPDATE

REV 0 EDMS/RIMS NO.	CTS TYPE:	EDMS TYPE:	EDMS ACCESSION NO (N/A for REV. 0)
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	ORG	PLANT	BRANCH	NUMBER	CUR REV	NEW REV
CALC ID	NUC	WBN	NTB	WBNNAL3003	005	006

CTS UPDATE ONLY ☐ (Verifier and Approval Signatures Not Required)

NO CTS CHANGES ☐ (For calc revision, CTS has been reviewed and no CTS changes required)

UNIT (check one)	SYSTEMS	UNIDS
0 ☒, 1 ☐, 2 ☐, 3 ☐	N/A	N/A

ECP. N/A	APPLICABLE DESIGN DOCUMENT(S)	CLASSIFICATION
DCN 61599	N/A	E

QUALITY RELATED?	SAFETY RELATED? (If yes, QR = yes)	UNVERIFIED ASSUMPTION	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?	DESIGN OUTPUT ATTACHMENT?	SAR/TS and/or ISFSI SAR/CoC AFFECTED
Yes ☒ No ☐	Yes ☒ No ☐	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒

CALCULATION NUMBER REQUESTOR	PREPARING DISCIPLINE	VERIFICATION METHOD	NEW METHOD OF ANALYSIS
Name: PHONE:	N	Design Review	☐ Yes ☒ No

PREPARER (PRINT NAME AND SIGN)	DATE	CHECKER (PRINT NAME AND SIGN)	DATE
Aleksandar Milicevic [signature]	3/7/16	Mehran A. Mohammadian [signature]	3/7/16
VERIFIER (PRINT NAME AND SIGN)	**DATE**	**APPROVAL (PRINT NAME AND SIGN)**	**DATE**
Mehran A. Mohammadian [signature]	3/7/16	James V Ware [signature] AAK 3/7/16	3/14/16

STATEMENT OF PROBLEM/ABSTRACT

The methodology of ANSI/ANS-18.1-1984 was followed to calculate reactor coolant activities except for Tritium Production Core (TPC) tritium values which are discussed in the calculation. The calculations begin with the use of the base activities in Table 6 of ANSI/ANS-18.1-1984. Each base nuclide activity (in microcuries per gram) is multiplied by an adjustment factor. The referenced standard provides the method for using plant-specific parameters to determine the adjustment factors.

The expected reactor coolant activities that are calculated are intended for possible use in environmental reports, in normal dose calculations for equipment qualification purposes, and for other applications where the use of expected average data over the life of the plant would be appropriate.

MICROFICHE/EFICHE Yes ☐ No ☒ FICHE NUMBER(S)

NPG CALCULATION COVERSHEET / CTS UPDATE

CALC ID	ORG	PLANT	BRANCH	NUMBER	REV
	NUC	WBN	NTB	WBNNAL3003	006

BUILDING	ROOM	ELEVATION	COORD/AZIM	FIRM
NA	N/A	N/A	N/A	S&L

CATEGORIES N/A

KEYWORDS (A-add, D-delete)

ACTION (A/D)	KEYWORD	A/D	KEYWORD

CROSS-REFERENCES (A-add, D-delete)

ACTION (A/D)	XREF CODE	XREF PLANT	XREF TYPE	XREF NUMBER	XREF REV

CTS ONLY UPDATES:

Following are required only when making keyword/cross reference CTS updates and page 1 of form NEDP-2-1 is not included:

PREPARER (PRINT NAME AND SIGN)	DATE	CHECKER (PRINT NAME AND SIGN)	DATE
PREPARER PHONE NO.	EDMS ACCESSION NO.		

TVA NUCLEAR CALCULATION RECORD OF REVISION

CALCULATION IDENTIFIER WBNNAL3003

Title **Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984**

Revision No.	DESCRIPTION OF REVISION
0	Initial Issue
1	Revision 1 was performed to add secondary side activities (water and steam) to the analysis. R0 reactor coolant activities were not changed. All pages were rewritten for legibility and to bring the calculation into conformance with NEP 3.1 R2 Pages Changed : all
2	Revision 2 was performed to incorporate EDC E50629A which addresses the use of a Tritium Production Core. Only the amount of tritium produced in the core is changed. Since operations will not change all other isotopes remain the same. Updated file to new format. Revision bars will indicate changes from last revision. Classification forms were deleted as they are no longer needed. Pages changed : 1-18 Pages added :1, Computer file storage (2), Computer microfiche sheet (3) Pages deleted : classification forms (2&3) R2: total 43 pages
3	Revision 3 is in addition to R2, it includes the possibility of 1 or 2 rod failure in a TPC . The only change is to the tritium concentration in the event of these failures. It also changes the expected concentration of H3 with a TPC from a maximum value to an average (9 to 3.7μCi/g). All margins in the body of the calculation were changed, so all pages are new, but actual text changes are marked by revision bars. Pages changed:1c,2,3,7-8,11,16-18 Pages added: 1 cover sheet Pages deleted : none R3 : total 44 pages
4	Revision 4 is in support of DCN D51754, Steam Generator Replacement. With the new steam generators, the steam and water masses changed as well as the steam flow rate. These changes are incorporated in Appendix A and the results are shown to be less than that determined in R3. Therefore the user can choose which set of concentrations to use. The values in Appendix A will not be valid until DCN 51754 is in RTO status and the plant is out of the refueling outage it was installed. The current results are conservative for dose calculations, but are non-conservative for leak detection (WBNAPS3-052, -053) and minimum concentration calculations (WBNAPS3048). The FSAR and Technical Specifications impacts, if any, are addressed in the screening review for DCN 51754. Pages Replaced or Revised:1, 3, 4, 7-9, 16, 17 Pages Added: CCRIS Update Sheet (2), Appendix A (4 pages) Pages Deleted: Design Verification Form, Computer Output Form Total R4 pages - 44

TVA NUCLEAR CALCULATION RECORD OF REVISION

CALCULATION IDENTIFIER WBNNAL3003

Title **Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984**

Revision No.	DESCRIPTION OF REVISION
5	Revision 5 is performed in support of increasing the TPC tritium permeation rate to reflect a realistic and design basis source term. The realistic source term is based on a permeation rate of 5 Ci/TPBAR/yr and is calculated for 704 TPBARs and 1900 TPBARs. The design basis source term is based on a permeation rate of 10 Ci/TPBAR/yr and is calculated for 2500 TPBARs. This change impacts the TPC average tritium concentrations as well as the 1 and 2 TPBAR failure concentrations. CTS was reviewed for successor documents, and multiple successor documents were identified. The following documents are impacted by this revision: WBN Calculations TI534, TI535, WBNAPS3044, WBNAPS3077, WBNAPS3118, WBNTSR008, WBNTSR064, WBNTSR068, WBNTSR080, WBNTSR084, WBNTSR088, WBNTSR093, and WBNTSR100. There are a significant number of additional successor calculations. However, those calculations do not use the TPC values for tritium, therefore they are not impacted by this revision. See DCN 61599 for SAR/Tech Spec impact determination. Pages Replaced or Revised: 1-5, 7-9, 13, 18, 19, 21, 22, 24 Pages Added: NPG Calculation Verification Form (page 6) Pages Deleted: none Total Revision 5 pages: 48 The revision of all the pages has been changed to revision 5 and changes to the calculation are indicated by change bars.
6	The original design basis Tritium Production Core (TPC) tritium source terms were too conservative and resulted in a concern of overloading the Tritiated Water Storage Tank (TWST) storage capability over the course of multiple years. Revision 6 is performed in support of removing the original design basis TPC tritium source terms from the calculation, which were based on a permeation rate of 10 Ci/TPBAR/yr and were calculated for 2500 TPBARs. The realistic source terms, which are based on a permeation rate of 5 Ci/TPBAR/yr and are calculated for 704 TPBARs and 1900 TPBARs, are unchanged and maintained in the calculation. This change impacts the number of TPC average tritium concentration cases presented in this calculation. Per TVA review of CTS for successor documents, the only document impacted by this revision is WBNTSR100. Other successor documents are either not impacted or are conservative as-is. See DCN 61599 for SAR/Tech Spec impact determination. Pages Replaced or Revised: 1-9, 13, 19, 21, and 23 Pages Added: None Pages Deleted: Pages 19-20 from Rev 5 were combined into one page (page 19) in Rev. 6 Total Revision 6 pages: 47 The revision of all the pages has been changed to revision 6 and changes to the calculation are indicated by change bars.

TVA NUCLEAR CALCULATION TABLE OF CONTENTS

Calculation Identifier: WBNNAL3003	Revision: 006

TABLE OF CONTENTS

Note: Attachment B has been removed from this document

TVA NUCLEAR CALCULATION VERIFICATION FORM

Calculation Identifier WBNNAL3003 Revision 006

Method of verification used:		
1. Design Review	☒	Verifier [signature] Date 3/7/16
2. Alternate Calculation	☐	Mehran A. Mohammadian
3. Qualification Test	☐	

Comments:

I have reviewed Calculation WBNNAL3003 Revision 6 and have found the portions of the calculation revised to be technically adequate. In conducting the verification of the portions of WBNNAL3003 revised as part of Revision 6, I have reviewed the inputs, computations, and results, which I have found to be complete and accurate. All comments have been resolved with the preparer.

TVA NUCLEAR COMPUTER INPUT FILE
STORAGE INFORMATION SHEET

Document WBNNAL3003	Rev. 006	Plant: WBN	

Subject:
Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984

☐ Electronic storage of the input files for this calculation is not required. Comments:

☒ Input files for this calculation have been stored electronically and sufficient identifying information is provided below for each input file. (Any retrieved file requires re-verification of its contents before use.)

The Microsoft Word file for R5 is permanently stored in FILEKEEPER # 321761.

(file *Calculation WBNNAL3003, Rev. 5.doc*)

The Microsoft Word file for R6 is permanently stored in FILEKEEPER # 338106.

(file *WBNNAL3003R6.doc*)

☐ Microfiche/eFiche

Purpose

The purpose of this calculation is to calculate the expected primary coolant activities in accordance with ANSI/ANS-18.1-1984 methodology, except for the Tritium Production Core (TPC) tritium values which are discussed later. Revision 1 of this calculation added the expected secondary side (water and steam) activities to the analysis.

Introduction

ANSI/ANS-18.1-1984 (ref.2) (Attachment B) provides numerical values of coolant and steam activity based on available data from operating plants. For pressurized water reactors with U-tube steam generators, such as WBN, the numerical values are given in Table 6 of reference 2. These activities may be used without correction only if all plant parameters in Table 2 of reference 2 that affect primary coolant and steam activity have the same parameter nominal values presented in Table 2 of reference 2. Since some applicable parameter values for WBN are not identical to the nominal values in Table 2 of reference 2, adjustment factors need to be calculated. The adjustment factors are calculated and applied to the base activities in Table 6 of reference 2 to obtain the normal coolant and secondary side (water and steam) activities for WBN.

Assumptions

1. Several isotopes listed in ANS-18.1 ANSI N237-1976 are not present in ANSI/ANS-18.1-1984. The isotopes deleted were Kr-83m, Br-83, Br-85, Rb-86, Te-125m, Te-127m, Te-127, and I-130. These isotopes will not be in the final list of this calculation.
   Technical Justification: These isotopes were dropped from the list because they were deemed to be not as important as the listed isotopes. This was justified by one of the authors of ANSI/ANS-18.1-1984 (see Attachment A for detailed justification).
2. Other isotopes were left off the ANSI/ANS-18.1-1984 listing because they are in secular equilibrium with the parent isotope. These isotopes are Y-90, Rh-103m, Rh-106, Ba-137m, and Pr-144. This report will include these isotopes in the final listing, however they will be marked as being in secular equilibrium with its parent.
   Technical Justification: As in Assumption #1, the authors of ANSI/ANS-18.1-1984 deemed these isotopes to be insignificant (see Attachment A). For conservatism and completeness, these isotopes will be included.
3. Pr-143 was left off the ANSI/ANS-18.1-1984 list because it is a pure beta emitter (see Attachment A). This will be included in the final listing of this report, with the appropriate notation. It is assumed that Pr-143 is in secular equilibrium with its parent Ce-143.
   Technical Justification: This isotope is included for completeness. The assumption that it is in secular equilibrium (the activity is the same as the parent) is not true because Pr-143 has a longer half life (13.58 days, ref.5) than Ce-143 (33 hours, ref.5). However, this assumption is deemed conservative because it is in addition to those isotopes given in the ANSI/ANS-18.1-1984 standard.
4. A peak concentration of 2.5 μCi/g of tritium is assumed before dilution occurs in a normal, non-TPC cycle.
   Technical justification: Operational plant data for WBN and SQN show a peak of 2.5 μCi/g before dilution occurs in a normal cycle, i.e. no scrams or interruptions (ref.13).
5. A TPC will not change the normal operations of the plant, e.g no extra dilution, no recycle of primary coolant, and/or other changes that would impact coolant tritium concentrations, and thus the concentration of all other isotopes except tritium will remain the same.
   Technical Justification: Tritium has a long half life and does not produce any daughters, therefore all other isotopes are not impacted by the increase in tritium.
6. A TPC tritium permeation rate of 5 Ci/TPBAR/yr is assumed.
   Technical justification: The permeation rate of 5 Ci/TPBAR/yr is acceptable because it bounds the observed permeation rate.

Special Requirements/Limiting Conditions

none

Calculations

Plant specific parameters used to obtain adjustment factors are shown in the following table. The adjustment factors are calculated for all ref.2 Table 6 primary coolant and secondary side activities in classes 1, 2, 3, and 6. Note that the corrosion product activities are included in the Table 6 class 6 list. Calculation of the adjustment factors follows the procedures given in ref.2.

Operation with a TPC has a potential to increase the amounts of tritium in the reactor coolant. To address the TPC impacts, the methodology in ref.15 is used in establishing an average tritium concentration in the reactor coolant. The average tritium concentration is 5.1 μCi/g for 704 TPBARs and 12.0 μCi/g for 1900 TPBARs. This is roughly based on WBN having an average tritium concentration during normal operation of 1.0 μCi/g and projecting an approximate 5.1 and 12.0 times increase above the tritium production in a non-TPC core. This is based on assumption 5 that says the plant will operate the same as for a non-TPC, i.e. no extra dilution (feed and bleed) or other changes.

PARAMETERS USED TO DESCRIBE THE REACTOR SYSTEM REALISTIC BASIS
(values in parentheses are those given for the Replacement Steam Generators)

	Symbol	Units	Nominal ANS-18.1-1984 1984 Value	WBN Value	Reference (note)
Thermal power	P	MWt	3400	3582	3,4 (a)
Steam flow rate	FS	lb/hr	1.5E+07	1.5E+07 (1.54E+07)	3,6 16, (j)
Weight of water in reactor coolant system	WP	lb	5.5E+05	5.4E+05 (5.78E+05)	3,10 (b) 16, (j)
Weight of water in all steam generators	WS	lb	4.50E+05	3.48E+05 (6.79E+05)	3,9 (c) 16, (j)
Reactor coolant letdown flow rate (purification)	FD	lb/hr	3.7E+04	3.7E+04	3,8 (d)
Reactor coolant letdown flow rate (yearly average for boron control)	FB	lb/hr	500	845	3
Steam Generator Blowdown flow (total)	FBD	lb/hr	7.50E+04	3.00E+04	3,9 (e)
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	NBD	-	1.0	1.0	(f)
Flow through the purification system cation demineralizer	FA	lb/hr	3.7E+03	3.7E+03	3,8 (g)

PARAMETERS USED TO DESCRIBE THE REACTOR SYSTEM REALISTIC BASIS Cont'd

	Symbol	Units	Nominal ANS-18.1-1984 1984 Value	WBN Value	Reference (note)
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	-	0.0	0.55	3,9 (h)
Fraction of the noble gas activity in the letdown stream which is not returned to the reactor coolant system (not including the boron recovery system)	Y	-	0.0	0.0	3(i)

Notes:

a) The thermal power is 3411 MWt (ref. 4) . The value used in this calculation is 105% of 3411 = 3582 MWt.

b) Reference 10 gives the reactor coolant volume as 11375 cuft. Reference 4 gives the reactor inlet temperature as 558.1 degF and the outlet temperature as 618.2 degF. From ref.11, the specific volume v_f for compressed liquid at 500 degF is 0.0204 cuft/lb, for 600 degF v_f is 0.0236 cuft/lb, and for 620 degF v-f is 0.0247. Using linear interpolation, v_f for 558.1 degF is then 0.0223 cuft/lb and v_f for 618.2 degF is 0.0246 cuft/lb. These then give an estimate of the mass of the reactor coolant between 11375 cuft/0.0204 cuft/lb = 5.576E5 lb and 11375 cuft/0.0246 cuft/lb = 4.624E5 lb. The use of 5.4E5 lb is therefore appropriate. Specific volume for the RSG was based on the reactor vessel inlet temperature of 557.3 (ref. 16) and was calculated to be 0.021986 ft^3/lb.

c) Reference 9 gives the weight of water in a steam generator as 4.745E7g which translates to 4*4.745E7 g /453.59 g/lb = 4.184E5 lb. The use of 3.48E5 will result in higher concentrations and can therefore be used.

d) Reference 8 gives the maximum letdown as 120 gpm = 120 gal/min * 60 min/hr * 62.4 lb/cuft /7.48 gal/cuft = 6E4 lb/hr. The use of 3.7E4lb/hr is therefore an average flow which will result in larger radioisotope concentrations. Note that 3.7E4 is the design flow through the cation demineralizers (see note g below).

e) The blowdown is given as 28,900 lb/hr ref.9. The use of 30,000lb/hr is a rounded off value and is not inappropriate.

f) This means that all radioisotopes are removed. This is not a bad assumption because virtually all noble gasses will be removed, and reference 12 gives the condensate demineralizers removal efficiencies of a factor of 10. A NBD factor of 0.9 could be used, but to be consistent with previous calculations, the factor will remain at 1.0.

g) The design flow through the cation demineralizers is 75 gpm = 75 gal/min * 60 min/hr * 62.4 lb/cuft /7.48 gal/cuft = 3.75E4 lb/hr (ref. 8). The use of 3.7E4 lb/hr will increase the radioisotope inventory and is thus conservative to use.

h) Reference 9 gives a value for NC as 0.589. 0.55 is the minimum value from reference 3.

i) The use of 0.0 will maximize the noble gas inventories by not stripping any noble gasses from the reactor coolant. Note that the Y values given in Table 11.6 of ref. 3 are not used in the ANS-18.1-1984 corrections methodology.

j) These values are those for the Replacement Steam Generators. They result in lower concentrations and thus the original values are conservative. See Appendix A for the results using these values.

Equations used: (ref.2)

Lambda=ln(2)/(Half Life)

$$R_1=\frac{FB+(FD-FB)Y}{WP}$$

$$R_{2,3,6}=\frac{(FD)(NB_{2,3,6})+(1-NB_{2,3,6})[FB+FA(NA_{2,3,6})]}{WP}$$

R_{n1}=9.0E-4 (this value comes directly from ANSI/ANS-18.1-1984)

R_{n2}=6.7E-2 (this value comes directly from ANSI/ANS-18.1-1984)

R_{n3}=3.7E-2 (this value comes directly from ANSI/ANS-18.1-1984)

R_{n4}=0 (this value comes directly from ANSI/ANS-18.1-1984)

R_{n6}=6.6E-2 (this value comes directly from ANSI/ANS-18.1-1984)

NB_2=.99 (this value comes directly from ANSI/ANS-18.1-1984)

NB_3=0.5 (this value comes directly from ANSI/ANS-18.1-1984)

NB_6=0.98 (this value comes directly from ANSI/ANS-18.1-1984)

NA_2=0 (this value comes directly from ANSI/ANS-18.1-1984)

NA_3=0.9 (this value comes directly from ANSI/ANS-18.1-1984)

NA_6=0.9 (this value comes directly from ANSI/ANS-18.1-1984)

$$r_{2,3,6}=\frac{(FBD)(NBD)+(NS)(FS)(NC)(NX)}{WS}$$

NS_2=0.01 (this value comes directly from ANSI/ANS-18.1-1984)

$NS_{3,6}$=5E-3 (this value comes directly from ANSI/ANS-18.1-1984)

NS_5=1.0 (this value comes directly from ANSI/ANS-18.1-1984)

$NX_{1,4,5}$=0 (this value comes directly from ANSI/ANS-18.1-1984)

$NX_{2,6}$=0.9 (this value comes directly from ANSI/ANS-18.1-1984)

NX_3=0.5 (this value comes directly from ANSI/ANS-18.1-1984)

$f_{1,2,3,6} = \frac{P(WP_n)(R_{n1,n2,n3,n6}+LAMBDA)}{WP(P_n)(R_{1,2,3,6}+LAMBDA)}$; $f_{4,5}=1.0$; $f'_{2,3,6} = \frac{(WS_n)(r_{n2,n3,n6}+LAMBDA)(f_{2,3,6})}{(WS)(r_{2,3,6}+LAMBDA)}$

$r_{n2,n6}=0.17$ (this value comes directly from ANSI/ANS-18.1-1984)

$r_{n3}=0.15$ (this value comes directly from ANSI/ANS-18.1-1984)

$f'_1=0.0$ (negligible noble gasses in secondary water)

$f'_4 = \frac{WS_n}{WS}$; $f'_5=1.0$; $f''_{2,3,6} = \frac{(WS_n)(r_{n2,n3,n6}+LAMBDA)(f_{2,3,6})}{(WS)(r_{2,3,6}+LAMBDA)}$

$f''_1 = \frac{(FS_n)(f_1)}{FS}$; $f''_4 = \frac{WS_n}{WS}$; $f''_5=1.0$; $A=f*A_n$; $A'=f'*A'_n$; $A''=f''*A''_n$

where subscript n refers to the nominal value of the variable
subscript (number) refers to class number
Lambda=nuclide decay constant hr^{-1}
R=removal rate (reactor coolant) hr^{-1}
r=removal rate (steam/secondary water) hr^{-1}
f=adjustment factor, reactor coolant
f'=adjustment factor, secondary water
f"=adjustment factor, secondary steam
NA=fraction of material removed by the cation demineralizer
NB=fraction of material removed by the purification demineralizer
NS=Ratio of concentration in steam to that of the steam generator
NX=Fraction of activity removed by the condensate demineralizers
A=activity uCi/g, reactor coolant
A'=activity uCi/g, secondary water
A"=activity uCi/g, secondary steam

TPC Average Concentration and TPBAR Failure: methodology and values from ref. 15

Tritium Sources = TPBAR releases + IFBA releases + Non-TPC sources

TPBAR releases = #TPBARs * 5 Ci/TPBAR/yr
IFBA releases = 40 Ci/yr
Non-TPC sources = 870 Ci/yr

TPC coolant average concentration is the non-TPC average concentration (1 μCi/g) multiplied by the ratio of the tritium sources for the TPC core to the normal tritium sources

C_{TPC} = (1 μCi/g)(#TPBARs * 5 + 40 + 870)/(870)
(1 μCi/g)(704 * 5 + 40 + 870)/870 = 5.1 μCi/g for 704 TPBARs
(1 μCi/g)(1900 * 5 + 40 + 870)/870 = 12.0 μCi/g for 1900 TPBARs

Increase in coolant average concentrations due to TPBAR failure:

mass of RCS = 5.4E5 lb * 453.9 g/lb = 2.45E8 g
I_{TPBARs} = 11,600 Ci
I_{RCS} = RCS inventory prior to failure = average concentration x RCS mass
5.1 μCi/g x 2.45E8 g = 1249.5 Ci for 704 TPBARs
12.0 μCi/g x 2.45E8 g = 2940.0 Ci for 1900 TPBARs

1 TPBAR Failure

C_{new} = (11,600 + 1249.5) Ci/ 2.45E8 g = 52.45 μCi/g for 704 TPBARs
C_{new} = (11,600 + 2940.0) Ci/ 2.45E8 g = 59.35 μCi/g for 1900 TPBARs

2 TPBAR Failure

C_{new} = (2*11,600 + 1249.5) Ci/ 2.45E8 g = 99.79 μCi/g for 704 TPBARs
C_{new} = (2*11,600 + 2940.0) Ci/ 2.45E8 g = 106.69 μCi/g for 1900 TPBARs

where I is the inventory, and C is the average concentration

REACTOR COOLANT ACTIVITIES

Nuclide	half life ref.5	Lambda 1/hr	removal rate R	adjust. factor f	nominal reactor coolant ANSI-18.1 uCi/gm	calculated reactor coolant WBN uCi/gm
<u>Class 1</u>						
Kr-85m	4.48h	1.55E-01	1.56E-03	1.07E+00	1.6E-01	1.71E-01
Kr-85	10.7y	7.39E-06	1.56E-03	6.19E-01	4.3E-01	2.66E-01
Kr-87	76m	5.47E-01	1.56E-03	1.07E+00	1.5E-01	1.61E-01
Kr-88	2.84h	2.44E-01	1.56E-03	1.07E+00	2.8E-01	3.00E-01
Xe-131m	11.77d	2.45E-03	1.56E-03	8.96E-01	7.3E-01	6.54E-01
Xe-133m	2.19d	1.32E-02	1.56E-03	1.02E+00	7.0E-02	7.17E-02
Xe-133	5.25d	5.50E-03	1.56E-03	9.72E-01	2.6E+00	2.53E+00
Xe-135m	15.6m	2.67E+00	1.56E-03	1.07E+00	1.3E-01	1.39E-01
Xe-135	9.1h	7.62E-02	1.56E-03	1.06E+00	8.5E-01	9.04E-01
Xe-137	3.82m	1.09E+01	1.56E-03	1.07E+00	3.4E-02	3.65E-02
Xe-138	14.1m	2.95E+00	1.56E-03	1.07E+00	1.2E-01	1.29E-01
<u>Class 2</u>						
Br-84	31.8m	1.31E+00	6.78E-02	1.07E+00	1.6E-02	1.72E-02
I-131	8.04d	3.59E-03	6.78E-02	1.06E+00	4.5E-02	4.77E-02
I-132	2.28h	3.04E-01	6.78E-02	1.07E+00	2.1E-01	2.25E-01
I-133	20.9h	3.32E-02	6.78E-02	1.06E+00	1.4E-01	1.49E-01
I-134	52.6m	7.91E-01	6.78E-02	1.07E+00	3.4E-01	3.64E-01
I-135	6.61h	1.05E-01	6.78E-02	1.07E+00	2.6E-01	2.78E-01
<u>Class 3</u>						
Rb-88	17.8m	2.34E+00	3.81E-02	1.07E+00	1.9E-01	2.04E-01
Cs-134	2.062y	3.84E-05	3.81E-02	1.04E+00	7.1E-03	7.39E-03
Cs-136	13.1d	2.20E-03	3.81E-02	1.04E+00	8.7E-04	9.08E-04
Cs-137	30.17y	2.62E-06	3.81E-02	1.04E+00	9.4E-03	9.79E-03
<u>Class 4</u>						
N-16	7.13s	3.50E+02	-	1.00E+00	4.0E+01	4.00E+01

Note: N-16 will be 0.0 uCi/g outside the shield building, and up to 40 uCi/gm inside. The exact concentration will be highly dependent on how long the water has been outside the core (due to the very short half life).

REACTOR COOLANT ACTIVITIES (continued)

Nuclide	half life ref.5	Lambda 1/hr	removal rate R	adjust. factor f	nominal reactor coolant ANSI-18.1 uCi/gm	calculated reactor coolant WBN uCi/gm
<u>Class 5</u>						
H-3	12.33y	6.42E-06	-	1.00E+00	1.0E+00	1.00E+00
<u>Class 6</u>						
Na-24	15.02h	4.61E-02	6.73E-02	1.06E+00	4.7E-02	4.99E-02
Cr-51	27.7d	1.04E-03	6.73E-02	1.05E+00	3.1E-03	3.26E-03
Mn-54	312d	9.26E-05	6.73E-02	1.05E+00	1.6E-03	1.68E-03
Fe-55	2.7y	2.93E-05	6.73E-02	1.05E+00	1.2E-03	1.26E-03
Fe-59	44.6d	6.48E-04	6.73E-02	1.05E+00	3.0E-04	3.16E-04
Co-58	70.8d	4.08E-04	6.73E-02	1.05E+00	4.6E-03	4.84E-03
Co-60	5.271y	1.50E-05	6.73E-02	1.05E+00	5.3E-04	5.58E-04
Zn-65	244.1d	1.18E-04	6.73E-02	1.05E+00	5.1E-04	5.37E-04
Sr-89	50.5d	5.72E-04	6.73E-02	1.05E+00	1.4E-04	1.47E-04
Sr-90	28.8y	2.75E-06	6.73E-02	1.05E+00	1.2E-05	1.26E-05
Sr-91	9.5h	7.30E-02	6.73E-02	1.06E+00	9.6E-04	1.02E-03
Y-91m	49.7m	8.37E-01	6.73E-02	1.07E+00	4.6E-04	4.93E-04
Y-91	58.5d	4.94E-04	6.73E-02	1.05E+00	5.2E-06	5.47E-06
Y-93	10.2h	6.80E-02	6.73E-02	1.06E+00	4.2E-03	4.46E-03
Zr-95	64d	4.51E-04	6.73E-02	1.05E+00	3.9E-04	4.10E-04
Nb-95	35d	8.25E-04	6.73E-02	1.05E+00	2.8E-04	2.95E-04
Mo-99	66.02h	1.05E-02	6.73E-02	1.06E+00	6.4E-03	6.75E-03
Tc-99m	6.02h	1.15E-01	6.73E-02	1.07E+00	4.7E-03	5.01E-03
Ru-103	39.4d	7.33E-04	6.73E-02	1.05E+00	7.5E-03	7.89E-03
Ru-106	367d	7.87E-05	6.73E-02	1.05E+00	9.0E-02	9.47E-02
Ag-110m	252d	1.15E-04	6.73E-02	1.05E+00	1.3E-03	1.37E-03
Te-129m	33.5d	8.62E-04	6.73E-02	1.05E+00	1.9E-04	2.00E-04
Te-129	67m	6.21E-01	6.73E-02	1.07E+00	2.4E-02	2.57E-02
Te-131m	30h	2.31E-02	6.73E-02	1.06E+00	1.5E-03	1.59E-03
Te-131	25m	1.66E+00	6.73E-02	1.07E+00	7.7E-03	8.26E-03
Te-132	78h	8.89E-03	6.73E-02	1.05E+00	1.7E-03	1.79E-03
Ba-140	12.79d	2.26E-03	6.73E-02	1.05E+00	1.3E-02	1.37E-02
La-140	40.3h	1.72E-02	6.73E-02	1.06E+00	2.5E-02	2.64E-02
Ce-141	32.5d	8.89E-04	6.73E-02	1.05E+00	1.5E-04	1.58E-04
Ce-143	33h	2.10E-02	6.73E-02	1.06E+00	2.8E-03	2.96E-03
Ce-144	284d	1.02E-04	6.73E-02	1.05E+00	4.0E-03	4.21E-03
W-187	23.9h	2.90E-02	6.73E-02	1.06E+00	2.5E-03	2.65E-03
Np-239	2.35d	1.23E-02	6.73E-02	1.06E+00	2.2E-03	2.32E-03

SECONDARY SIDE ACTIVITIES

Nuclide	removal rate r	adjust. factor f'	adjust. factor f"	nominal secondary water ANSI-18.1 uCi/gm	secondary water WBN uCi/gm	nominal secondary steam ANSI-18.1 uCi/gm	secondary steam WBN uCi/gm
Class 1							
Kr-85m	0.00E+00	0.00E+00	1.07E+00	0.0E+00	0.00E+00	3.4E-08	3.63E-08
Kr-85	0.00E+00	0.00E+00	6.19E-01	0.0E+00	0.00E+00	8.9E-08	5.51E-08
Kr-87	0.00E+00	0.00E+00	1.07E+00	0.0E+00	0.00E+00	3.0E-08	3.22E-08
Kr-88	0.00E+00	0.00E+00	1.07E+00	0.0E+00	0.00E+00	5.9E-08	6.31E-08
Xe-131m	0.00E+00	0.00E+00	8.96E-01	0.0E+00	0.00E+00	1.5E-07	1.34E-07
Xe-133m	0.00E+00	0.00E+00	1.02E+00	0.0E+00	0.00E+00	1.5E-08	1.54E-08
Xe-133	0.00E+00	0.00E+00	9.72E-01	0.0E+00	0.00E+00	5.4E-07	5.25E-07
Xe-135m	0.00E+00	0.00E+00	1.07E+00	0.0E+00	0.00E+00	2.7E-08	2.90E-08
Xe-135	0.00E+00	0.00E+00	1.06E+00	0.0E+00	0.00E+00	1.8E-07	1.91E-07
Xe-137	0.00E+00	0.00E+00	1.07E+00	0.0E+00	0.00E+00	7.1E-09	7.62E-09
Xe-138	0.00E+00	0.00E+00	1.07E+00	0.0E+00	0.00E+00	2.5E-08	2.68E-08
Class 2							
Br-84	3.00E-01	1.27E+00	1.27E+00	7.5E-08	9.56E-08	7.5E-10	9.56E-10
I-131	3.00E-01	7.85E-01	7.85E-01	1.8E-06	1.41E-06	1.8E-08	1.41E-08
I-132	3.00E-01	1.09E+00	1.09E+00	3.1E-06	3.37E-06	3.1E-08	3.37E-08
I-133	3.00E-01	8.40E-01	8.40E-01	4.8E-06	4.03E-06	4.8E-08	4.03E-08
I-134	3.00E-01	1.22E+00	1.22E+00	2.4E-06	2.93E-06	2.4E-08	2.93E-08
I-135	3.00E-01	9.38E-01	9.38E-01	6.6E-06	6.19E-06	6.6E-08	6.19E-08
Class 3							
Rb-88	1.45E-01	1.39E+00	1.39E+00	5.3E-07	7.36E-07	2.6E-09	3.61E-09
Cs-134	1.45E-01	1.39E+00	1.39E+00	3.3E-07	4.58E-07	1.7E-09	2.36E-09
Cs-136	1.45E-01	1.39E+00	1.39E+00	4.0E-08	5.56E-08	2.0E-10	2.78E-10
Cs-137	1.45E-01	1.39E+00	1.39E+00	4.4E-07	6.11E-07	2.2E-09	3.05E-09
Class 4							
N-16	0.00E+00	1.29E+00	1.29E+00	1.00E-06	1.29E-06	1.0E-07	1.29E-07

Note: N-16 will be 0.0 uCi/g outside the shield building, and up to 1E-6 uCi/gm inside. The exact concentration will be highly dependent on how long the water has been outside the core (due to the very short half life).

SECONDARY SIDE ACTIVITIES (continued)

Nuclide	removal rate r	adjust. factor f'	adjust. factor f"	nominal secondary water ANSI-18.1 uCi/gm	secondary water WBN uCi/gm	nominal secondary steam ANSI-18.1 uCi/gm	secondary steam WBN uCi/gm
Class 5							
H-3	0.00E+00	1.00E+00	1.00E+00	1.0E-03	1.00E-03	1.0E-03	1.00E-03
Class 6							
Na-24	1.93E-01	1.24E+00	1.24E+00	1.5E-06	1.86E-06	7.5E-09	9.30E-09
Cr-51	1.93E-01	1.20E+00	1.20E+00	1.3E-07	1.56E-07	6.3E-10	7.56E-10
Mn-54	1.93E-01	1.20E+00	1.20E+00	6.5E-08	7.80E-08	3.3E-10	3.96E-10
Fe-55	1.93E-01	1.20E+00	1.20E+00	4.9E-08	5.88E-08	2.5E-10	3.00E-10
Fe-59	1.93E-01	1.20E+00	1.20E+00	1.2E-08	1.44E-08	6.1E-11	7.32E-11
Co-58	1.93E-01	1.20E+00	1.20E+00	1.9E-07	2.28E-07	9.4E-10	1.13E-09
Co-60	1.93E-01	1.20E+00	1.20E+00	2.2E-08	2.64E-08	1.1E-10	1.32E-10
Zn-65	1.93E-01	1.20E+00	1.20E+00	2.1E-08	2.52E-08	1.0E-10	1.20E-10
Sr-89	1.93E-01	1.20E+00	1.20E+00	5.7E-09	6.84E-09	2.9E-11	3.48E-11
Sr-90	1.93E-01	1.20E+00	1.20E+00	4.9E-10	5.88E-10	2.5E-12	3.00E-12
Sr-91	1.93E-01	1.26E+00	1.26E+00	2.8E-08	3.52E-08	1.4E-10	1.76E-10
Y-91m	1.93E-01	1.35E+00	1.35E+00	3.2E-09	4.34E-09	1.6E-11	2.17E-11
Y-91	1.93E-01	1.20E+00	1.20E+00	2.1E-10	2.52E-10	1.1E-12	1.32E-12
Y-93	1.93E-01	1.25E+00	1.25E+00	1.2E-07	1.50E-07	6.1E-10	7.65E-10
Zr-95	1.93E-01	1.20E+00	1.20E+00	1.6E-08	1.92E-08	7.9E-11	9.48E-11
Nb-95	1.93E-01	1.20E+00	1.20E+00	1.1E-08	1.32E-08	5.7E-11	6.84E-11
Mo-99	1.93E-01	1.21E+00	1.21E+00	2.5E-07	3.03E-07	1.2E-09	1.45E-09
Tc-99m	1.93E-01	1.28E+00	1.28E+00	1.1E-07	1.40E-07	5.7E-10	7.27E-10
Ru-103	1.93E-01	1.20E+00	1.20E+00	3.1E-07	3.72E-07	1.6E-09	1.92E-09
Ru-106	1.93E-01	1.20E+00	1.20E+00	3.7E-06	4.44E-06	1.8E-08	2.16E-08
Ag-110m	1.93E-01	1.20E+00	1.20E+00	5.3E-08	6.36E-08	2.7E-10	3.24E-10
Te-129m	1.93E-01	1.20E+00	1.20E+00	7.8E-09	9.36E-09	3.9E-11	4.68E-11
Te-129	1.93E-01	1.35E+00	1.35E+00	2.2E-07	2.96E-07	1.1E-09	1.48E-09
Te-131m	1.93E-01	1.22E+00	1.22E+00	5.4E-08	6.60E-08	2.7E-10	3.30E-10
Te-131	1.93E-01	1.37E+00	1.37E+00	2.9E-08	3.97E-08	1.5E-10	2.05E-10
Te-132	1.93E-01	1.21E+00	1.21E+00	6.6E-08	7.98E-08	3.3E-10	3.99E-10
Ba-140	1.93E-01	1.20E+00	1.20E+00	5.2E-07	6.25E-07	2.6E-09	3.12E-09
La-140	1.93E-01	1.22E+00	1.22E+00	9.3E-07	1.13E-06	4.6E-09	5.60E-09
Ce-141	1.93E-01	1.20E+00	1.20E+00	6.1E-09	7.32E-09	3.1E-11	3.72E-11
Ce-143	1.93E-01	1.22E+00	1.22E+00	1.0E-07	1.22E-07	5.1E-10	6.23E-10
Ce-144	1.93E-01	1.20E+00	1.20E+00	1.6E-07	1.92E-07	8.2E-10	9.83E-10
W-187	1.93E-01	1.23E+00	1.23E+00	8.7E-08	1.07E-07	4.4E-10	5.40E-10
Np-239	1.93E-01	1.21E+00	1.21E+00	8.4E-08	1.02E-07	4.2E-10	5.09E-10

Results

The following table presents the adjusted expected reactor coolant and secondary side equilibrium nuclide activities. These activities are intended for possible use in environmental reports, in normal dose calculations for equipment qualification purposes, and for other applications where the use of expected average data over the life of the plant would be appropriate. It should be noted that the failure of a TPBAR is considered an abnormal event.

Nuclide	Reactor Coolant WBN uCi/gm	Secondary Water WBN uCi/gm	Secondary Steam WBN uCi/gm
Class 1			
Kr-85m	1.71E-01	0.00E+00	3.63E-08
Kr-85	2.66E-01	0.00E+00	5.51E-08
Kr-87	1.61E-01	0.00E+00	3.22E-08
Kr-88	3.00E-01	0.00E+00	6.31E-08
Xe-131m	6.54E-01	0.00E+00	1.34E-07
Xe-133m	7.17E-02	0.00E+00	1.54E-08
Xe-133	2.53E+00	0.00E+00	5.25E-07
Xe-135m	1.39E-01	0.00E+00	2.90E-08
Xe-135	9.04E-01	0.00E+00	1.91E-07
Xe-137	3.65E-02	0.00E+00	7.62E-09
Xe-138	1.29E-01	0.00E+00	2.68E-08
Class 2			
Br-84	1.72E-02	9.56E-08	9.56E-10
I-131	4.77E-02	1.41E-06	1.41E-08
I-132	2.25E-01	3.37E-06	3.37E-08
I-133	1.49E-01	4.03E-06	4.03E-08
I-134	3.64E-01	2.93E-06	2.93E-08
I-135	2.78E-01	6.19E-06	6.19E-08
Class 3			
Rb-88	2.04E-01	7.36E-07	3.61E-09
Cs-134	7.39E-03	4.58E-07	2.36E-09
Cs-136	9.08E-04	5.56E-08	2.78E-10
Cs-137	9.79E-03	6.11E-07	3.05E-09
Class 4			
N-16	4.00E+01	1.29E-06	1.29E-07

Nuclide Class 5	Reactor Coolant uCi/gm	Secondary Water uCi/gm	Secondary Steam uCi/gm
H-3	1.00E+00	1.00E-03	1.00E-03
TPC 704	5.10E+00'	5.10E-03''	5.10E-03''
1 rod	5.25E+01	5.25E-02''	5.25E-02''
2 rod	9.98E+01	9.98E-02''	9.98E-02''
TPC 1900	1.20E+01'	1.20E-02''	1.20E-02''
1 rod	5.94E+01	5.94E-02''	5.94E-02''
2 rod	1.07E+02	1.07E-01''	1.07E-01''
Class 6			
Na-24	4.99E-02	1.86E-06	9.30E-09
Cr-51	3.26E-03	1.56E-07	7.56E-10
Mn-54	1.68E-03	7.80E-08	3.96E-10
Fe-55	1.26E-03	5.88E-08	3.00E-10
Fe-59	3.16E-04	1.44E-08	7.32E-11
Co-58	4.84E-03	2.28E-07	1.13E-09
Co-60	5.58E-04	2.64E-08	1.32E-10
Zn-65	5.37E-04	2.52E-08	1.20E-10
Sr-89	1.47E-04	6.84E-09	3.48E-11
Sr-90	1.26E-05	5.88E-10	3.00E-12
Sr-91	1.02E-03	3.52E-08	1.76E-10
Y-90 *	1.26E-05	5.88E-10	3.00E-12
Y-91m	4.93E-04	4.34E-09	2.17E-11
Y-91	5.47E-06	2.52E-10	1.32E-12
Y-93	4.46E-03	1.50E-07	7.65E-10
Zr-95	4.10E-04	1.92E-08	9.48E-11
Nb-95	2.95E-04	1.32E-08	6.84E-11
Mo-99	6.75E-03	3.03E-07	1.45E-09
Tc-99m	5.01E-03	1.40E-07	7.27E-10
Ru-103	7.89E-03	3.72E-07	1.92E-09
Ru-106	9.47E-02	4.44E-06	2.16E-08
Rh-103m *	7.89E-03	3.72E-07	1.92E-09
Rh-106 *	9.47E-02	4.44E-06	2.16E-08
Ag-110m	1.37E-03	6.36E-08	3.24E-10
Te-129m	2.00E-04	9.36E-09	4.68E-11
Te-129	2.57E-02	2.96E-07	1.48E-09
Te-131m	1.59E-03	6.60E-08	3.30E-10
Te-131	8.26E-03	3.97E-08	2.05E-10
Te-132	1.79E-03	7.98E-08	3.99E-10
Ba-137m *	9.79E-03	6.11E-07	3.05E-09
Ba-140	1.37E-02	6.25E-07	3.12E-09
La-140	2.64E-02	1.13E-06	5.60E-09
Ce-141	1.58E-04	7.32E-09	3.72E-11
Ce-143	2.96E-03	1.22E-07	6.23E-10
Ce-144	4.21E-03	1.92E-07	9.83E-10
Pr-143 **	2.96E-03	1.22E-07	6.23E-10
Pr-144 *	4.21E-03	1.92E-07	9.83E-10
W-187	2.65E-03	1.07E-07	5.40E-10
Np-239	2.32E-03	1.02E-07	5.09E-10

* Assumption 2 -daughter which is in secular equilibrium with parent see text for more detail
** See Assumption 3
' Maximum for TPC with no dilution under normal operation.
'' Secondary side is scaled to the primary side i.e. ratio is same for conventional core and TPC.

References

*1. ANS N237-1976 (ANS -18.1), "Radioactive Materials in Principal Fluid Streams of Light-Water-Cooled Nuclear Power Plants," May 11, 1976

2. ANSI/ANS-18.1-1984, Radioactive Source Term for Normal Operation of Light Water Reactors -Tables 2,6,9,and 11 attached

3. WBN FSAR Table 11.1.6 Amendment 72

4. N3-68-4001 R2 "Reactor Coolant System" RIMS# B26 890403 004

5. Lederer, C.Micheal and Shirley, Virginia eds. "Table of Isotopes," seventh edition, John Wiley and Sons, Inc., New York, 1978

6. N3-1-4002 R3 "Main Steam System" RIMS# B26 880719 079

7. telephone conversation between M. Berg and C.D. Thomas, Jr. of Yankee Atomic Electric Company on 3/13/87 -attached

8. N3-62-4001 R2 "Chemical and Volume Control System" RIMS# B26 880726 058

9. WBNAPS3-052 R1 "Minimum Detectable Leak Rate For the Steam Generator Blowdown System" RIMS# B45 880620 237

10. WBNNAL3-002 R3 "100-Day LOCA-DBA Source Terms for the EGTS and ABGTS Filters, Containment, Sump, and Shield Building Annulus"

11. Kennan, Joseph H. and Keyes, Frederick G. "Thermodynamic Properties of Steam" 1st edition, John Wiley and Sons, Inc. New York

12. NUREG-0017 R1 "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors" April 1976

13. Letter from Jim Chardos, TVA, to M.L. Travis, Wesdyne, "TVA Plant Specific Tritium Concentration Data," Sept. 11, 2000, RIMS# T35 000911 913

14. EDC E50629A

15. WAT-D-10890, Westinghouse document NDP-00-0326, "Transmittal of Waste Management Evaluation for the Watt's Bar Tritium Production Core," RIMS# T71 001204 807

16. WB1RSG-TR-02 R4, "RSG-OSG Comparison Document" transmitted via TVWES-0725

17. DCN 51754A

18. DCN 61599

* This reference is for information only. It is not used for design input.

Appendix A
Steam Generator Replacement Results

This project revised the following parameters used in the body of this calculation (OSG is the original steam generators):

	OSG	RSG (ref. 16)
Steam Flow Rate - FS	1.5E+07	1.54E+07
Weight of Water in the RCS - WP (based on density at RV inlet temp)	5.4E+05	5.78E+05
Weight of Water in all SG's - WS	3.48E+05	6.79E+05

The following are the results of following the same methodology in the body of this calculation using the above RSG parameters.

Nuclide	half life ref.5		Lambda 1/hr	removal rate R	adjust. factor f	nominal reactor coolant ANSI-18.1 uCi/gm	calculated reactor coolant WBN uCi/gm
Class 1							
Kr-85m	4.48	h	1.55E-01	1.46E-03	9.99E-01	1.60E-01	1.60E-01
Kr-85	10.7	y	7.39E-06	1.46E-03	6.19E-01	4.30E-01	2.66E-01
Kr-87	76	m	5.47E-01	1.46E-03	1.00E+00	1.50E-01	1.50E-01
Kr-88	2.84	h	2.44E-01	1.46E-03	1.00E+00	2.80E-01	2.80E-01
Xe-131m	11.77	d	2.45E-03	1.46E-03	8.59E-01	7.30E-01	6.27E-01
Xe-133m	2.19	d	1.32E-02	1.46E-03	9.64E-01	7.00E-02	6.75E-02
Xe-133	5.25	d	5.50E-03	1.46E-03	9.22E-01	2.60E+00	2.40E+00
Xe-135m	15.6	m	2.67E+00	1.46E-03	1.00E+00	1.30E-01	1.30E-01
Xe-135	9.1	h	7.62E-02	1.46E-03	9.95E-01	8.50E-01	8.46E-01
Xe-137	3.82	m	1.09E+01	1.46E-03	1.00E+00	3.40E-02	3.41E-02
Xe-138	14.1	m	2.95E+00	1.46E-03	1.00E+00	1.20E-01	1.20E-01
Class 2							
Br-84	31.8	m	1.31E+00	6.34E-02	1.01E+00	1.60E-02	1.61E-02
I-131	8.04	d	3.59E-03	6.34E-02	1.06E+00	4.50E-02	4.75E-02
I-132	2.28	h	3.04E-01	6.34E-02	1.01E+00	2.10E-01	2.13E-01
I-133	20.9	h	3.32E-02	6.34E-02	1.04E+00	1.40E-01	1.46E-01
I-134	52.6	m	7.91E-01	6.34E-02	1.01E+00	3.40E-01	3.42E-01
I-135	6.61	h	1.05E-01	6.34E-02	1.02E+00	2.60E-01	2.66E-01
Class 3							
Rb-88	17.8	m	2.34E+00	3.56E-02	1.00E+00	1.90E-01	1.91E-01
Cs-134	2.062	y	3.84E-05	3.56E-02	1.04E+00	7.10E-03	7.39E-03
Cs-136	13.1	d	2.20E-03	3.56E-02	1.04E+00	8.70E-04	9.04E-04
Cs-137	30.17	y	2.62E-06	3.56E-02	1.04E+00	9.40E-03	9.79E-03
Class 4							
N-16	7.13	s	3.50E+02	0.00E+00	1.00E+00	4.00E+01	4.00E+01
Class 5							
H-3	12.33	y	6.42E-06	0.00E+00	1.00E+00	1.00E+00	1.00E+00
TPC 704/1900							5.10E+00'/1.20E+01'
1 rod							5.25E+01/5.94E+01
2 rod							9.98E+01/1.07E+02

Appendix A - Steam Generator Replacement Results(continued)

Nuclide	half life ref.5		Lambda 1/hr	removal rate R	adjust. factor f	nominal reactor coolant ANSI-18.1 uCi/gm	calculated reactor coolant WBN uCi/gm
Class 6							
Na-24	15.02	h	4.61E-02	6.29E-02	1.03E+00	4.70E-02	4.85E-02
Cr-51	27.7	d	1.04E-03	6.29E-02	1.05E+00	3.10E-03	3.26E-03
Mn-54	312	d	9.26E-05	6.29E-02	1.05E+00	1.60E-03	1.68E-03
Fe-55	2.7	y	2.93E-05	6.29E-02	1.05E+00	1.20E-03	1.26E-03
Fe-59	44.6	d	6.48E-04	6.29E-02	1.05E+00	3.00E-04	3.16E-04
Co-58	70.8	d	4.08E-04	6.29E-02	1.05E+00	4.60E-03	4.84E-03
Co-60	5.271	y	1.50E-05	6.29E-02	1.05E+00	5.30E-04	5.58E-04
Zn-65	244.1	d	1.18E-04	6.29E-02	1.05E+00	5.10E-04	5.37E-04
Sr-89	50.5	d	5.72E-04	6.29E-02	1.05E+00	1.40E-04	1.47E-04
Sr-90	28.8	y	2.75E-06	6.29E-02	1.05E+00	1.20E-05	1.26E-05
Sr-91	9.5	h	7.30E-02	6.29E-02	1.03E+00	9.60E-04	9.85E-04
Y-90*							1.26E-05
Y-91m	49.7	m	8.37E-01	6.29E-02	1.01E+00	4.60E-04	4.63E-04
Y-91	58.5	d	4.94E-04	6.29E-02	1.05E+00	5.20E-06	5.47E-06
Y-93	10.2	h	6.80E-02	6.29E-02	1.03E+00	4.20E-03	4.31E-03
Zr-95	64	d	4.51E-04	6.29E-02	1.05E+00	3.90E-04	4.10E-04
Nb-95	35	d	8.25E-04	6.29E-02	1.05E+00	2.80E-04	2.94E-04
Mo-99	66.02	h	1.05E-02	6.29E-02	1.05E+00	6.40E-03	6.69E-03
Tc-99m	6.02	h	1.15E-01	6.29E-02	1.02E+00	4.70E-03	4.79E-03
Ru-103	39.4	d	7.33E-04	6.29E-02	1.05E+00	7.50E-03	7.89E-03
Ru-106	367	d	7.87E-05	6.29E-02	1.05E+00	9.00E-02	9.47E-02
Rh-103m*							7.89E-03
Rh-106*							9.47E-02
Ag-110m	252	d	1.15E-04	6.29E-02	1.05E+00	1.30E-03	1.37E-03
Te-129m	33.5	d	8.62E-04	6.29E-02	1.05E+00	1.90E-04	2.00E-04
Te-129	67	m	6.21E-01	6.29E-02	1.01E+00	2.40E-02	2.42E-02
Te-131m	30	h	2.31E-02	6.29E-02	1.04E+00	1.50E-03	1.56E-03
Te-131	25	m	1.66E+00	6.29E-02	1.00E+00	7.70E-03	7.73E-03
Te-132	78	h	8.89E-03	6.29E-02	1.05E+00	1.70E-03	1.78E-03
Ba-137m*							9.79E-03
Ba-140	12.79	d	2.26E-03	6.29E-02	1.05E+00	1.30E-02	1.37E-02
La-140	40.3	h	1.72E-02	6.29E-02	1.04E+00	2.50E-02	2.60E-02
Ce-141	32.5	d	8.89E-04	6.29E-02	1.05E+00	1.50E-04	1.58E-04
Ce-143	33	h	2.10E-02	6.29E-02	1.04E+00	2.80E-03	2.91E-03
Ce-144	284	d	1.02E-04	6.29E-02	1.05E+00	4.00E-03	4.21E-03
Pr-143**							2.91E-03
Pr-144*							4.21E-03
W-187	23.9	h	2.90E-02	6.29E-02	1.04E+00	2.50E-03	2.59E-03
Np-239	2.35	d	1.23E-02	6.29E-02	1.04E+00	2.20E-03	2.30E-03

* Assumption 2 -daughter which is in secular equilibrium with parent see text for more detail
** See Assumption 3
' Maximum for TPC with no dilution under normal operation.

Appendix A (continued)
Steam Generator Replacement Results

Nuclide	removal rate r	adjust. factor f'	adjust. factor f"	nominal secondary water ANSI-18.1 uCi/gm	secondary water WBN uCi/gm	nominal secondary steam ANSI-18.1 uCi/gm	secondary steam WBN uCi/gm
Class 1							
Kr-85m	0.00E+00	0.00E+00	9.73E-01	0.00E+00	0.00E+00	3.40E-08	3.31E-08
Kr-85	0.00E+00	0.00E+00	6.03E-01	0.00E+00	0.00E+00	8.90E-08	5.37E-08
Kr-87	0.00E+00	0.00E+00	9.75E-01	0.00E+00	0.00E+00	3.00E-08	2.93E-08
Kr-88	0.00E+00	0.00E+00	9.74E-01	0.00E+00	0.00E+00	5.90E-08	5.75E-08
Xe-131m	0.00E+00	0.00E+00	8.36E-01	0.00E+00	0.00E+00	1.50E-07	1.25E-07
Xe-133m	0.00E+00	0.00E+00	9.39E-01	0.00E+00	0.00E+00	1.50E-08	1.41E-08
Xe-133	0.00E+00	0.00E+00	8.98E-01	0.00E+00	0.00E+00	5.40E-07	4.85E-07
Xe-135m	0.00E+00	0.00E+00	9.76E-01	0.00E+00	0.00E+00	2.70E-08	2.64E-08
Xe-135	0.00E+00	0.00E+00	9.69E-01	0.00E+00	0.00E+00	1.80E-07	1.74E-07
Xe-137	0.00E+00	0.00E+00	9.76E-01	0.00E+00	0.00E+00	7.10E-09	6.93E-09
Xe-138	0.00E+00	0.00E+00	9.76E-01	0.00E+00	0.00E+00	2.50E-08	2.44E-08
Class 2							
Br-84	1.56E-01	6.72E-01	6.72E-01	7.50E-08	5.04E-08	7.50E-10	5.04E-10
I-131	1.56E-01	7.59E-01	7.59E-01	1.80E-06	1.37E-06	1.80E-08	1.37E-08
I-132	1.56E-01	6.90E-01	6.90E-01	3.10E-06	2.14E-06	3.10E-08	2.14E-08
I-133	1.56E-01	7.38E-01	7.38E-01	4.80E-06	3.54E-06	4.80E-08	3.54E-08
I-134	1.56E-01	6.76E-01	6.76E-01	2.40E-06	1.62E-06	2.40E-08	1.62E-08
I-135	1.56E-01	7.14E-01	7.14E-01	6.60E-06	4.71E-06	6.60E-08	4.71E-08
Class 3							
Rb-88	7.53E-02	6.85E-01	6.85E-01	5.30E-07	3.63E-07	2.60E-09	1.78E-09
Cs-134	7.53E-02	1.37E+00	1.37E+00	3.30E-07	4.53E-07	1.70E-09	2.33E-09
Cs-136	7.53E-02	1.35E+00	1.35E+00	4.00E-08	5.40E-08	2.00E-10	2.70E-10
Cs-137	7.53E-02	1.37E+00	1.37E+00	4.40E-07	6.04E-07	2.20E-09	3.02E-09
Class 4							
N-16	0.00E+00	6.62E-01	6.62E-01	1.00E-06	6.62E-07	1.00E-07	6.62E-08
Class 5							
H-3	0.00E+00	1.00E+00	1.00E+00	1.00E-03	1.00E-03	1.00E-03	1.00E-03
TPC-704					5.10E-03’		5.10E-03’
1 rod					5.25E-02’		5.25E-02’
2 rod					9.98E-02’		9.98E-02’
TPC-1900					1.20E-02’		1.20E-02’
1 rod					5.94E-02’		5.94E-02’
2 rod					1.07E-01’		1.07E-01’

’ Secondary side is scaled to the primary side i.e. ratio is same for conventional core and TPC.

Appendix A (continued)
Steam Generator Replacement Results

Nuclide	removal rate r	adjust. factor f'	adjust. factor f"	nominal secondary water ANSI-18.1 uCi/gm	secondary water WBN uCi/gm	nominal secondary steam ANSI-18.1 uCi/gm	secondary steam WBN uCi/gm
Class 6							
Na-24	1.00E-01	1.01E+00	1.01E+00	1.50E-06	1.51E-06	7.50E-09	7.56E-09
Cr-51	1.00E-01	1.18E+00	1.18E+00	1.30E-07	1.53E-07	6.30E-10	7.41E-10
Mn-54	1.00E-01	1.18E+00	1.18E+00	6.50E-08	7.68E-08	3.30E-10	3.90E-10
Fe-55	1.00E-01	1.18E+00	1.18E+00	4.90E-08	5.79E-08	2.50E-10	2.95E-10
Fe-59	1.00E-01	1.18E+00	1.18E+00	1.20E-08	1.41E-08	6.10E-11	7.19E-11
Co-58	1.00E-01	1.18E+00	1.18E+00	1.90E-07	2.24E-07	9.40E-10	1.11E-09
Co-60	1.00E-01	1.18E+00	1.18E+00	2.20E-08	2.60E-08	1.10E-10	1.30E-10
Zn-65	1.00E-01	1.18E+00	1.18E+00	2.10E-08	2.48E-08	1.00E-10	1.18E-10
Sr-89	1.00E-01	1.18E+00	1.18E+00	5.70E-09	6.72E-09	2.90E-11	3.42E-11
Sr-90	1.00E-01	1.18E+00	1.18E+00	4.90E-10	5.79E-10	2.50E-12	2.95E-12
Sr-91	1.00E-01	9.53E-01	9.53E-01	2.80E-08	2.67E-08	1.40E-10	1.33E-10
Y-90*					5.79E-10		2.95E-12
Y-91m	1.00E-01	7.16E-01	7.16E-01	3.20E-09	2.29E-09	1.60E-11	1.15E-11
Y-91	1.00E-01	1.18E+00	1.18E+00	2.10E-10	2.48E-10	1.10E-12	1.30E-12
Y-93	1.00E-01	9.62E-01	9.62E-01	1.20E-07	1.15E-07	6.10E-10	5.87E-10
Zr-95	1.00E-01	1.18E+00	1.18E+00	1.60E-08	1.89E-08	7.90E-11	9.32E-11
Nb-95	1.00E-01	1.18E+00	1.18E+00	1.10E-08	1.29E-08	5.70E-11	6.71E-11
Mo-99	1.00E-01	1.13E+00	1.13E+00	2.50E-07	2.82E-07	1.20E-09	1.35E-09
Tc-99m	1.00E-01	8.94E-01	8.94E-01	1.10E-07	9.84E-08	5.70E-10	5.10E-10
Ru-103	1.00E-01	1.18E+00	1.18E+00	3.10E-07	3.65E-07	1.60E-09	1.88E-09
Ru-106	1.00E-01	1.18E+00	1.18E+00	3.70E-06	4.37E-06	1.80E-08	2.13E-08
Rh-103m*					3.65E-07		1.88E-09
Rh-106*					4.37E-06		2.13E-08
Ag-110m	1.00E-01	1.18E+00	1.18E+00	5.30E-08	6.26E-08	2.70E-10	3.19E-10
Te-129m	1.00E-01	1.18E+00	1.18E+00	7.80E-09	9.18E-09	3.90E-11	4.59E-11
Te-129	1.00E-01	7.32E-01	7.32E-01	2.20E-07	1.61E-07	1.10E-09	8.05E-10
Te-131m	1.00E-01	1.08E+00	1.08E+00	5.40E-08	5.82E-08	2.70E-10	2.91E-10
Te-131	1.00E-01	6.92E-01	6.92E-01	2.90E-08	2.01E-08	1.50E-10	1.04E-10
Te-132	1.00E-01	1.14E+00	1.14E+00	6.60E-08	7.50E-08	3.30E-10	3.75E-10
Ba-137m*					6.04E-07		3.02E-09
Ba-140	1.00E-01	1.17E+00	1.17E+00	5.20E-07	6.08E-07	2.60E-09	3.04E-09
La-140	1.00E-01	1.10E+00	1.10E+00	9.30E-07	1.02E-06	4.60E-09	5.06E-09
Ce-141	1.00E-01	1.18E+00	1.18E+00	6.10E-09	7.18E-09	3.10E-11	3.65E-11
Ce-143	1.00E-01	1.08E+00	1.08E+00	1.00E-07	1.08E-07	5.10E-10	5.53E-10
Ce-144	1.00E-01	1.18E+00	1.18E+00	1.60E-07	1.89E-07	8.20E-10	9.69E-10
Pr-143**					1.08E-07		5.53E-10
Pr-144*					1.89E-07		9.69E-10
W-187	1.00E-01	1.06E+00	1.06E+00	8.70E-08	9.20E-08	4.40E-10	4.65E-10
Np-239	1.00E-01	1.12E+00	1.12E+00	8.40E-08	9.41E-08	4.20E-10	4.70E-10

* Assumption 2 -daughter which is in secular equilibrium with parent see text for more detail
** See Assumption 3
It is noted that these values are slightly less than those determined in the body of this calculation. Therefore either set of values can be used as the values in the body of the calculation are conservative.

Attachment A

Telephone conversation between M. Berg and C. D. Thomas, Jr. of Yankee Atomic Electric Co. (co-author of ANSI/ANS-18.1) on 3/13/87

Topic: The reason for the deletion of several isotopes from ANSI/ANS-18.1-1984 isotope listing.

The report ANS N237-1976 (ANS -18.1) was based on scant data. The report ANSI/ANS-18.1-1984 was based on a larger set of data and therefore more accurate. Some isotopes on the former report were left out of the 18.1-1984 report. The 18.1-1984 report utilized a weighting factor to determine the relative importance of an isotope. This factor was based on fission yield, number and energy of the gamma rays, and the efficiency of detection. Thus the isotopes Kr-83m, Br-83, Br-85, Rb-86, Te-125m, Te-127m, Te-127, and I-130 were deemed to be unimportant with respect to the other isotopes and were left out of the report.

Some of the isotopes were left out because they were in secular equilibrium with the parent, and it was felt by the authors that the parents were more important. The isotopes left out were Y-90, Rh-103m, Rh-106, Ba-137m, and Pr-144.

The isotope Pr-143 was left off the list because it is a pure beta emitter.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1

WBNTSR100, Revision 13, "Design Releases to Show Compliance with 10CFR20"

(35 pages including this cover)

TVA NUCLEAR CALCULATION COVERSHEET / CTS UPDATE

REV 0 EDMS/RIMS NO.	CTS TYPE:	EDMS TYPE:	EDMS ACCESSION NO (N/A for REV. 0)
B26950110310	Calculation	CALCULATIONS (NUCLEAR)	

Calc Title: Design Releases to Show Compliance with 10CFR20

	ORG	PLANT	BRANCH	NUMBER	CUR REV	NEW REV
CALC ID	NUC	WBN	NTB	WBNTSR100	012	013

CTS UPDATE ONLY ☐ (Verifier and Approval Signatures Not Required)

NO CTS CHANGES ☐ (For calc revision, CTS has been reviewed and no CTS changes required)

UNIT (check one)	SYSTEMS	UNIDS
0 ☒, 1 ☐, 2 ☐, 3 ☐	N/A	N/A

ECP, N/A	APPLICABLE DESIGN DOCUMENT(S)	CLASSIFICATION
DCN 61599	N/A	EO

QUALITY RELATED?	SAFETY RELATED? (If yes, QR = yes)	UNVERIFIED ASSUMPTION	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?	DESIGN OUTPUT ATTACHMENT?	SAR/TS and/or ISFSI SAR/CoC AFFECTED
Yes ☒ No ☐	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☐ No ☒	Yes ☒ No ☐	Yes ☐ No ☒

CALCULATION NUMBER REQUESTOR	PREPARING DISCIPLINE	VERIFICATION METHOD	NEW METHOD OF ANALYSIS
Name: N/A PHONE: N/A	N	Design Review	☐ Yes ☒ No

PREPARER (PRINT NAME AND SIGN)	DATE	CHECKER (PRINT NAME AND SIGN)	DATE
Mehran A. Mohammadian [signature]	3/7/16	Aleksandar Milicevic [signature]	3/7/16
VERIFIER (PRINT NAME AND SIGN)	**DATE**	**APPROVAL (PRINT NAME AND SIGN)**	**DATE**
Aleksandar Milicevic [signature]	3/7/16	James V Ware [signature] (QAK 3/7/16)	3/14/16

STATEMENT OF PROBLEM/ABSTRACT

Determine the design liquid and gaseous releases and show that these releases are less than the 10CFR20 App.B Table 2 Effluent Concentration Limits (ECL).

Abstract:

The Standard Review Plan sections 11.2.III.2.c, 11.2.IV.3, 11.3.III.2.b, and 11.3.IV.3 require that the gaseous and liquid releases based on 1% failed fuel be within 10CFR20 App.B Table 2 limits. This calculation took the expected gaseous releases from calculation TI534 and the expected liquid releases from WBNTSR093. These releases were scaled to design (1% failed fuel) levels from FSAR data (and justified with Westinghouse WCAP-7664 R1). The design release concentration for each isotope was divided by the 10CFR20 App.B Table 2 Effluent Concentration Limit (ECL). The concentration/ECL fraction was summed over all isotopes. Also analyzed was the case of liquid releases at design levels, except for iodines, which were limited to Technical Specification limits of 0.265 µCi/gm I-131 equivalent. The results for each case can be found in the results section.

The 10CFR20 limits will not be exceeded for design (1% failed fuel) releases for gas, or liquids when the Condensate Polisher Demineralizer regeneration waste is processed by the mobile demineralizers. In the event that the long term release of this waste is projected to result in exceeding the 10CFR20 limits, the design of the plant allows the waste to be processed by the mobile demineralizer system. With the mobile demineralizer system processing the regeneration waste, the liquid design releases are below the 10CFR20 limits. The release concentrations determined in this calculation are not expected to occur because the design reactor coolant concentrations exceed the Technical Specification limits. With iodine limited to the Technical Specification limit of 0.265 µCi/gm I-131 equivalent, the releases will be less than the 10CFR20 limits. The design of the gas and liquid radwaste systems meet the requirements of 10CFR20.

The presence of a Tritiated Water Storage Tank (TWST) has no affect on the results or conclusions of this calculation.

The case of operating with a Tritium Production Core (TPC) is also analyzed in this calculation. Gaseous and liquid releases of tritium (H-3) are based on a H-3 permeation rate of 5 Ci/TPBAR/year and 1900 TPBARs loaded in the TPC.

MICROFICHE/EFICHE Yes ☐ No ☒ FICHE NUMBER(S)

NPG CALCULATION COVERSHEET / CTS UPDATE

CALC ID	ORG	PLANT	BRANCH	NUMBER	REV
	NUC	WBN	NTB	WBNTSR100	013

BUILDING	ROOM	ELEVATION	COORD/AZIM	FIRM
NA	N/A	N/A	N/A	S&L

CATEGORIES N/A

KEYWORDS (A-add, D-delete)

ACTION (A/D)	KEYWORD	A/D	KEYWORD

CROSS-REFERENCES (A-add, D-delete)

ACTION (A/D)	XREF CODE	XREF PLANT	XREF TYPE	XREF NUMBER	XREF REV

CTS ONLY UPDATES:
Following are required only when making keyword/cross reference CTS updates and page 1 of form NEDP-2-1 is not included:

PREPARER (PRINT NAME AND SIGN)	DATE	CHECKER (PRINT NAME AND SIGN)	DATE
PREPARER PHONE NO.	EDMS ACCESSION NO.		

TVAN CALCULATION RECORD OF REVISION

CALCULATION IDENTIFIER WBNTSR-100

Title Design Releases to Show Compliance with 10CFR20

Revision No.	DESCRIPTION OF REVISION
0	Initial Issue
1	Revision 1 was performed because the expected liquid releases from WBNTSR-093 have changed. All pages were renumbered, but only the pages that had text changes are listed in the pages changed section below. pages changed: 1-6, 10-12 pages added: 11 paged deleted: none
2	Revision 2 was performed to incorporate gas releases from the alternate mode of operation of venting containment instead of purging. Also incorporated is a case with Technical Justification limits for iodines in the liquid (instead of 1% failed fuel design). Pages added: 9.1 pages deleted: none pages changed: 1-5, 7-12 R2: 13 total pages
3	Revision 3 incorporates the continuous containment filtered vent to the annulus case. DCN D-50165-A added a filter to the vent line. The unfiltered case determined in revision 2 was retained for historical purposes. Pages added: 9.2 Pages deleted: none Pages changed: 1-5, 7-12, 9.1 R3 total pages: 14
4	Revision 4 was performed to add the alternate operation mode of direct Steam Generator Blowdown release to the river without Condensate Polisher Demineralizer processing, and no Condensate Polisher Demineralizer processing of the Condensate. Additionally, cases were also analyzed with no Cooling Tower Blowdown dilution flow (20,000 gpm). This revision supports DCN D-50502-A. All pages were renumbered. Pages with actual text changes are marked with revision bars. Pages added: new cover Pages deleted: none Pages changed:1a (old cover), 2-23 R4: 24 total pages
5	Revision 5 was performed to add the use of a Tritium Production Core. This revision supports EDC E50629A. The amount of gaseous release of tritium was changed in TI-534 and is thus reflected in this calculation. Pages added: 1 Pages deleted: 0 Pages changed: 1b,2,3,7,9-14,16-18,22,23 R5 : 25 total pages
6	Revision 6 provides additional results associated with EDC-E50629A in conjunction with R5. This revision evaluates the unlikely event of a 1 or 2 TPBAR failure. Tritium releases are the only values effected by this failure. Non-TPC tritium value was revised per revision of WBNTSR-093, which changed the power from 105% of a nominal power to 102% of nominal power. This calculation will need a 50.59 reviews as it affects FSAR Tables 11.3-8a and -8b, and 11.2-4a and -4b. Pages added: none Pages deleted: none Pages changed : 2,3,7,9-14,16-18,22,23 R6 : 25 total pages

TVAN CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER WBNTSR100	
Title	Design Releases to Show Compliance with 10CFR20
Revision No.	**DESCRIPTION OF REVISION**
7	Revision 7 of this calculation was created to add/update Unit 2 applicability. This calculation is applicable to Unit 2 based on the following: • An unverified assumption has been added as part of this revision which assumes Unit 1 design changes are applicable to Unit 2. The unverified assumptions only apply to the Unit 2 portions of this calculation. Refer to the Assumptions section of this calculation for further information. • The table in the Results section of this calculation was edited to add a column showing, for 2-unit operation, the total Conc/ECL for each release scenario. This column reflects twice the value for 1-unit operation without a Tritium Production Core (TPC), as well as, in parentheses, the 2-unit value with TPC reflected in Unit 1. • Unit 2 will not be licensed for TPC. However, the inclusion of TPC in the analysis was shown not to impact the results. Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy. The effect of Unit 2 operation on Unit 1 margins and the impact on an Ultimate Heat Sink temperature of 88 °F have been reviewed with no impact. FSAR AND TECHNICAL SPECIFICATIONS HAVE BEEN REVIEWED AND ARE NOT AFFECTED BY THIS REVISION OF THE CALCULATION. Reviewer: R. White 7-1-08 Pages Added: i, ii, 2a, 7a Pages Revised/Replaced: 3, 4, 7, 22 Pages Deleted: none Total number of pages in this revision including Attachments: 29 Pages

NPG CALCULATION RECORD OF REVISION

CALCULATION IDENTIFIER: WBNTSR100	Page 2b
Title	Design Releases to Show Compliance with 10CFR20

Revision No.	DESCRIPTION OF REVISION
008	• Revision 8 of this calculation adds updated gaseous and liquid release tables to support Unit 2/dual unit operation. These tables have been added as Appendix A. • The previous revision of this calculation justified the applicability of this calculation to Unit 2/dual unit operation and updated the Results Table at the end of the calculation. • This revision does not impact the applicability of these results to Unit 2/dual unit operation. • An EDCR search has been performed for any EDCR associated with the Referenced DCNs. EDCR 52841 has been issued to incorporate the changes of DCN 50165 for Unit 2. All other referenced DCNs can be found in UVA 2. • A value on Table 11 was updated for a historical issue. The value appears to have been transcribed incorrectly as 1.89E+00 rather than the correct 1.09E+00. The results of the table and therefore the summary table of this calculation were not impacted as the correct concentration was represented in the totals. There is no impact on the results/conclusions of this calculation. Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy. Ultimate heat sink (UHS) temperature was not used as an input to the calculation analyses. Therefore, existing calculation results will not be affected by changing the UHS technical specification temperature. The WBN SAR has been reviewed by ENGINEERING LICENSING for applicability to Unit 2/dual unit operation and this revision of the calculation affects SAR section 11.2. The SAR change has ← been submitted to Unit 2 Licensing in accordance with TVA NGDC Project Procedure, PP-10, *Watts Bar Nuclear Plant Unit 2 Changes to Final Safety Analysis Report, Technical Specifications, TS Bases, and/or Technical Requirements Manual*, and is tracked as Unit 2 SAR change * . TECH SPECS, TS BASES AND TRM ARE NOT IMPACTED BY THIS REVISION. Reviewer: [signature] 11/12/2009 H. DAVID KNIGHT II Pages Added: i (Rev. 8 Coversheet), ii (Rev. 8 CCRIS Update), iii (Rev. 7 Coversheet), 2b, 3a, and Appendix A (pages A-1 to A-12) Pages Revised/Replaced: 4, 7a, and 18 Pages Deleted: i (Rev. 7 Coversheet), ii (Rev. 7 CCRIS Update) Total number of pages in this revision including Attachments: 44 Pages 29 pages (Rev. 7 ROR) + 17 pages (Rev. 8) – 2 page (Rev. 8) = 44 ← = FSAR SECTION 11.3 (SPECIFICALLY TABLES 11.3-8a AND 11.3-8c) ARE ALSO IMPACTED BY THIS REVISION * = SAR CHANGE TRACKING NUMBER TO BE PROVIDED BY TVA UPON APPROVAL OF PP-10 CHANGE PACKAGE. This page added by Revision 008.

NPG CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER: WBNTSR100	Page 2c
Title	Design Releases to Show Compliance with 10CFR20

Revision No.	DESCRIPTION OF REVISION
009	Revision 9 of this calculation was created to incorporate the corrective actions of PER 386966, and remove the remaining UVA in support of dual unit operation. • EDCR 54449 has been issued to incorporate the design changes of DCN 50502 for Unit 2. Therefore, Assumption 2 (UVA) has been removed by this revision. • This revision updated the liquid releases based on revision of calculation WBNTSR093 (Ref. 3) for PER 386966. The tables for the liquid releases have been entirely updated and incorporate the single unit and dual unit operations tables into the main body. Therefore, Appendix A has been modified to remove the liquid release tables. • PER 386966 also identifies the need to make this calculation classification "EO". A Design Output Attachment has been added (Attachment 1). • The coversheet has been corrected to show the calculation is not Safety Related. This is consistent with Revision 7 and earlier revisions of the calculation. Successors WBNAPS3083 and WBNTSR024 are impacted by this revision. The updates to these successor documents are tracked by WITEL PL-11-3841. All other successors have been reviewed and are not impacted by this revision. *Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy.* Ultimate heat sink (UHS) temperature was not used as an input to the calculation analyses. Therefore, existing calculation results will not be affected by changing the UHS technical specification temperature. The effect of Unit 2/dual unit operation on Unit 1 margins has been reviewed with no impact. Unit 1, FSAR, Section 11.2, Table 11.2- 4a and 11.2- 4b is being revised via NLDP-5 form as part of EDC 58273. A PP-10 form is being generated to revise Unit 2 FSAR Section 11.2, Table 11.2-5a through 11.2-5d in the next amendment of the Unit 2 Living FSAR. Reviewer/ [signature] 7/26/2011 J.H. KAENIK Pages Added: ia (Rev. 9 CCRIS), 2c, 3b, 12a, 12b, 13a, 14a, 15a, 16a, 17a, 18a, 19a, 20a, Attachment 1 (1 page) Pages Revised/Replaced: i (Rev. 9 CS), ii (Rev. 8 CS), 4, 7, 12, 13, 14, 15, 16, 17, 18, 19, 20, 22, 23, A-1 Pages Deleted: 7a, A-5 through A-12 Total number of pages in this revision including Attachments: 44 pages (Rev. 8) + 14 pages (Rev. 9) - 9 page (Rev. 9) = 49 Appendix A: 4 pages Attachment 1: 1 page

This page added by Revision 009.

NPG CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER WBNTSR-100	
Title	Design Releases to Show Compliance with 10CFR20
Revision No.	**DESCRIPTION OF REVISION**
010	Revision 10 of this calculation was created to incorporate updates to predecessor calculations - Calculation WBNTSR093 (ref.3) was revised as the result of PER 546323. The liquid releases used by this calculation were impacted by this revision and therefore require update. The appropriate tables have been updated. - Dual unit gaseous release tables previously contained in Appendix A have been moved into the main body. As a result Appendix A has been deleted. - The entire calculation is renumbered Successor calculation WBNTSR024 and WBNAPS3083 are impacted by this revision. The updates to these successor documents are tracked by PER 546323. All other successors have been reviewed and are not impacted by this revision. Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy. Ultimate heat sink (UHS) temperature was not used as an input to the calculation analyses. Therefore, existing calculation results will not be affected by changing the UHS technical specification temperature. The affect of Unit 2/dual unit operation on Unit 1 margins has been reviewed with no impact. Unit 1, FSAR, Section 11.2, Table 11.2-4a and 11.2-4b is being revised via NLDP-5 form as part of EDC 56202. A PP-10 form is being generated to revise Unit 2 FSAR section 11.2, Table 11.2-5a through 11.2-5d in the next amendment of the Unit 2 Living FSAR Reviewer: D.G.Fickey Pages added: 2d, 24, 25, 3a, 3b Pages revised/replaced: 1, 1a, 3-23, Attachment 1 Pages deleted: i, ia, ii, iii, 1b, 3a, 3b, 12a, 13a, 12b, 13a, 14a, 15a, 16a, 17a, 18a, 19a, 20a, Appendix A (4 pages) Total number pages in this revision: 49 pages (Rev.9)+5 pages(Rev.10)-21 page (Rev.10)=33 pages Attachment 1: 1 page
11	DCN 59397 is installing a Tritiated Water Storage Tank (TWST). The TWST does not increase the amount of the releases, therefore the addition of the tank does not affect the results of this calculation. Successor calculations are not impacted by this revision, as there are no changes to the results. The FSAR is not impacted by this revision since the results did not change. The Technical Specifications are not affected by this revision. The attached Design Output determined in revision 10 has not changed due to this revision. Reviewed by Marc Berg Pages added: none Pages deleted: none Pages changed: 1, 1a, 2d, 3, 5, 25 Total Number of Pages: 33

NPG CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER WBNTSR100	
Title	Design Releases to Show Compliance with 10CFR20
Revision No.	DESCRIPTION OF REVISION
12	Revision 12 is performed to reflect a change regarding the tritium permeation rate for a TPBAR in the tritium production core (TPC) configuration, which affects predecessor calculations WBNNAL3003, TI534, and WBNTSR093. Gaseous releases of tritium (H-3) are based on the design basis average permeation rate and 2500 TPBARs. For liquid releases of H-3, the maximum liquid concentration of H-3 that can be tolerated is calculated. The calculated maximum allowable H-3 concentration that can be released to the environment without exceeding the 10CFR20 App.B limit is made a special requirement/limiting condition of the calculation. The TPC 1 rod and 2 rod failure cases were removed from this revision because they do not reflect normal plant operation. Additionally, C-14 expected releases were revised in predecessor calculation TI534 Revision 10; therefore, C-14 values were updated in Revision 12. Tables 2, 3, and 4 are affected by this change, but the conclusions with regard to gaseous releases are not affected. Typographical and editorial corrections were made on pages 8 (4.25E+12 to 4.25E-12), 9 (2.07E-04 to 2.07E+04), and 24 (grammatical corrections). A reference to the 2.5 μCi/gm I-131 concentration in the reactor coolant was corrected on pg. 24 (ref.3 to Table 1). An error on Table 5 was corrected, where the concentration column had units of μCi/gm. The units on the concentration column were changed to μCi/cc. An error was identified in Tables 2, 3, and 4 in the dual unit H-3 (TPC) values where the non-TPC H-3 from the second operating unit was not counted in the calculation of the dual unit operation C/ECL column. The tables have been updated with the corrected dual unit H-3 (TPC) values that account for the additional non-TPC H-3 from the second operating unit. Pages replaced in Revision 12 contain an updated page header for Revision 12. All unaffected pages retain their original headers. CTS was reviewed for successor documents, and calculation WBNAPS3128 is affected. Other successor documents reviewed were not affected by the changes made in WBNTSR100 Revision 12 either because they did not utilize TPC values or values used were not revised. See DCN 61599 for SAR/Tech Spec determination. Pages added: 2e, 23a Pages deleted: 3b Pages replaced: 1, 1a, 3, 3a, 4, 5, 7-12, 14, 16, 17, 19, 24, 25, Attachment A Revision 12: 34 total pages Attachment A – NPG Calculation Design Output (1 page)

TVA NUCLEAR CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER WBNTSR100	
Title	Design Releases to Show Compliance With 10CFR20
Revision No.	DESCRIPTION OF REVISION
13	Revision 13 is performed to remove discussion of releases using an overly conservative Tritium Production Core (TPC) H-3 source term of 10 Ci/TPBAR/yr and 2500 TPBARs. Instead, a source term consisting of a H-3 permeation rate of 5 Ci/TPBAR/yr and 1900 TPBARs loaded into the TPC is used. All associated gaseous and liquid release tables were updated. It is no longer necessary to back-calculate the maximum allowable H-3 concentration that can be released to the environment. As a result, the Special Requirement/Limiting Condition created in Revision 12 related to this subject is removed from the calculation. Pages replaced in Revision 13 contain an updated page header for Revision 13. All unaffected pages retain their original headers. CTS was reviewed for successor documents, and calculation WBNAPS3128 is affected. Other successor documents reviewed were not affected by the changes made in WBNTSR100 Revision 13 either because they did not utilize TPC values or values used were not revised. See DCN 61599 for SAR/Tech Spec determination. Pages added: 2f Pages deleted: 23a Pages revised/replaced: 1, 1a, 3, 3a, 4, 5, 7-12, 14, 16, 17, 19, 24-26 Revision 13: 34 total pages Attachment A – NPG Calculation Design Output (1 page)

TVA NUCLEAR CALCULATION VERIFICATION FORM

Calculation Identifier WBNTSR100	Revision 13
Method of verification used: 1. Design Review ☒ 2. Alternate Calculation ☐ 3. Qualification Test ☐	Verifier [signature] Date 3/7/16 Aleksandar Milicevic

Comments:
I have reviewed Calculation WBNTSR100 Revision 13 and have found the portions of the calculation revised to be technically adequate. In conducting the verification of the portions of WBNTSR100 revised as part of Revision 13, I have reviewed the inputs, computations, transcription, and results, which I have found to be complete and accurate. All comments have been resolved with the preparer.

TVA NUCLEAR COMPUTER INPUT FILE
STORAGE INFORMATION SHEET

Document WBNTSR100	Rev. 13	Plant: WBN	

Subject:
Design Releases to Show Compliance with 10CFR20

☐ Electronic storage of the input files for this calculation is not required. Comments:

☒ Input files for this calculation have been stored electronically and sufficient identifying information is provided below for each input file. (Any retrieved file requires re-verification of its contents before use.)

The Excel spreadsheet for R12 is permanently stored in FILEKEEPER # 321871
The Excel spreadsheet and Word files for selected replacement pages of R13 are permanently stored in FILEKEEPER # 338107

☐ Microfiche/eFiche

TVA NUCLEAR CALCULATION TABLE OF CONTENTS

Calculation Identifier: WBNTSR100	Revision: 13

TABLE OF CONTENTS

Purpose

The purpose of this calculation is to determine if 1% failed fuel (design fuel damage) will result in effluent releases exceeding 10CFR20 App.B Table 2 limits.

Introduction

The Standard Review Plan sections 11.2.III.2.c, 11.2.IV.3, 11.3.111.2.b, and 11.3.IV.3 (ref.1) require that the gaseous and liquid releases based on 1% failed fuel be within 10CFR20 App.B Table 2 limits. This calculation takes the expected gaseous releases from TI-534 (ref.2) and the expected liquid releases from WBNTSR093 (ref.3) and scales these releases to design (1% failed fuel) levels taken from FSAR data. Not all isotopes are used from these sources as many are insignificant and short-lived. The design release concentration for each isotope is divided by the 10CFR20 App.B Table 2 Effluent Concentration Limit (ECL). The concentration/ECL fraction is summed over all isotopes. The 10CFR20 limit requires this sum to be less than unity. Note that the Standard Review Plan requires the design of the plant to have the concentration/ECL fraction to be less than unity. In actual practice, the ODCM allows the concentration/ECL fraction to be less than 10. Since this calculation is to show the adequacy of the plant design, the acceptance criterion is for the concentration/ECL fraction to be less than unity.

Since the design (1% failed fuel) reactor coolant inventory exceeds the technical specification equivalent of 0.265 uCi/gm 1-131 dose equivalent, a liquid release case utilizing the technical specification limit for iodine (and 1% failed fuel for the remaining isotopes) is performed. This will show that the plant will meet 10CFR20 limits under maximum allowable operating conditions, because the design inventories cannot be experienced in the plant .

Three cases for gaseous release are performed based on the TI-534 modes of operation. One is for the containment portion of the gaseous effluents released via filtered purging. The other cases assume the containment is vented to the annulus, and then released via the Auxiliary Building Vent.

DCN 59397 is installing a Tritiated Water Storage Tank (TWST). This tank will receive radioactive liquid waste processed by the Mobile Demineralizers. The purpose of the TWST is to temporarily hold processed liquids until more favorable release conditions occur. The TWST does not process any liquid itself, although additional processing may occur by routing stored liquid back to the Mobile Demineralizers at the discretion of the plant. Therefore, the TWST will not increase any releases, and could potentially reduce the total amount released. Therefore, the TWST is not considered further in this calculation.

The concentration/ECL fractions due to implementing a tritium production core (TPC) in Unit 1 are determined in this calculation. Gaseous and liquid releases of tritium (H-3) are based on an average H-3 permeation rate of 5 Ci/TPBAR/yr and 1900 TPBARs loaded in the TPC.

Assumptions

1. It is assumed that the ratio of design to expected reactor coolant concentrations can be applied to the expected releases to obtain design releases.

   Technical Justification: All releases are ultimately based on the reactor coolant isotopic concentrations. It follows that any change in the reactor coolant concentrations will result in a proportional change in releases. Therefore, scaling from expected to design releases based on a design/expected reactor coolant ratio is valid.

2. For a TPC, it is assumed that the H-3 source term (1900 tritium-producing burnable absorber rods (TPBARs) each with an average H-3 permeation rate of 5 Ci/TPBAR/yr) is used for both gaseous and liquid releases.

   Technical Justification: H-3 releases follow the tritium production rate, which is linear, so the average H-3 permeation rate is appropriate. 1900 TPBARs at 5 Ci/TPBAR/yr is assumed because it yields the largest, most conservative-source terms in ref.5.

Special Requirements/Limiting Conditions

None.

Calculations:

I. Expected to Design Source Term and Design to Technical Specification Scaling

The normal gas and liquid releases (ref.2,3) are based on expected reactor coolant isotopic concentrations (ref.5). The design (1% failed fuel) concentrations are taken from reference 7. The iodine Technical Specification limit of 0.265 uCi/gm I-131dose equivalent is calculated by multiplying the equivalence factor from ref. 14 (11.348) and 0.265 µci/g by the expected concentration.. The ratio of design to expected concentrations gives a scaling factor to establish the design releases. The technical specification to design concentration gives a scaling factor to establish the maximum iodine releases possible. The scaling factors are:

TABLE 1: Expected to Design and Tech. Spec. to Design Scaling

Nuclide	Expected RCS Conc. [uCi/gm]	Design RCS Conc. [uCi/gm]	Des/Exp Ratio	Tech Spec 0.265 uCi/gm I-131 Equivalent [uCi/gm]	TechSpec/ Design Ratio [uCi/gm]
Kr-85m	1.71E-01	2.10E+00	12.28		
Kr-85	2.66E-01	8.80E+00	33.08		
Kr-87	1.61E-01	1.20E+00	7.45		
Kr-88	3.00E-01	3.70E+00	12.33		
Xe-131m	6.54E-01	1.90E+00	2.91		
Xe-133m	7.17E-02	3.10E+00	43.24		
Xe-133	2.53E+00	2.81E+02	111.07		
Xe-135m	1.39E-01	7.00E-01	5.04		
Xe-135	9.04E-01	6.30E+00	6.97		
Xe-138	1.29E-01	7.00E-01	5.43		
Br-84	1.72E-02	4.30E-02	2.50		
I-131	4.77E-02	2.50E+00	52.41	0.143	0.057
I-132	2.25E-01	9.00E-01	4.00	0.677	0.752
I-133	1.49E-01	4.00E+00	26.85	0.448	0.112
I-134	3.64E-01	6.00E-01	1.65	1.095	1.824
I-135	2.78E-01	2.20E+00	7.91	0.836	0.380
Rb-88	2.04E-01	3.70E+00	18.14		
Cs-134	7.39E-03	3.00E-01	40.60		
Cs-136	9.08E-04	1.50E-01	165.20		
Cs-137	9.79E-03	1.50E+00	153.22		
Cr-51	3.26E-03	9.50E-04	0.29		
Mn-54	1.68E-03	7.90E-04	0.47		
Fe-59	3.16E-04	1.10E-03	3.48		
Co-58	4.84E-03	2.60E-02	5.37		
Co-60	5.58E-04	7.70E-04	1.38		
Sr-89	1.47E-04	3.30E-03	22.45		
Sr-90	1.26E-05	1.70E-04	13.49		
Sr-91	1.02E-03	1.90E-03	1.86		
Y-90	1.26E-05	2.00E-04	15.87		
Y-91	5.47E-06	6.10E-03	1115.17		
Zr-95	4.10E-04	7.00E-04	1.71		
Nb-95	2.95E-04	6.90E-04	2.34		
Mo-99	6.75E-03	5.30E+00	785.19		
Te-132	1.79E-03	2.60E-01	145.25		
Ba-140	1.37E-02	4.30E-03	0.31		
La-140	2.64E-02	1.50E-03	0.06		
Ce-144	4.21E-03	3.40E-04	0.08		
Pr-144	4.21E-03	3.40E-04	0.08		

This page replaced by Revision 10

II. Gaseous Releases

The expected annual gas release is taken from TI-534 (ref.2). TI-534 presents three different modes of operation: one with containment purge, and others with containment venting to the annulus (one with no filtration, and the other with continuous venting with filters). All three options are analyzed here. The expected release is multiplied by the scaling factor determined earlier to obtain the design releases. To determine the average design concentration, this release is multiplied by the site boundary X/Q value of 1.09E-5 sec/m^3 (ref.4). To correct the units, the formula used is:

$$[\mu Ci/cc] = [Ci/yr]*X/Q[s/m^3]*[1(uCi/cc)/(Ci/m^3)/(60\ s/min*60\ min/hr*24\ hr/day*365\ day/yr)]$$

The design concentration release of each isotope is then divided by the 10CFR20 App.B Table 2 Effluent Concentration Limit (ECL). This fraction is then summed over all isotopes. The acceptance criteria is for this sum to be less than unity.

Per Assumption 2, the TPC with 1900 tritium-producing burnable absorber rods (TPBARs) at an average permeation rate of 5 Ci/TPBAR/yr contains an additional amount of tritium equal to (5 Ci/TPBAR/yr * 1900 TPBARs) 9,500 Ci/yr. Based on the methodology established in TI-534, 10% of this amount is available for gaseous release, so the additional amount of gaseous tritium for a TPC is (10% of 9,500 Ci/yr) 950 Ci/yr. Note that the total value with the TPC substitutes the H-3 (TPC) value for the non-TPC H-3. The value used in the table for H-3 consists of the tritium value activity above added to the non-TPC's expected release value.

Table 2: Gas Releases, Containment Purge Option

Nuclide	Exp. Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (µCi/cc)	10CFR20 (ECL, µCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Kr-85m	2.58E+01	1.23E+01	3.17E+02	1.10E-10	1E-7	1.10E-03	2.19E-03
Kr-85	6.99E+02	3.31E+01	2.31E+04	7.99E-09	7E-7	1.14E-02	2.28E-02
Kr-87	1.62E+01	7.45E+00	1.21E+02	4.18E-11	2E-8	2.09E-03	4.18E-03
Kr-88	3.85E+01	1.23E+01	4.75E+02	1.64E-10	9E-9	1.82E-02	3.65E-02
Xe-131m	1.19E+03	2.91E+00	3.45E+03	1.19E-09	2E-6	5.97E-04	1.19E-03
Xe-133m	4.88E+01	4.32E+01	2.11E+03	7.29E-10	6E-7	1.21E-03	2.43E-03
Xe-133	3.20E+03	1.11E+02	3.55E+05	1.23E-07	5E-7	2.46E-01	4.91E-01
Xe-135m	8.52E+00	5.04E+00	4.29E+01	1.48E-11	4E-8	3.71E-04	7.42E-04
Xe-135	1.85E+02	6.97E+00	1.29E+03	4.46E-10	7E-8	6.38E-03	1.28E-02
Xe-138	7.66E+00	5.43E+00	4.16E+01	1.44E-11	2E-8	7.19E-04	1.44E-03
Br-84	5.07E-02	2.50E+00	1.27E-01	4.38E-14	8E-8	5.48E-07	1.10E-06
I-131	1.53E-01	5.24E+01	8.03E+00	2.77E-12	2E-10	1.39E-02	2.78E-02
I-132	6.75E-01	4.00E+00	2.70E+00	9.33E-13	2E-8	4.67E-05	9.34E-05
I-133	4.58E-01	2.69E+01	1.23E+01	4.25E-12	1E-9	4.25E-03	8.51E-03
I-134	1.08E+00	1.65E+00	1.78E+00	6.14E-13	6E-8	1.02E-05	2.05E-05
I-135	8.45E-01	7.91E+00	6.69E+00	2.31E-12	6E-9	3.85E-04	7.70E-04
Cs-134	2.27E-03	4.06E+01	9.20E-02	3.18E-14	2E-10	1.59E-04	3.18E-04
Cs-136	8.01E-05	1.65E+02	1.32E-02	4.57E-15	9E-10	5.08E-06	1.02E-05
Cs-137	3.48E-03	1.53E+02	5.33E-01	1.84E-13	2E-10	9.20E-04	1.84E-03
Cr-51	5.92E-04	2.90E-01	1.73E-04	5.96E-17	3E-8	1.99E-09	3.98E-09
Mn-54	4.31E-04	4.70E-01	2.03E-04	7.01E-17	1E-9	7.01E-08	1.40E-07
Fe-59	7.70E-05	3.48E+00	2.68E-04	9.27E-17	5E-10	1.85E-07	3.71E-07
Co-58	2.32E-02	5.37E+00	1.24E-01	4.30E-14	1E-9	4.30E-05	8.60E-05
Co-60	8.74E-03	1.38E+00	1.21E-02	4.17E-15	5E-11	8.33E-05	1.67E-04
Sr-89	2.98E-03	2.25E+01	6.69E-02	2.31E-14	1E-9	2.31E-05	4.63E-05
Sr-90	1.14E-03	1.35E+01	1.54E-02	5.33E-15	6E-12	8.88E-04	1.78E-03
Zr-95	1.00E-03	1.71E+00	1.71E-03	5.92E-16	4E-10	1.48E-06	2.96E-06
Nb-95	2.45E-03	2.34E+00	5.73E-03	1.98E-15	2E-9	9.90E-07	1.98E-06
Ba-140	4.00E-04	3.10E-01	1.26E-04	4.34E-17	2E-9	2.17E-08	4.34E-08
H-3	1.39E+02	1.00E+00	1.39E+02	4.80E-11	1E-7	4.81E-04	9.62E-04
H-3 (TPC)	1.09E+03	1.00E+00	1.09E+03	3.76E-10	1E-7	3.76E-03	4.24E-03
C-14	1.12E+01	1.00E+00	1.12E+01	3.87E-12	3E-9	1.29E-03	2.58E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.18E-11	1E-8	1.18E-03	2.35E-03
total						3.11E-01	6.23E-01
total (TPC)						3.15E-01	6.26E-01

Note: The "Dual Unit Operation" column in the above calculation considers dual unit operation. Based on the evaluation done for Revision 7, the per unit concentrations are the same for both units. Therefore, the last column is twice the preceding column.
Note: Dual unit operations consider only Unit 1 with the TPC.
Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

Table 3: Gas Releases, Unfiltered Containment Vent Option

Nuclide	Exp. Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (µCi/cc)	10CFR20 (ECL, µCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Kr-85m	2.16E+01	1.23E+01	2.65E+02	9.17E-11	1E-7	9.17E-04	1.83E-03
Kr-85	6.27E+02	3.31E+01	2.07E+04	7.17E-09	7E-7	1.02E-02	2.05E-02
Kr-87	8.53E+00	7.45E+00	6.36E+01	2.20E-11	2E-8	1.10E-03	2.20E-03
Kr-88	2.66E+01	1.23E+01	3.28E+02	1.13E-10	9E-9	1.26E-02	2.52E-02
Xe-131m	1.15E+03	2.91E+00	3.34E+03	1.15E-09	2E-6	5.77E-04	1.15E-03
Xe-133m	5.81E+01	4.32E+01	2.51E+03	8.68E-10	6E-7	1.45E-03	2.89E-03
Xe-133	3.37E+03	1.11E+02	3.74E+05	1.29E-07	5E-7	2.59E-01	5.17E-01
Xe-135m	4.86E+00	5.04E+00	2.45E+01	8.46E-12	4E-8	2.12E-04	4.24E-04
Xe-135	1.99E+02	6.97E+00	1.39E+03	4.79E-10	7E-8	6.85E-03	1.37E-02
Xe-138	4.48E+00	5.43E+00	2.43E+01	8.40E-12	2E-8	4.20E-04	8.40E-04
Br-84	5.07E-02	2.50E+00	1.27E-01	4.38E-14	8E-8	5.00E-07	1.00E-06
I-131	1.67E-01	5.24E+01	8.75E+00	3.03E-12	2E-10	1.51E-02	3.03E-02
I-132	6.75E-01	4.00E+00	2.70E+00	9.33E-13	2E-8	4.67E-05	9.34E-05
I-133	4.70E-01	2.69E+01	1.26E+01	4.36E-12	1E-9	4.36E-03	8.72E-03
I-134	1.08E+00	1.65E+00	1.78E+00	6.14E-13	6E-8	1.02E-05	2.04E-05
I-135	8.52E-01	7.91E+00	6.74E+00	2.33E-12	6E-9	3.88E-04	7.77E-04
Cs-134	4.74E-03	4.06E+01	1.92E-01	6.65E-14	2E-10	3.33E-04	6.65E-04
Cs-136	3.25E-03	1.65E+02	5.37E-01	1.86E-13	9E-10	2.06E-04	4.12E-04
Cs-137	8.92E-03	1.53E+02	1.37E+00	4.72E-13	2E-10	2.36E-03	4.72E-03
Cr-51	9.70E-03	2.90E-01	2.83E-03	9.77E-16	3E-8	0.00E+00	0.00E+00
Mn-54	5.68E-03	4.70E-01	2.67E-03	9.23E-16	1E-9	9.00E-07	1.80E-06
Fe-59	2.75E-03	3.48E+00	9.57E-03	3.31E-15	5E-10	6.60E-06	1.32E-05
Co-58	4.79E-02	5.37E+00	2.57E-01	8.89E-14	1E-9	8.89E-05	1.78E-04
Co-60	1.13E-02	1.38E+00	1.56E-02	5.39E-15	5E-11	1.08E-04	2.16E-04
Sr-89	1.59E-02	2.25E+01	3.57E-01	1.23E-13	1E-9	1.23E-04	2.47E-04
Sr-90	6.29E-03	1.35E+01	8.49E-02	2.93E-14	6E-12	4.89E-03	9.78E-03
Zr-95	1.00E-03	1.71E+00	1.71E-03	5.90E-16	4E-10	1.50E-06	3.00E-06
Nb-95	4.23E-03	2.34E+00	9.89E-03	3.42E-15	2E-9	1.70E-06	3.40E-06
Ba-140	4.00E-04	3.10E-01	1.26E-04	4.34E-17	2E-9	0.00E+00	0.00E+00
H-3	1.39E+02	1.00E+00	1.39E+02	4.80E-11	1E-7	4.81E-04	9.62E-04
H-3 (TPC)	1.09E+03	1.00E+00	1.09E+03	3.76E-10	1E-7	3.76E-03	4.24E-03
C-14	1.12E+01	1.00E+00	1.12E+01	3.87E-12	3E-9	1.29E-03	2.58E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.18E-11	1E-8	1.18E-03	2.35E-03
total						3.24E-01	6.48E-01
total (TPC)						3.27E-01	6.51E-01

Note: The "Dual Unit Operation" column in the above calculation considers dual unit operation. Based on the evaluation done for Revision 7, the per unit concentrations are the same for both units. Therefore, the last column is twice the preceding column.
Note: Dual unit operations consider only Unit 1 with the TPC.
Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

Table 4: Gas Releases, Continuous Filtered Containment Vent Option

Nuclide	Exp. Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (µCi/cc)	10CFR20 (ECL, µCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Kr-85m	9.48E+00	1.23E+01	1.16E+02	4.02E-11	1E-7	4.02E-04	8.05E-04
Kr-85	6.78E+02	3.31E+01	2.24E+04	7.75E-09	7E-7	1.11E-02	2.21E-02
Kr-87	5.81E+00	7.45E+00	4.33E+01	1.50E-11	2E-8	7.48E-04	1.50E-03
Kr-88	1.32E+01	1.23E+01	1.63E+02	5.63E-11	9E-9	6.25E-03	1.25E-02
Xe-131m	1.09E+03	2.91E+00	3.18E+03	1.10E-09	2E-6	5.49E-04	1.10E-03
Xe-133m	4.31E+01	4.32E+01	1.86E+03	6.44E-10	6E-7	1.07E-03	2.15E-03
Xe-133	2.90E+03	1.11E+02	3.22E+05	1.11E-07	5E-7	2.23E-01	4.45E-01
Xe-135m	4.68E+00	5.04E+00	2.36E+01	8.15E-12	4E-8	2.04E-04	4.08E-04
Xe-135	8.88E+01	6.97E+00	6.19E+02	2.14E-10	7E-8	3.06E-03	6.11E-03
Xe-138	4.34E+00	5.43E+00	2.36E+01	8.15E-12	2E-8	4.07E-04	8.15E-04
Br-84	5.07E-02	2.50E+00	1.27E-01	4.38E-14	8E-8	5.00E-07	1.00E-06
I-131	1.53E-01	5.24E+01	8.00E+00	2.77E-12	2E-10	1.38E-02	2.77E-02
I-132	6.73E-01	4.00E+00	2.69E+00	9.30E-13	2E-8	4.65E-05	9.30E-05
I-133	4.57E-01	2.69E+01	1.23E+01	4.24E-12	1E-9	4.24E-03	8.49E-03
I-134	1.07E+00	1.65E+00	1.77E+00	6.10E-13	6E-8	1.02E-05	2.04E-05
I-135	8.42E-01	7.91E+00	6.66E+00	2.30E-12	6E-9	3.84E-04	7.67E-04
Cs-134	2.27E-03	4.06E+01	9.20E-02	3.18E-14	2E-10	1.59E-04	3.18E-04
Cs-136	8.01E-05	1.65E+02	1.32E-02	4.57E-15	9E-10	5.10E-06	1.02E-05
Cs-137	3.48E-03	1.53E+02	5.33E-01	1.84E-13	2E-10	9.20E-04	1.84E-03
Cr-51	5.92E-04	2.90E-01	1.73E-04	5.96E-17	3E-8	0.00E+00	0.00E+00
Mn-54	4.31E-04	4.70E-01	2.03E-04	7.01E-17	1E-9	1.00E-07	2.00E-07
Fe-59	7.70E-05	3.48E+00	2.68E-04	9.27E-17	5E-10	2.00E-07	4.00E-07
Co-58	2.32E-02	5.37E+00	1.24E-01	4.30E-14	1E-9	4.30E-05	8.60E-05
Co-60	8.74E-03	1.38E+00	1.21E-02	4.17E-15	5E-11	8.33E-05	1.67E-04
Sr-89	2.98E-03	2.25E+01	6.69E-02	2.31E-14	1E-9	2.31E-05	4.62E-05
Sr-90	1.14E-03	1.35E+01	1.54E-02	5.33E-15	6E-12	8.88E-04	1.78E-03
Zr-95	1.00E-03	1.71E+00	1.71E-03	5.92E-16	4E-10	1.50E-06	3.00E-06
Nb-95	2.45E-03	2.34E+00	5.73E-03	1.98E-15	2E-9	1.00E-06	2.00E-06
Ba-140	4.00E-04	3.10E-01	1.26E-04	4.34E-17	2E-9	0.00E+00	0.00E+00
H-3	1.39E+02	1.00E+00	1.39E+02	4.80E-11	1E-7	4.81E-04	9.62E-04
H-3 (TPC)	1.09E+03	1.00E+00	1.09E+03	3.76E-10	1E-7	3.76E-03	4.24E-03
C-14	1.12E+01	1.00E+00	1.12E+01	3.87E-12	3E-9	1.29E-03	2.58E-03
Ar-41	3.40E+01	1.00E+00	3.40E+01	1.18E-11	1E-8	1.18E-03	2.35E-03
total						2.70E-01	5.40E-01
total (TPC)						2.73E-01	5.43E-01

Note: The "Dual Unit Operation" column in the above calculation considers dual unit operation. Based on the evaluation done for Revision 7, the per unit concentrations are the same for both units. Therefore, the last column is twice the preceding column.
Note: Dual unit operations consider only Unit 1 with TPC.
Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

III. Liquid Releases

The expected annual liquid release is taken from WBNTSR093 (ref.3). The expected release is multiplied by the scaling factor determined earlier to obtain the design releases. To determine the average design concentration, the release is divided by volume released and the dilution flow (the minimum cooling tower blowdown flow = 20,000 gpm = 2.88E7 gal/day). The volume released is sum of all sources from WBNTSR093 = 16141.654 gal/day. Note: the WBNTSR093 flow values for the condensate resin regeneration waste is for the input streams to the condensate polishers, not the waste. The waste volume is 3400 gpd per NUREG-0017 when the waste stream is processed by the condensate polishers. The formula used for the condensate polisher processed waste is then:

[μCi/cc] = [Ci/yr]*(1e6 μCi/Ci)/((16141.654+2.88E7gal/day)*8.34 lb/gal * 453.59 g/lb * 365 day/yr)

The design concentration release of each isotope is then divided by the 10CFR20 App.B Table 2 Effluent Concentration Limit (ECL). This fraction is then summed over all isotopes. The acceptance criterion is for this sum to be less than unity. A second calculation (last column of the following table) is determined using the Technical Specification limits (0.265 μCi/gm I-131) for iodines instead of design values. The C/ECL values are determined by multiplying the Design C/ECL value by the scaling factor in Table 1. The Tech Spec C/ECLs are determined by multiplying the Design C/ECL by the appropriate Tech Spec scaling factor in Table 1. The sum of the concentration/ECL for this scenario will be for the Tech Spec iodines and the previously determined design values for all other isotopes. Again, the acceptance criterion is for the sum to be less than unity.

For a TPC, 90% of the tritium available for release to the environment is contained in the liquid releases in accordance with the methodology of WBNTSR093 (ref. 3). For a TPC with 1900 TPBARs each with an average permeation rate of 5 Ci/TPBAR/yr, the average liquid H-3 release rate is (90% * 1900 TPBARs * 5 Ci/TPBAR/yr) 8,550 Ci/yr. The use of an average annual liquid H-3 concentration is consistent with 10CFR20.1302(b)(2)(i).

Table 5: Liquid Release, No Processing

Nuclide	[1]Exp.Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (µCi/cc)	10CFR20 (ECL, µCi/cc)	**Single Unit Operation Design** C/ECL	**Dual Unit Operation Design** C/ECL	Tech Spec **Single Unit Operation** 0.265 µCi/cc	Tech Spec **Dual Unit Operation** 0.265 µCi/cc
Br-84	6.94E-04	2.50	1.73E-03	4.36E-11	4E-4	1.09E-07	2.18E-07		
I-131	1.06E+00	52.41	5.53E+01	1.39E-06	1E-6	1.39E+00	2.78E+00	7.98E-02	1.60E-01
I-132	1.23E-01	4.00	4.93E-01	1.24E-08	1E-4	1.24E-04	2.48E-04	9.32E-05	1.86E-04
I-133	8.40E-01	26.85	2.25E+01	5.67E-07	7E-6	8.10E-02	1.62E-01	9.07E-03	1.81E-02
I-134	3.23E-02	1.65	5.32E-02	1.34E-09	4E-4	3.35E-06	6.69E-06	6.10E-06	1.22E-05
I-135	4.43E-01	7.91	3.50E+00	8.80E-08	3E-5	2.93E-03	5.87E-03	1.11E-03	2.23E-03
Rb-88	1.03E-02	18.14	1.88E-01	4.72E-09	4E-4	1.18E-05	2.36E-05		
Cs-134	2.38E-01	40.60	9.64E+00	2.42E-07	9E-7	2.69E-01	5.39E-01		
Cs-136	2.39E-02	165.20	3.95E+00	9.94E-08	6E-6	1.66E-02	3.31E-02		
Cs-137	3.18E-01	153.22	4.87E+01	1.22E-06	1E-6	1.22E+00	2.45E+00		
Cr-51	1.21E-01	0.29	3.51E-02	8.82E-10	5E-4	1.76E-06	3.53E-06		
Mn-54	6.70E-02	0.47	3.15E-02	7.91E-10	3E-5	2.64E-05	5.27E-05		
Fe-59	1.40E-02	3.48	4.86E-02	1.22E-09	1E-5	1.22E-04	2.44E-04		
Co-58	2.01E-01	5.37	1.08E+00	2.72E-08	2E-5	1.36E-03	2.72E-03		
Co-60	4.01E-02	1.38	5.53E-02	1.39E-09	3E-6	4.63E-04	9.27E-04		
Sr-89	5.36E-03	22.45	1.20E-01	3.02E-09	8E-6	3.78E-04	7.56E-04		
Sr-90	4.87E-04	13.49	6.57E-03	1.65E-10	5E-7	3.30E-04	6.60E-04		
Sr-91	2.98E-03	1.86	5.54E-03	1.39E-10	2E-5	6.97E-06	1.39E-05		
Y-90	0	15.87	0	0	7E-6	0	0		
Y-91	4.75E-04	1115.17	5.30E-01	1.33E-08	8E-6	1.67E-03	3.33E-03		
Zr-95	1.62E-02	1.71	2.78E-02	6.98E-10	2E-5	3.49E-05	6.98E-05		
Nb-95	1.34E-02	2.34	3.13E-02	7.88E-10	3E-5	2.63E-05	5.25E-05		
Mo-99	1.26E-01	785.19	9.88E+01	2.48E-06	2E-5	1.24E-01	2.48E-01		
Te-132	3.64E-02	145.25	5.28E+00	1.33E-07	9E-6	1.47E-02	2.95E-02		
Ba-140	4.27E-01	0.31	1.32E-01	3.33E-09	8E-6	4.16E-04	8.31E-04		
La-140	6.14E-01	0.06	3.69E-02	9.26E-10	9E-6	1.03E-04	2.06E-04		
Ce-144	1.58E-01	0.08	1.26E-02	3.17E-10	3E-6	1.06E-04	2.11E-04		
Pr-144	0	0.08	0	0	6E-4	0	0		
H-3	1252.8	1	1252.8	3.15E-05	1E-3	3.15E-02	6.30E-02		
H-3(TPC)	9802.80	1	9802.80	2.46E-04	1E-3	2.46E-01	2.78E-01		
Total						3.16E+00	6.32E+00	1.77E+00	3.55E+00
Total(TPC)						3.37E+00	6.53E+00	1.99E+00	3.77E+00

[1] Expected release from Table 6, Case #1 (MD processing for tanks and CVCS only) of WBNTSR093 (ref.3)

Note: The "Dual Unit Operation" column in the above calculation considers dual unit operation. Based on the evaluation done for Revision 7, the per unit concentrations are the same for both units. Therefore, the last column is twice the preceding column.

Note: Dual unit operations consider only Unit 1 with TPC.

Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

The sum over all isotopes of the concentrations/ECL value from the previous table is greater than unity for the case where all isotopes are at design values. The bulk of the release is due to the condensate resin regeneration waste (untreated, ref.3). Per the ODCM (ref.10), the condensate regeneration waste will not be a continuous release if the activity in the secondary side is greater than 1E-6 uCi/gm. From WBNNAL3003 (ref.5), the activity of some of the isotopes in the secondary side is greater than 1E-6 uCi/gm. This means that the regeneration waste will be in batch mode, and monitored. For conservatism, it was assumed that it is being released continuously. From references 11, 12 and 13 the condensate regeneration waste can be rerouted through the mobile demineralizers. If the long term releases from the condensate regeneration waste is greater than the 10CFR20 concentration limits, then routing the fluid stream through the mobile demineralizers will be performed. The expected release is from Table 6, Case #2 of WBNTSR093 (ref. 3). With mobile demineralizer processing of condensate regeneration waste the release concentrations become:

This page replaced by Revision 010.

Table 6: Liquid Release, Mobile Demineralizer Processing

Nuclide	[1]Exp.Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (µCi/cc)	10CFR20 (ECL, µCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Br-84	2.26E-04	2.50	5.65E-04	1.42E-11	4E-4	3.55E-08	7.11E-08
I-131	3.70E-02	52.41	1.94E+00	4.87E-08	1E-6	4.87E-02	9.74E-02
I-132	1.81E-02	4.00	7.23E-02	1.82E-09	1E-4	1.82E-05	3.63E-05
I-133	7.29E-02	26.85	1.96E+00	4.92E-08	7E-6	7.03E-03	1.41E-02
I-134	8.57E-03	1.65	1.41E-02	3.56E-10	4E-4	8.89E-07	1.78E-06
I-135	6.52E-02	7.91	5.16E-01	1.30E-08	3E-5	4.32E-04	8.65E-04
Rb-88	9.41E-03	18.14	1.71E-01	4.29E-09	4E-4	1.07E-05	2.15E-05
Cs-134	4.02E-02	40.60	1.63E+00	4.10E-08	9E-7	4.55E-02	9.11E-02
Cs-136	3.50E-03	165.20	5.79E-01	1.45E-08	6E-6	2.42E-03	4.85E-03
Cs-137	5.52E-02	153.22	8.46E+00	2.13E-07	1E-6	2.13E-01	4.25E-01
Cr-51	9.70E-03	0.29	2.81E-03	7.07E-11	5E-4	1.41E-07	2.83E-07
Mn-54	6.87E-03	0.47	3.23E-03	8.12E-11	3E-5	2.71E-06	5.41E-06
Fe-59	3.31E-03	3.48	1.15E-02	2.90E-10	1E-5	2.90E-05	5.80E-05
Co-58	3.18E-02	5.37	1.71E-01	4.29E-09	2E-5	2.14E-04	4.29E-04
Co-60	1.97E-02	1.38	2.72E-02	6.83E-10	3E-6	2.28E-04	4.55E-04
Sr-89	2.67E-04	22.45	5.98E-03	1.50E-10	8E-6	1.88E-05	3.76E-05
Sr-90	3.04E-05	13.49	4.10E-04	1.03E-11	5E-7	2.06E-05	4.13E-05
Sr-91	3.90E-04	1.86	7.26E-04	1.82E-11	2E-5	9.12E-07	1.82E-06
Y-90	0	15.87	0	0	7E-6	0	0
Y-91	1.23E-04	1115.17	1.37E-01	3.45E-09	8E-6	4.32E-04	8.63E-04
Zr-95	1.91E-03	1.71	3.27E-03	8.22E-11	2E-5	4.11E-06	8.22E-06
Nb-95	2.88E-03	2.34	6.75E-03	1.70E-10	3E-5	5.65E-06	1.13E-05
Mo-99	5.85E-03	785.19	4.59E+00	1.15E-07	2E-5	5.77E-03	1.15E-02
Te-132	1.55E-03	145.25	2.26E-01	5.67E-09	9E-6	6.30E-04	1.26E-03
Ba-140	1.44E-02	0.31	4.46E-03	1.12E-10	8E-6	1.40E-05	2.80E-05
La-140	2.28E-02	0.06	1.37E-03	3.43E-11	9E-6	3.81E-06	7.63E-06
Ce-144	9.49E-03	0.08	7.59E-04	1.91E-11	3E-6	6.36E-06	1.27E-05
Pr-144	0	0.08	0	0	6E-4	0	0
H-3	1252.80	1	1252.80	3.149E-05	1E-3	3.15E-02	6.30E-02
H-3(TPC)	9802.80	1	9802.80	2.46E-04	1E-3	2.46E-01	2.78E-01
Total						3.56E-01	7.11E-01
Total(TPC)						5.70E-01	9.26E-01

[1] Expected release from Table 6, Case #2 (MD processing) of WBNTSR093 (ref.3)

Note: The "Dual Unit Operation" column in the above calculation considers dual unit operation. Based on the evaluation done for Revision 7, the per unit concentrations are the same for both units. Therefore, the last column is twice the preceding column.

Note: Dual unit operations consider only Unit 1 with TPC.

Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

DCN D-50502-A (ref.16) will allow the direct release of Steam Generator Blowdown to the river without Condensate Polisher Demineralizers, and no processing of the Condensate by the Condensate Polisher Demineralizers. Therefore there is no Condensate source term. Calculation WBNTSR-093 performed 5 cases. One case (Case #3) was a release with expected concentrations. That release exceeded the App. I limit of 5 Ci/unit and will not be evaluated in this calculation. A second case (Case #4) limited the release of the SGB fluid to the Lower Limit of Detection (LLD = 5E-7 uCi/cc gross gamma). The third case (Case #5) was for the SGB fluid stream limited to the maximum allowable and still meet the App.I limit of 5 Ci/unit. Because the SGB is not processed by the CD, then the volume of liquid will change as the 3400 gpd analyzed previously to the max steam generator blowdown of 262 gpm (377,280 gpd). This makes the total flow 390,021.654 gpd (16,141.654 – 3400 + 377,280 = 390,021.654). The concentration is calculated as follows:

[uCi/gm] = [Ci/yr]*(1e6 uCi/Ci)/((16141.654-3400+377,820 gpd+2.88E7gal/day)*8.34 lb/gal * 453.59 g/lb * 365 day/yr)

Tables 7 and 8 below are based on input from Table 5, of WBNTSR093 (ref. 3). In both tables the LRW values are multiplied by the "Des/Exp Ratio", and then the SGB values are added. There are several nuclides which do not have design values and therefore Des/Exp ratios were not generated. In these cases a multiplier of 1.0 has been applied in determining the design releases.

This page replaced by Revision 010.

Table 7: Direct Steam Generator Blowdown Release/SGBD at LLD with 20,000 gpm Dilution

Nuclide	[1]LRW (Ci/yr)	[2]SGB Ci/yr Scaled to 0.26 Ci	Des/Exp Ratio	Des (Ci/yr)	Liquid (μCi/cc)	Liquid 10CFR20 (ECL, μCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Br-84	2.26E-04	8.15E-04	2.50	1.38E-03	3.42E-11	4E-4	8.56E-08	1.71E-07
I-131	3.60E-02	1.20E-02	52.41	1.90E+00	4.71E-08	1E-6	4.71E-02	9.41E-02
I-132	1.80E-02	2.87E-02	4.00	1.01E-01	2.50E-09	1E-4	2.50E-05	4.99E-05
I-133	7.22E-02	3.44E-02	26.85	1.97E+00	4.89E-08	7E-6	6.99E-03	1.40E-02
I-134	8.55E-03	2.50E-02	1.65	3.91E-02	9.70E-10	4E-4	2.42E-06	4.85E-06
I-135	6.49E-02	5.28E-02	7.91	5.66E-01	1.40E-08	3E-5	4.68E-04	9.36E-04
Rb-88	9.41E-03	6.28E-03	18.14	1.77E-01	4.39E-09	4E-4	1.10E-05	2.20E-05
Cs-134	4.00E-02	3.91E-03	40.60	1.63E+00	4.04E-08	9E-7	4.48E-02	8.97E-02
Cs-136	3.48E-03	4.74E-04	165.20	5.76E-01	1.43E-08	6E-6	2.38E-03	4.76E-03
Cs-137	5.50E-02	5.21E-03	153.22	8.43E+00	2.09E-07	1E-6	2.09E-01	4.18E-01
Na-24	2.54E-02	1.59E-02	1.00	4.13E-02	1.02E-09	5E-5	2.05E-05	4.09E-05
Cr-51	9.59E-03	1.33E-03	0.29	4.11E-03	1.02E-10	5E-4	2.04E-07	4.08E-07
Mn-54	6.81E-03	6.65E-04	0.47	3.87E-03	9.59E-11	3E-5	3.20E-06	6.39E-06
Fe-55	1.11E-02	5.01E-04	1.00	1.16E-02	2.87E-10	1E-4	2.87E-06	5.73E-06
Fe-59	3.30E-03	1.23E-04	3.48	1.16E-02	2.88E-10	1E-5	2.88E-05	5.76E-05
Co-58	3.01E-02	1.94E-03	5.37	1.63E-01	4.06E-09	2E-5	2.03E-04	4.06E-04
Co-60	1.97E-02	2.25E-04	1.38	2.74E-02	6.79E-10	3E-6	2.26E-04	4.52E-04
Zn-65	5.22E-04	2.15E-04	1.00	7.37E-04	1.83E-11	5E-6	3.66E-06	7.31E-06
Sr-89	2.61E-04	5.83E-05	22.45	5.93E-03	1.47E-10	8E-6	1.84E-05	3.68E-05
Sr-90	3.00E-05	5.01E-06	13.49	4.09E-04	1.02E-11	5E-7	2.03E-05	4.06E-05
Sr-91	3.88E-04	3.00E-04	1.86	1.02E-03	2.53E-11	2E-5	1.27E-06	2.53E-06
Y-90	0	0	15.87	0	0	7E-6	0	0
Y-91m	2.30E-04	3.70E-05	1.00	2.67E-04	6.62E-12	2E-3	3.31E-09	6.62E-09
Y-91	1.23E-04	2.15E-06	1115.17	1.37E-01	3.40E-09	8E-6	4.25E-04	8.50E-04
Y-93	1.73E-03	1.28E-03	1.00	3.01E-03	7.48E-11	2E-5	3.74E-06	7.48E-06
Zr-95	1.90E-03	1.64E-04	1.71	3.41E-03	8.46E-11	2E-5	4.23E-06	8.46E-06
Nb-95	2.87E-03	1.13E-04	2.34	6.84E-03	1.70E-10	3E-5	5.65E-06	1.13E-05
Mo-99	5.73E-03	2.58E-03	785.19	4.50E+00	1.12E-07	2E-5	5.58E-03	1.12E-02
Tc-99m	4.57E-03	1.19E-03	1.00	5.77E-03	1.43E-10	1E-3	1.43E-07	2.86E-07
Ru-103	8.03E-03	3.17E-03	1.00	1.12E-02	2.78E-10	3E-5	9.27E-06	1.85E-05
Ru-106	1.04E-01	3.79E-02	1.00	1.42E-01	3.52E-09	3E-6	1.17E-03	2.35E-03
Te-129m	1.92E-04	7.98E-05	1.00	2.72E-04	6.75E-12	7E-6	9.65E-07	1.93E-06
Te-129	9.97E-04	2.52E-03	1.00	3.52E-03	8.74E-11	4E-4	2.18E-07	4.37E-07
Te-131m	1.10E-03	5.63E-04	1.00	1.66E-03	4.12E-11	8E-6	5.15E-06	1.03E-05
Te-131	2.77E-04	3.39E-04	1.00	6.16E-04	1.53E-11	8E-5	1.91E-07	3.82E-07
Te-132	1.52E-03	6.81E-04	145.25	2.21E-01	5.49E-09	9E-6	6.10E-04	1.22E-03
Ba-140	1.40E-02	5.33E-03	0.31	9.67E-03	2.40E-10	8E-6	3.00E-05	6.00E-05
La-140	2.22E-02	9.64E-03	0.06	1.10E-02	2.72E-10	9E-6	3.02E-05	6.05E-05
Ce-141	4.65E-04	6.24E-05	1.00	5.27E-04	1.31E-11	3E-5	4.36E-07	8.72E-07
Ce-143	2.08E-03	1.04E-03	1.00	3.12E-03	7.75E-11	2E-5	3.87E-06	7.75E-06
Ce-144	9.34E-03	1.64E-03	0.08	2.38E-03	5.92E-11	3E-6	1.97E-05	3.94E-05
Pr-144	0	0	0.08	0	0	6E-4	0	0
Np-239	1.88E-03	8.70E-04	1.00	2.74E-03	6.81E-11	2E-5	3.41E-06	6.81E-06
H-3	1252.80		1	1252.80	3.11E-05	1E-3	3.11E-02	6.22E-02
H-3(TPC)	9802.80		1	9802.80	2.43E-04	1E-3	2.43E-01	2.74E-01
Total							3.50E-01	7.01E-01
Total(TPC)							5.63E-01	9.13E-01

1 Source: WBNTSR093 (ref.3), Table 5, LRW only, no SGB
2 Source: WBNTSR093 (ref.3), Table 5, SGB scaled to LLD
Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

Table 8: Direct Steam Generator Blowdown Release/SGBD at Max App. I with 20,000 gpm Dilution

Nuclide	[1]LRW (Ci/yr)	[2]SGB Ci/yr Scaled to 4.40 Ci	Des/Exp Ratio	Des (Ci/yr)	Liquid (μCi/cc)	Liquid 10CFR20 (ECL, μCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Br-84	2.26E-04	1.38E-02	2.50	1.44E-02	3.56E-10	4E-4	8.91E-07	1.78E-06
I-131	3.60E-02	2.03E-01	52.41	2.09E+00	5.18E-08	1E-6	5.18E-02	1.04E-01
I-132	1.80E-02	4.86E-01	4.00	5.58E-01	1.38E-08	1E-4	1.38E-04	2.77E-04
I-133	7.22E-02	5.82E-01	26.85	2.52E+00	6.25E-08	7E-6	8.93E-03	1.79E-02
I-134	8.55E-03	4.23E-01	1.65	4.37E-01	1.08E-08	4E-4	2.71E-05	5.42E-05
I-135	6.49E-02	8.93E-01	7.91	1.41E+00	3.49E-08	3E-5	1.16E-03	2.33E-03
Rb-88	9.41E-03	1.06E-01	18.14	2.77E-01	6.87E-09	4E-4	1.72E-05	3.44E-05
Cs-134	4.00E-02	6.61E-02	40.60	1.69E+00	4.19E-08	9E-7	4.66E-02	9.31E-02
Cs-136	3.48E-03	8.02E-03	165.20	5.83E-01	1.45E-08	6E-6	2.41E-03	4.83E-03
Cs-137	5.50E-02	8.82E-02	153.22	8.51E+00	2.11E-07	1E-6	2.11E-01	4.22E-01
Na-24	2.54E-02	2.68E-01	1.00	2.94E-01	7.29E-09	5E-5	1.46E-04	2.92E-04
Cr-51	9.59E-03	2.25E-02	0.29	2.53E-02	6.28E-10	5E-4	1.26E-06	2.51E-06
Mn-54	6.81E-03	1.13E-02	0.47	1.45E-02	3.59E-10	3E-5	1.20E-05	2.39E-05
Fe-55	1.11E-02	8.49E-03	1.00	1.95E-02	4.85E-10	1E-4	4.85E-06	9.70E-06
Fe-59	3.30E-03	2.08E-03	3.48	1.36E-02	3.37E-10	1E-5	3.37E-05	6.74E-05
Co-58	3.01E-02	3.29E-02	5.37	1.94E-01	4.82E-09	2E-5	2.41E-04	4.82E-04
Co-60	1.97E-02	3.81E-03	1.38	3.09E-02	7.68E-10	3E-6	2.56E-04	5.12E-04
Zn-65	5.22E-04	3.64E-03	1.00	4.16E-03	1.03E-10	5E-6	2.06E-05	4.13E-05
Sr-89	2.61E-04	9.87E-04	22.45	6.86E-03	1.70E-10	8E-6	2.13E-05	4.25E-05
Sr-90	3.00E-05	8.49E-05	13.49	4.89E-04	1.21E-11	5E-7	2.43E-05	4.85E-05
Sr-91	3.88E-04	5.08E-03	1.86	5.80E-03	1.44E-10	2E-5	7.20E-06	1.44E-05
Y-90	0	0	15.87	0	0	7E-6	0	0
Y-91m	2.30E-04	6.26E-04	1.00	8.56E-04	2.12E-11	2E-3	1.06E-08	2.12E-08
Y-91	1.23E-04	3.64E-05	1115.17	1.37E-01	3.40E-09	8E-6	4.25E-04	8.50E-04
Y-93	1.73E-03	2.16E-02	1.00	2.34E-02	5.80E-10	2E-5	2.90E-05	5.80E-05
Zr-95	1.90E-03	2.77E-03	1.71	6.02E-03	1.49E-10	2E-5	7.46E-06	1.49E-05
Nb-95	2.87E-03	1.91E-03	2.34	8.63E-03	2.14E-10	3E-5	7.14E-06	1.43E-05
Mo-99	5.73E-03	4.37E-02	785.19	4.54E+00	1.13E-07	2E-5	5.63E-03	1.13E-02
Tc-99m	4.57E-03	2.02E-02	1.00	2.48E-02	6.15E-10	1E-3	6.15E-07	1.23E-06
Ru-103	8.03E-03	5.37E-02	1.00	6.17E-02	1.53E-09	3E-5	5.10E-05	1.02E-04
Ru-106	1.04E-01	6.41E-01	1.00	7.45E-01	1.85E-08	3E-6	6.16E-03	1.23E-02
Te-129m	1.92E-04	1.35E-03	1.00	1.54E-03	3.83E-11	7E-6	5.47E-06	1.09E-05
Te-129	9.97E-04	4.27E-02	1.00	4.37E-02	1.08E-09	4E-4	2.71E-06	5.42E-06
Te-131m	1.10E-03	9.53E-03	1.00	1.06E-02	2.64E-10	8E-6	3.29E-05	6.59E-05
Te-131	2.77E-04	5.73E-03	1.00	6.01E-03	1.49E-10	8E-5	1.86E-06	3.73E-06
Te-132	1.52E-03	1.15E-02	145.25	2.32E-01	5.76E-09	9E-6	6.40E-04	1.28E-03
Ba-140	1.40E-02	9.02E-02	0.31	9.45E-02	2.35E-09	8E-6	2.93E-04	5.86E-04
La-140	2.22E-02	1.63E-01	0.06	1.64E-01	4.08E-09	9E-6	4.53E-04	9.07E-04
Ce-141	4.65E-04	1.06E-03	1.00	1.52E-03	3.77E-11	3E-5	1.26E-06	2.52E-06
Ce-143	2.08E-03	1.76E-02	1.00	1.97E-02	4.89E-10	2E-5	2.44E-05	4.89E-05
Ce-144	9.34E-03	2.77E-02	0.08	2.85E-02	7.06E-10	3E-6	2.35E-04	4.71E-04
Pr-144	0	0	0.08	0	0	6E-4	0	0
Np-239	1.88E-03	1.47E-02	1.00	1.66E-02	4.12E-10	2E-5	2.06E-05	4.12E-05
H-3	1252.80		1	1252.80	3.11E-05	1E-3	3.11E-02	6.22E-02
H-3(TPC)	9802.80		1	9802.80	2.43E-04	1E-3	2.43E-01	2.74E-01
Total							3.68E-01	7.36E-01
Total(TPC)							5.80E-01	9.48E-01

1 Source: WBNTSR093 (ref.3), Table 5, LRW only, no SGB

2 Source: WBNTSR093 (ref.3), Table 5, SGB scaled to LLD

Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

To determine if the SGB with no processing can be released to the river without the Cooling Tower Blowdown (CTB) dilution flow of 20,000 gpm. Table 9 takes the Table 7 annual release and converts it to a concentration without the 20,000 gpm CTB dilution.

[uCi/gm] = [Ci/yr]*(1e6 uCi/Ci)/((390,021.654)*8.34 lb/gal * 453.59 g/lb * 365 day/yr)

This page replaced by Revision 010.

Table 9: Direct Steam Generator Blowdown Release/SGBD at LLD without 20,000 gpm Dilution

Nuclide	[1]Des (Ci/yr)	Liquid Des (µCi/cc)	Liquid 10CFR20 (ECL, µCi/cc)	**Single Unit Operation** C/ECL	**Dual Unit Operation** C/ECL
Br-84	1.38E-03	2.56E-09	4E-4	6.40E-06	1.28E-05
I-131	1.90E+00	3.52E-06	1E-6	3.52E+00	7.05E+00
I-132	1.01E-01	1.87E-07	1E-4	1.87E-03	3.74E-03
I-133	1.97E+00	3.66E-06	7E-6	5.23E-01	1.05E+00
I-134	3.91E-02	7.26E-08	4E-4	1.81E-04	3.63E-04
I-135	5.66E-01	1.05E-06	3E-5	3.50E-02	7.00E-02
Rb-88	1.77E-01	3.29E-07	4E-4	8.22E-04	1.64E-03
Cs-134	1.63E+00	3.02E-06	9E-7	3.36E+00	6.71E+00
Cs-136	5.76E-01	1.07E-06	6E-6	1.78E-01	3.56E-01
Cs-137	8.43E+00	1.56E-05	1E-6	1.56E+01	3.13E+01
Na-24	4.13E-02	7.66E-08	5E-5	1.53E-03	3.06E-03
Cr-51	4.11E-03	7.64E-09	5E-4	1.53E-05	3.05E-05
Mn-54	3.87E-03	7.18E-09	3E-5	2.39E-04	4.79E-04
Fe-55	1.16E-02	2.15E-08	1E-4	2.15E-04	4.29E-04
Fe-59	1.16E-02	2.16E-08	1E-5	2.16E-03	4.31E-03
Co-58	1.63E-01	3.04E-07	2E-5	1.52E-02	3.04E-02
Co-60	2.74E-02	5.08E-08	3E-6	1.69E-02	3.39E-02
Zn-65	7.37E-04	1.37E-09	5E-6	2.74E-04	5.47E-04
Sr-89	5.93E-03	1.10E-08	8E-6	1.38E-03	2.75E-03
Sr-90	4.09E-04	7.60E-10	5E-7	1.52E-03	3.04E-03
Sr-91	1.02E-03	1.90E-09	2E-5	9.48E-05	1.90E-04
Y-90	0	0	7E-6	0	0
Y-91m	2.67E-04	4.96E-10	2E-3	2.48E-07	4.96E-07
Y-91	1.37E-01	2.54E-07	8E-6	3.18E-02	6.36E-02
Y-93	3.01E-03	5.59E-09	2E-5	2.80E-04	5.59E-04
Zr-95	3.41E-03	6.33E-09	2E-5	3.17E-04	6.33E-04
Nb-95	6.84E-03	1.27E-08	3E-5	4.23E-04	8.46E-04
Mo-99	4.50E+00	8.36E-06	2E-5	4.18E-01	8.36E-01
Tc-99m	5.77E-03	1.07E-08	1E-3	1.07E-05	2.14E-05
Ru-103	1.12E-02	2.08E-08	3E-5	6.94E-04	1.39E-03
Ru-106	1.42E-01	2.64E-07	3E-6	8.79E-02	1.76E-01
Te-129m	2.72E-04	5.05E-10	7E-6	7.22E-05	1.44E-04
Te-129	3.52E-03	6.54E-09	4E-4	1.63E-05	3.27E-05
Te-131m	1.66E-03	3.08E-09	8E-6	3.86E-04	7.71E-04
Te-131	6.16E-04	1.14E-09	8E-5	1.43E-05	2.86E-05
Te-132	2.21E-01	4.11E-07	9E-6	4.56E-02	9.13E-02
Ba-140	9.67E-03	1.79E-08	8E-6	2.24E-03	4.49E-03
La-140	1.10E-02	2.04E-08	9E-6	2.26E-03	4.53E-03
Ce-141	5.27E-04	9.79E-10	3E-5	3.26E-05	6.53E-05
Ce-143	3.12E-03	5.80E-09	2E-5	2.90E-04	5.80E-04
Ce-144	2.38E-03	4.43E-09	3E-6	1.48E-03	2.95E-03
Pr-144	0	0	6E-4	0	0
Np-239	2.74E-03	5.10E-09	2E-5	2.55E-04	5.10E-04
H-3	1252.80	2.33E-03	1E-3	2.33E+00	4.65E+00
H-3(TPC)	9802.80	1.82E-02	1E-3	1.82E+01	2.05E+01
Total				2.62E+01	5.25E+01
Total(TPC)				4.21E+01	6.83E+01

[1]Design Release from Table 7

Note: Dual unit operation H-3 (TPC) value is the sum of normal H-3 from two units and the TPC-specific H-3 from one unit.

The above scenario exceeds all release limits (design and ODCM), and therefore is not to be implemented

From Table 9, as long as the normal radwaste system operates, the 10CFR20 limits will be exceeded with no SGB processing. Table 10 presents the case where the SGB is the only release and is limited to the LLD. The annual release is at the 0.26 Ci limit as determined in WBNTSR-093. The concentration is calculated as follows in the third column:

[uCi/gm] = [Ci/yr]*(1e6 uCi/Ci)/((377,280)*8.34 lb/gal * 453.59 g/lb * 365 day/yr)

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Table 10: Direct Steam Generator Blowdown Release/SGBD at LLD without 20,000 gpm Dilution/SGB is only Release Path

Nuclide	SGB at LLD (Ci/yr)	SGB at LLD (µCi/cc)	Liquid 10CFR20 (ECL, µCi/cc)	Single Unit Operation Liquid C/ECL	Dual Unit Operation Liquid C/ECL
Br-84	8.15E-04	1.57E-09	4E-4	3.91E-06	7.83E-06
I-131	1.20E-02	2.31E-08	1E-6	2.31E-02	4.62E-02
I-132	2.87E-02	5.52E-08	1E-4	5.52E-04	1.10E-03
I-133	3.44E-02	6.60E-08	7E-6	9.43E-03	1.89E-02
I-134	2.50E-02	4.80E-08	4E-4	1.20E-04	2.40E-04
I-135	5.28E-02	1.01E-07	3E-5	3.38E-03	6.76E-03
Rb-88	6.28E-03	1.20E-08	4E-4	3.01E-05	6.02E-05
Cs-134	3.91E-03	7.50E-09	9E-7	8.33E-03	1.67E-02
Cs-136	4.74E-04	9.10E-10	6E-6	1.52E-04	3.03E-04
Cs-137	5.21E-03	1.00E-08	1E-6	1.00E-02	2.00E-02
Na-24	1.59E-02	3.05E-08	5E-5	6.09E-04	1.22E-03
Cr-51	1.33E-03	2.55E-09	5E-4	5.11E-06	1.02E-05
Mn-54	6.65E-04	1.28E-09	3E-5	4.26E-05	8.51E-05
Fe-55	5.01E-04	9.63E-10	1E-4	9.63E-06	1.93E-05
Fe-59	1.23E-04	2.36E-10	1E-5	2.36E-05	4.71E-05
Co-58	1.94E-03	3.73E-09	2E-5	1.87E-04	3.73E-04
Co-60	2.25E-04	4.32E-10	3E-6	1.44E-04	2.88E-04
Zn-65	2.15E-04	4.13E-10	5E-6	8.25E-05	1.65E-04
Sr-89	5.83E-05	1.12E-10	8E-6	1.40E-05	2.80E-05
Sr-90	5.01E-06	9.63E-12	5E-7	1.93E-05	3.85E-05
Sr-91	3.00E-04	5.76E-10	2E-5	2.88E-05	5.76E-05
Y-90	0	0	7E-6	0	0
Y-91m	3.70E-05	7.11E-11	2E-3	3.55E-08	7.11E-08
Y-91	2.15E-06	4.13E-12	8E-6	5.16E-07	1.03E-06
Y-93	1.28E-03	2.46E-09	2E-5	1.23E-04	2.46E-04
Zr-95	1.64E-04	3.14E-10	2E-5	1.57E-05	3.14E-05
Nb-95	1.13E-04	2.16E-10	3E-5	7.20E-06	1.44E-05
Mo-99	2.58E-03	4.96E-09	2E-5	2.48E-04	4.96E-04
Tc-99m	1.19E-03	2.29E-09	1E-3	2.29E-06	4.58E-06
Ru-103	3.17E-03	6.09E-09	3E-5	2.03E-04	4.06E-04
Ru-106	3.79E-02	7.27E-08	3E-6	2.42E-02	4.85E-02
Te-129m	7.98E-05	1.53E-10	7E-6	2.19E-05	4.38E-05
Te-129	2.52E-03	4.85E-09	4E-4	1.21E-05	2.42E-05
Te-131m	5.63E-04	1.08E-09	8E-6	1.35E-04	2.70E-04
Te-131	3.39E-04	6.50E-10	8E-5	8.12E-06	1.62E-05
Te-132	6.81E-04	1.31E-09	9E-6	1.45E-04	2.90E-04
Ba-140	5.33E-03	1.02E-08	8E-6	1.28E-03	2.56E-03
La-140	9.64E-03	1.85E-08	9E-6	2.06E-03	4.11E-03
Ce-141	6.24E-05	1.20E-10	3E-5	3.99E-06	7.99E-06
Ce-143	1.04E-03	2.00E-09	2E-5	9.99E-05	2.00E-04
Ce-144	1.64E-03	3.14E-09	3E-6	1.05E-03	2.10E-03
Pr-144	0	0	6E-4	0	0
Np-239	8.70E-04	1.67E-09	2E-5	8.35E-05	1.67E-04
H-3	0	0	1E-3	0	0
Total	2.6E-01	5.0E-07		8.60E-02	1.72E-01

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The same analysis as in the above Table 10 is performed in Table 11, except that the SGB release is at the maximum allowable for App.I limits. The maximum App.I release is taken from Table 5 of WBNTSR093. The C/ECL value is greater than unity. The release is normalized to 10 ECL to determine the maximum concentration (μCi/cc) which would result in a 10 ECL release. The concentration is calculated as follows in the third column:

[μCi/cc] = [Ci/yr]*(1e6 μCi/Ci)/((377,280 gpd)*8.34 lb/gal * 453.59 g/lb * 365 day/yr)

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Table 11: Direct Steam Generator Blowdown Release/SGBD at Max. App. I without 20,000 gpm Dilution/SGB is only Release Path

Nuclide	Max App. I (µCi/yr)	Max App. I (µCi/cc)	Liquid 10CFR20 (ECL, µCi/cc)	Single Unit Operation Liquid C/ECL	Dual Unit Operation Liquid C/ECL	Normalized 10 C/ECL	Max µCi/cc for 10 ECL
Br-84	1.38E-02	2.65E-08	4E-4	6.62E-05	1.32E-04	4.55E-04	1.82E-07
I-131	2.03E-01	3.91E-07	1E-6	3.91E-01	7.81E-01	2.69E+00	2.69E-06
I-132	4.86E-01	9.34E-07	1E-4	9.34E-03	1.87E-02	6.42E-02	6.42E-06
I-133	5.82E-01	1.12E-06	7E-6	1.60E-01	3.19E-01	1.10E+00	7.67E-06
I-134	4.23E-01	8.12E-07	4E-4	2.03E-03	4.06E-03	1.39E-02	5.58E-06
I-135	8.93E-01	1.71E-06	3E-5	5.72E-02	1.14E-01	3.93E-01	1.18E-05
Rb-88	1.06E-01	2.04E-07	4E-4	5.10E-04	1.02E-03	3.50E-03	1.40E-06
Cs-134	6.61E-02	1.27E-07	9E-7	1.41E-01	2.82E-01	9.69E-01	8.72E-07
Cs-136	8.02E-03	1.54E-08	6E-6	2.57E-03	5.13E-03	1.76E-02	1.06E-07
Cs-137	8.82E-02	1.69E-07	1E-6	1.69E-01	3.39E-01	1.16E+00	1.16E-06
Na-24	2.68E-01	5.15E-07	5E-5	1.03E-02	2.06E-02	7.08E-02	3.54E-06
Cr-51	2.25E-02	4.32E-08	5E-4	8.64E-05	1.73E-04	5.94E-04	2.97E-07
Mn-54	1.13E-02	2.16E-08	3E-5	7.20E-04	1.44E-03	4.95E-03	1.49E-07
Fe-55	8.49E-03	1.63E-08	1E-4	1.63E-04	3.26E-04	1.12E-03	1.12E-07
Fe-59	2.08E-03	3.99E-09	1E-5	3.99E-04	7.98E-04	2.74E-03	2.74E-08
Co-58	3.29E-02	6.32E-08	2E-5	3.16E-03	6.32E-03	2.17E-02	4.34E-07
Co-60	3.81E-03	7.31E-09	3E-6	2.44E-03	4.88E-03	1.68E-02	5.03E-08
Zn-65	3.64E-03	6.98E-09	5E-6	1.40E-03	2.79E-03	9.60E-03	4.80E-08
Sr-89	9.87E-04	1.90E-09	8E-6	2.37E-04	4.74E-04	1.63E-03	1.30E-08
Sr-90	8.49E-05	1.63E-10	5E-7	3.26E-04	6.52E-04	2.24E-03	1.12E-09
Sr-91	5.08E-03	9.75E-09	2E-5	4.88E-04	9.75E-04	3.35E-03	6.70E-08
Y-90	0	0	7E-6	0	0	0	0
Y-91m	6.26E-04	1.20E-09	2E-3	6.01E-07	1.20E-06	4.13E-06	8.27E-09
Y-91	3.64E-05	6.98E-11	8E-6	8.73E-06	1.75E-05	6.00E-05	4.80E-10
Y-93	2.16E-02	4.16E-08	2E-5	2.08E-03	4.16E-03	1.43E-02	2.86E-07
Zr-95	2.77E-03	5.32E-09	2E-5	2.66E-04	5.32E-04	1.83E-03	3.66E-08
Nb-95	1.91E-03	3.66E-09	3E-5	1.22E-04	2.44E-04	8.38E-04	2.51E-08
Mo-99	4.37E-02	8.39E-08	2E-5	4.20E-03	8.39E-03	2.89E-02	5.77E-07
Tc-99m	2.02E-02	3.88E-08	1E-3	3.88E-05	7.76E-05	2.67E-04	2.67E-07
Ru-103	5.37E-02	1.03E-07	3E-5	3.44E-03	6.87E-03	2.36E-02	7.08E-07
Ru-106	6.41E-01	1.23E-06	3E-6	4.10E-01	8.20E-01	2.82E+00	8.46E-06
Te-129m	1.35E-03	2.59E-09	7E-6	3.70E-04	7.41E-04	2.55E-03	1.78E-08
Te-129	4.27E-02	8.20E-08	4E-4	2.05E-04	4.10E-04	1.41E-03	5.64E-07
Te-131m	9.53E-03	1.83E-08	8E-6	2.29E-03	4.57E-03	1.57E-02	1.26E-07
Te-131	5.73E-03	1.10E-08	8E-5	1.37E-04	2.75E-04	9.45E-04	7.56E-08
Te-132	1.15E-02	2.21E-08	9E-6	2.46E-03	4.91E-03	1.69E-02	1.52E-07
Ba-140	9.02E-02	1.73E-07	8E-6	2.16E-02	4.33E-02	1.49E-01	1.19E-06
La-140	1.63E-01	3.13E-07	9E-6	3.48E-02	6.96E-02	2.39E-01	2.15E-06
Ce-141	1.06E-03	2.03E-09	3E-5	6.76E-05	1.35E-04	4.65E-04	1.39E-08
Ce-143	1.76E-02	3.38E-08	2E-5	1.69E-03	3.38E-03	1.16E-02	2.32E-07
Ce-144	2.77E-02	5.32E-08	3E-6	1.77E-02	3.55E-02	1.22E-01	3.66E-07
Pr-144	0	0	6E-4	0	0	0	0
Np-239	1.47E-02	2.83E-08	2E-5	1.41E-03	2.83E-03	9.71E-03	1.94E-07
H-3			1E-3				
Total	4.40E+00	8.45E-06		1.45	2.91	1.00E+01	5.81E-05

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Results

The results of this calculation are:

Mode Of Release	Σ(Conc/ECL) 1 Unit (TPC)	Σ(Conc/ECL) 2 Units (TPC-U1 only)	Description - Table #
Gas	0.311 (0.315)	0.623 (0.626)	(with containment purge) - 2
	0.324 (0.327)	0.648 (0.651)	(with unfiltered containment venting) - 3
	0.270 (0.273)	0.540 (0.543)	(with continuous filtered containment venting) - 4
Liquid	3.16 (3.37)	6.32 (6.53)	(unprocessed cond.demin.waste) - 5
	1.77 (1.99)	3.55 (3.77)	(unprocessed cond.demin, iodine at 0.265 tech spec limits) - 5
	0.356 (0.570)	0.711 (0.926)	(processed cond.demin. waste) - 6
	0.350 (0.563)	0.701 (0.913)	(no SGB processing, SGB release limited to LLD) - 7
	0.368 (0.580)	0.736 (0.948)	(no SGB processing, SGB rel. limited to max App.I) - 8
	26.2 (42.1)	52.5 (68.3)	(no SGB processing, SGB at LLD, no 20,000 gpm dilution) - 9
	0.086 (0.086)	0.172 (0.172)	(SGB at LLD,SGB only release path,no 20,000gpm dilution) - 10
	1.45 (1.45)	2.91 (2.91)	(SGB at max App.I, no 20,000 gpm dilution) - 11

Maximum concentrations for no 20,000 gpm dilution, SGB only release path:
for 10 ECL release: 5.81E-05 µCi/cc – table 11
for 1 ECL release: 5.81E-06 µCi/cc

Discussion and Conclusion

The gas design release concentrations are below the 10CFR20 App.B Table 2 limits (one unit operation). The continuous filtered containment venting option results in the lowest gas release because Xe-133 does not reach the same levels in containment due to decreased residence time in containment of its parent I-133. The Xe-133 is the dominant gaseous isotope released. The liquid design release concentrations with no Condensate Polisher Demineralizer regeneration waste processing will be above the 10CFR20 limits. However, when the secondary side activity reaches 1E-6 µCi/gm the condensate regeneration waste is not continuously released, but is released in batch mode and is monitored. If the long term release of this waste is projected to result in exceeding the 10CFR20 limits, the design of the plant allows the waste to be processed by the mobile demineralizer system. With the mobile demineralizer system processing the regeneration waste, the liquid design releases are below the 10CFR20 limits. Note that the design concentrations under consideration in this calculation are significantly above the technical specification limits. For example, I-131 in the design reactor coolant is 2.5 µCi/gm (Table 1) but the technical specification is 0.265 µCi/gm I-131 equivalent. This means that the design reactor coolant has greater than 2.5 times the technical specification limits. When iodines are kept at the Technical Specification limit, and all other isotopes at design levels, then the release concentrations are below the limits. It can be concluded that the realistic release concentrations will then be less than the 10CFR20 limits. The design of the gas and liquid radwaste systems meet the requirements of 10CFR20.

Revision 4 incorporates the alternate operation mode of Steam Generator Blowdown released directly to the environment without processing by the Condensate Polisher Demineralizers, and no Condensate processing by the Condensate Polisher Demineralizers. It is concluded that under normal conditions this release could exceed the 10CFR20 limits. The concentration of the SGB release must be less than that which would exceed the App.I limit of 5 Ci/unit = 8.45E-06 µCi/cc gross gamma (Table 11), or even better, be restricted to the LLD (5E-7 uCi/cc gross gamma).

Revision 4 also evaluated releases without the 20,000 gpm Cooling Tower Blowdown dilution flow. The results indicate that this cannot be done if there are releases from sources other than the SGB. If there are no other radwaste releases, the SGB may be released with no dilution flow if the concentration is 5.81E-6 µCi/cc gross gamma or less.

The concentration/ECL fractions due to implementing a TPC in Unit 1 are determined in this calculation. Gaseous and liquid effluent releases of H-3 are based on a H-3 source term consisting of 1900 TPBARs at an average permeation rate of 5 Ci/TPBAR/yr. Conclusions for the gaseous and liquid releases are unchanged for a TPC.

References

1. NUREG-0800 R2 "Standard Review Plan"
2. TI534 R10 "Annual Routine Radioactive Airborne Releases from the Operation of One Unit"
3. WBNTSR093 R13 "Liquid Radioactive Waste Releases"
4. Memorandum RIMS# T50 950109 844
5. WBNNAL3003 R6 "Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984"
6. not used
7. WCAP 7664 R1 "Radiation Analysis Design Manual, 4-Loop Plant" RIMS# NEB 810126 316
8. System Description N3-77C-4001 R2 "Liquid Radwaste Processing System" RIMS# T29 930403 988
9. Technical Specification 3.4.16
10. Offsite Dose Calculation Manual (ODCM)
11. System Description N3-14-4002 R2 "Condensate Polishing Demineralizer System" RIMS# B26 880714 022
12. WBN CCD drawing 1-47W838-3 R11
13. WBN CCD drawing 1-47W830-7 R5
14. WBNTSR-008 R14"Control Room Operator and Offsite Dose Due to a Steam Generator Tube Rupture"
15. DCN D-50165-A
16. DCN D-50502-A
17. EDC E-50629-A
18. WAT-D-10890, Westinghouse document NDP-00-0326, " Transmittal of Waste Management Evaluation for the Watt's Bar Tritium Production Core," RIMS# T71 001204 807
19. PER 546323
20. Deleted in R11
21. Deleted in R11
22. Deleted in R11
23. DCN 59397
24. DCN 61599

TVA NUCLEAR CALCULATION DESIGN OUTPUT			
Calculation Identifier: WBNTSR100 Revision 13		Engineering Change Document	DCN 61599
Calculation Title: Design Releases to Show Compliance with 10CFR20			
[signature] Preparer – Mehran A. Mohammadian	3/7/16 Date	[signature] Verifier – Aleksandar Milicevic	3/7/16 Date
[signature] Checker – Aleksandar Milicevic	3/7/16 Date	Approval	Date

This Calculation Design Output is a compilation of the design output requirements of the referenced calculation. Review of the referenced calculation for additional design output information is not required nor allowed.
Any new design output or changes to design output portions of calculations shall be processed under the authority of the engineering change process.

The entire calculation WBNTSR100 Rev. 13 is considered design output.

There are no special requirements/limiting conditions in WBNTSR100 Rev. 13.

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1

Proposed Operating License Condition
(Markup Page)

D. The following exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Therefore, these exemptions are granted pursuant to 10 CFR 50.12.

(1) Deleted

(10) The licensee shall replace the WBN, Unit 1 upper compartment cooler cooling coils with safety-related cooling coils to eliminate a potential source of containment sump dilution during design basis events prior to increasing the number of Tritium Producing Burnable Absorber Rods (TPBARs) loaded in the reactor core above 704.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1

Watts Bar Nuclear Plant, Unit 1 Tritium Producing Burnable Absorber Rods License Amendment Request Regulatory Commitment List, Revision 4

This Enclosure provides the Watts Bar Nuclear Plant (WBN), Unit 1 Tritium Producing Burnable Absorber Rods (TPBARs) License Amendment Request (LAR) updated List of Regulatory Commitments. Changes to the list described in Enclosure 1 to this letter are indicated by a revision bar in the right-hand margin. The updated List of Regulatory Commitments provided in this Enclosure supersedes any previous WBN, Unit 1 TPBAR LAR List of Regulatory Commitments.

1. [Deleted]

2. [Regulatory Commitment replaced by Operating License Condition C(10)]

3. TVA will revise RCI-137, "Radiation Production Tritium Control Program," Table 3.1, "Tritium Action Levels," to incorporate the 0.01 Curies/kilogram (Ci/kg) (i.e., 10 µCi/g) criteria from NRC Regulatory Guide (RG) 8.32, "Criteria for Establishing a Tritium Bioassay Program," prior to increasing the number of TPBARs loaded in the reactor core above the currently allowed 704 TPBARs. The action levels will be specified by RCI-137, Table 3.1 as follows.

TRITIUM ACTION LEVELS					
Process Tritium Concentration (µCi/ml)	**DAC, DAC-hrs**	**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-hrs in 7 consecutive days	inhalation	tritium air samples with DAC-hr tracking	Urinalysis	RCDP-7, basis is 10 mrem/week. which is easily detected and

					verified by bioassay.

[Note 1] Underwater diving operations in tritiated water exceeding 0.01μCi/ml, RCDP-7 Bioassay and Internal Dose Program specifies collection and analysis of urine samples 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, and (e) whenever diving suit leakage results in skin contact with tritiated water.

[Note 2] Work activities where employees are known or may be exposed to tritium atmospheres exceeding 0.3 DAC, the collection and analysis of urine is recommended: (a) pre-job 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, and (d) upon completion of the work involving tritium exposure.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1

Watts Bar Nuclear Plant, Unit 1 Tritium Producing Burnable Absorber Rods License Amendment Request Regulatory Commitment List, Revision 4

This Enclosure provides the Watts Bar Nuclear Plant (WBN), Unit 1 Tritium Producing Burnable Absorber Rods (TPBARs) License Amendment Request (LAR) updated List of Regulatory Commitments. Changes to the list described in Enclosure 1 to this letter are indicated by a revision bar in the right-hand margin. The updated List of Regulatory Commitments provided in this Enclosure supersedes any previous WBN, Unit 1 TPBAR LAR List of Regulatory Commitments.

1. [Deleted]

2. [Regulatory Commitment replaced by Operating License Condition C(10)]

3. TVA will revise RCI-137, "Radiation Production Tritium Control Program," Table 3.1, "Tritium Action Levels," to incorporate the 0.01 Curies/kilogram (Ci/kg) (i.e., 10 µCi/g) criteria from NRC Regulatory Guide (RG) 8.32, "Criteria for Establishing a Tritium Bioassay Program," prior to increasing the number of TPBARs loaded in the reactor core above the currently allowed 704 TPBARs. The action levels will be specified by RCI-137, Table 3.1 as follows.

TRITIUM ACTION LEVELS					
Process Tritium Concentration (µCi/ml)	**DAC, DAC-hrs**	**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-hrs in 7 consecutive days	inhalation	tritium air samples with DAC-hr tracking	Urinalysis	RCDP-7, basis is 10 mrem/week. which is easily detected and

					verified by bioassay.

Note 1 Underwater diving operations in tritiated water exceeding 0.01μCi/ml, RCDP-7 Bioassay and Internal Dose Program specifies collection and analysis of urine samples 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, and (e) whenever diving suit leakage results in skin contact with tritiated water.

Note 2 Work activities where employees are known or may be exposed to tritium atmospheres exceeding 0.3 DAC, the collection and analysis of urine is recommended: (a) pre-job 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, and (d) upon completion of the work involving tritium exposure.