



1101 Market Street, Chattanooga, Tennessee 37402

CNL-15-216

December 22, 2015

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) - 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

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. The response to the request for additional information (RAI) was due November 2, 2015. The due date was subsequently extended to December 24, 2015. Enclosure 1 to this letter provides TVA's RAI response.

There is one new regulatory commitment associated with this submittal. New regulatory commitment 3 is associated with the response to Radiation Protection and Consequences Branch (ARCB) RAI 1.b. Enclosure 2 provides a complete updated commitment list that supersedes the previous commitment lists provided in the Reference 1 and Reference 6 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

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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.

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 22nd day of December 2015.

Respectfully,



J. W. Shea
Vice President, Nuclear Licensing

- Enclosures:
- 1 Tennessee Valley Authority, Watts Bar Nuclear Plant, Unit 1, Response to NRC Request for Additional Information
 2. Watts Bar Nuclear Plant, Unit 1 Tritium Producing Burnable Absorber Rods License Amendment Request Updated Regulatory Commitment List

Enclosures
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,
Response to NRC Request for Additional Information**

Contents:

NRC Request for Additional Information

Attachments:

- 1 DOE-HDBK-1079-94, "Primer on Tritium Safe Handling Practices"
2. WBNNAL3003, Revision 5, "Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984"
3. WBNTSR100, Revision 12, "Design Releases to Show Compliance with 10CFR20"

NRC Request for Additional Information

By letter dated March 31, 2015, the Tennessee Valley Authority (TVA) submitted an application for license amendment to revise the Technical Specifications to increase the maximum number of tritium producing burnable absorber rods and to delete outdated information related to the tritium production program at Watts Bar Nuclear Plant (WBN) Unit 1 (ADAMS Accession No. ML15098A446). These changes would revise TS 4.2.1, "Fuel Assemblies," TS 3.5.1 Accumulators," Surveillance Requirement (SR) 3.5.1.4, TS 3.5.4, "Refueling Water Storage Tank," and SR 3.5.4[.3]. TVA supplemented this request with letters dated in May and June 2015 (ADAMS Accession Nos. ML15147A611 and ML15167A359). In response to previous requests for information from NRC staff, TVA submitted letters dated September 14 and 25, 2015 (ADAMS Accession Nos. ML15258A204 and ML15268A568).

The staff of the Radiation Protection and Consequence Branch (ARCB) of the office of Nuclear Reactor Regulation (NRR) has reviewed the application and supplement dated June 15, 2015 (ADAMS Accession No. ML15167A359) and has determined that additional information is required to complete the review.

ARCB Request for Additional Information (RAI) 1

Supplement dated June 2015, Enclosure 1, Page 2 of 9, indicates that the radiation protection tritium control program is based in part on Regulatory guide 8.32, "Criteria for Establishing a Tritium Bioassay Program," and DOE-HDBK-1079-94, "Primer on Tritium Safe Handling Practices."

- a. Provide a copy of DOE-HDBK-1079-94, and describe how its guidance is incorporated into the actions outlined in the table on page 3 of 9.

TVA Response

A copy of DOE-HDBK-1079-94, "Primer on Tritium Safe Handling Practices," is provided as Attachment 1 to this enclosure. TVA has utilized DOE-HDBK-1079-94 since the start of the Tritium Production Program at TVA as an information resource for personnel who manage and supervise the tritium production function at TVA. Although this information was used as background information in the development of the tritium control procedures, DOE-HDBK-1079-94 was not used to establish any of the action levels in Radiological Control Instruction (RCI)-137, "Radiation Protection Tritium Control Program," as provided in the table on page 3 of 9 of TVA Supplement dated June 2015, Enclosure 1. Please note that RCI-137, Table 3.1, is being revised in response to ARCB RAI 1.b. Because this supplemental information merely clarifies the use of DOE-HDBK-1079-94 in developing TVA RCI-137, no change to the TVA letter dated June 15, 2015, is warranted.

- b. Regulatory Guide 8.32 stipulates that bioassay for tritium (H-3) be provided for individuals that work around 10 kg or more of open reactor coolant with H-3 concentrations above .01 Ci/Kg. Provide a basis for why this is not an action statement in the table on page 3 of 9.

TVA Response

TVA will revise RCI-137, "Radiation Protection Tritium Control Program," Table 3.1, "Tritium Action Levels," to incorporate the 0.01 Curies/kilogram (Ci/kg) (i.e., 10 μ Ci/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 associated 0.01 $\mu\text{Ci/ml}$ (i.e., 10 $\mu\text{Ci/g}$) action level will be specified by RCI-137, Table 3.1. In addition, RCI-137, Table 3.1 will be simplified as follows.

TRITIUM ACTION LEVELS					
Process Tritium Concentration ($\mu\text{Ci/ml}$)	DAC, DAC-hrs	Mode of Exposure	Tritium Survey Requirements	Recommended 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
≥ 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 shall be requested for any employee who exceeds this limit	RCDP-7, basis is 10 mrem/week, which is easily detected and verified by bioassay.

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

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

The proposed RCI-137, Table 3.1 simplifies the information provided in response to the NRC Request to Supplement E2-1 in TVA letter dated June 15, 2015.

TVA procedure RCI-137 along with TVA procedures NPG-SPP-05.1, "Radiological Controls," and RCDP-7, "Bioassay and Internal Dose Program," provide a graded approach for bioassay based on risk, work, and airborne conditions. TVA's program establishes criteria for performing in vitro bioassay: (1) based on process water concentrations using the guidance in RCI-137, Table 3.1, (2) when an individual worker's exposure is greater than or equal to four Derived Air Concentration (DAC)-hours (hrs) in seven consecutive days, and (3) for skin contamination with tritiated water concentrations exceeding 0.01 $\mu\text{Ci/ml}$. Tritium DAC-hr tracking is initiated whenever a worker is in an airborne area, exposed to concentrations of 0.3 DAC or greater. TVA procedures cover types of bioassay, selection of individuals for bioassay, bioassay collection, sample volume, sample storage, packaging and shipping, detection limits, and internal dose calculation methods. TVA requires a Minimum Detectable Activity (MDA) of less than $1\text{E}+04$ pCi/L (i.e., 0.01 $\mu\text{Ci/L}$) tritium for urine bioassays. This MDA is less than the detection limit of 0.3 $\mu\text{Ci/L}$ specified in ANSI N13.14 and for most

conditions (e.g., bioassay obtained within two weeks of exposure) will achieve detection of an intake resulting in a committed effective dose equivalent less than 1 mrem. The frequency of bioassays is determined based on the work, the exposure scenario, and trigger levels as described in RCDP-7. Baseline tritium bioassays are obtained for all divers prior to the first on-site dive. Baseline (pre-job) tritium bioassays are also required for work activities where workers are known or may be exposed to tritium atmospheres exceeding the trigger levels of greater than 0.3 DAC. All data (i.e., bioassay, air samples with DAC-hr tracking, and whole body counts) are reviewed and evaluated to arrive at the best estimate of the worker's intake and Committed Effective Dose Equivalent (CEDE) for each radionuclide. In some cases, a single bioassay is enough to evaluate the exposure; in other cases, multiple bioassays are obtained for work in tritium airborne conditions that may continue for days or weeks.

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

ARCB RAI 2

Supplement dated June 2015, Enclosure 2, page 33, provides estimated doses to the maximally exposed member of the public in Table 10. Verify that these doses were calculated using the methods, assumptions, and input parameters consistent with the latest version of the Watts Bar Offsite Dose Calculation Manual (ODCM). If this is not the case provide the calculations and calculational basis for these estimated doses.

TVA Response

The calculated doses provided in Enclosure 2 of the June 15, 2015, TVA supplement were calculated using the methods and assumptions in Revision 24 of the WBN ODCM using the realistic source term data as input. Revision 25 of the WBN ODCM has since been issued. The changes made to the document did not affect the calculation results. Both revisions of the WBN ODCM were provided to the NRC in a letter dated May 1, 2015, "Watts Bar Nuclear Plant Unit 1, Annual Radioactive Effluent Release Report - 2014" (ML15121A826).

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

The 26,889 Ci/yr of tritium value was used to demonstrate the adequacy of the Radioactive (Radwaste) System. TVA does not intend to release 26,889 Ci/yr of tritium.

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. In no case is the tritium release from WBN, Unit 1 projected to exceed the established regulatory limits.

The information provided in Enclosure 2 of the June 2015 supplement on page 9 is revised as follows with deleted text lined out:

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 record¹⁰, 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 2,500 TPBARs (Design Basis) 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 waste¹¹. ~~The Radwaste System Design Basis tritium sources are estimated to increase the amount of tritium that is discharged annually by a factor of about fourteen.~~
- 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). ~~However, Radwaste System Design Basis tritium sources with 2,500 TPBARs are expected to increase the total amount of tritium that is generated in the plant by a factor of fourteen (i.e., from about 1,889 curies per year to approximately 26,889 curies per year).~~

The information provided in Enclosure 2 of the June 2015 supplement on page 23 is revised as follows with deleted text lined out and inserted text underlined:

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 max~~ just less than 7.0 μCi/gm. With the assumption of routine boron control, and 2,500 TPBARs at 10 Ci/TPBAR/year, the ~~estimated~~ average Design Basis RCS tritium concentration is ~~calculated~~estimated to be 29.8 μCi/gm. With the assumption of 1,900 TPBARs at 5 Ci/TPBAR/year, the ~~estimated~~ average Realistic Basis RCS tritium concentration is ~~calculated~~estimated to be 12 μCi/gm³².

and

The ~~calculated~~projected 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 fourteen increase over the Non-TPC tritium production rate, that is,

$$\text{Ratio} = (\text{TPC}) 26,889 \text{ Ci/year} / (\text{Nominal Core}) 1,889 \text{ Ci/year} = 14.2$$

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.

The information provided in Enclosure 2 of the June 2015 supplement on page 24 is revised as follows with deleted text lined out and inserted text underlined:

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 10 Ci/TPBAR/year, which bounds the realistic permeation rate with 100% margin. This source term also has increased margin by assuming ~40% more TPBARs are loaded. 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 2,500 TPBARs (Design Basis) 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 waste³⁶. The Radwaste System Design Basis tritium sources are ~~assumed~~ ~~estimated~~ to increase ~~the amount of tritium that is discharged annually~~ by a factor of about 14. The analyzed gaseous releases are based on the radwaste system design basis tritium source term. For liquid releases, ~~at~~ the maximum allowable liquid concentration of tritium released to the environment was determined.

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 5, "Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984," is provided as Attachment 2 to this enclosure. 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 5

Supplement dated June 2015, Enclosure 2, page 25 implies that the radwaste mobile demineralizer is effective for processing H-3 from liquid radwaste. Provide the decontamination factor expected (and basis), or clarify this text.

TVA Response

The discussion of the radwaste system operation provided in Enclosure 2 of the June 15, 2015, TVA supplement describes the overall system operation with an effluent stream containing all of the expected radionuclides. It does not suggest that the system is capable of processing tritium. Tables 5, 6, and 7 provided in the Enclosure 2 of the June 15, 2015, TVA supplement are updates to the information presented in Updated Final Safety Analysis Report (UFSAR) Tables 11.2-4a, 11.2-4b, and 11.2-4c to reflect the requested TPBAR loading increase. Information on the radwaste system processing capabilities is provided in UFSAR Table 11.2-4. The decontamination factor provided for tritium is 1, which means no decontamination.

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 12, "Design Releases to Show Compliance with 10CFR20," is provided as Attachment 3 to this enclosure. The purpose of this calculation is to determine if 1% failed fuel (design fuel damage) would result in effluent releases exceeding 10 CFR Part 20 Appendix B Table 2 limits. This calculation takes the expected liquid releases and scales these releases to design levels (i.e., 1% failed fuel) taken from UFSAR data. The 10 CFR Part 20 limit requires this sum to not exceed unity. For liquid releases of tritium, a maximum allowable liquid concentration of tritium is back-calculated based on the available margin for the liquid release scenarios.

- 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 administers the maximum allowable tritium concentration in accordance with WBN, Unit 1 TS 5.7.2.3, "Offsite Dose Calculation Manual," and 5.2.7.7, "Radioactive Effluent Controls Program." The ODCM specifies that a sum of the ratios calculation be performed to demonstrate compliance with TS 5.7.2.7.b. The sum of the ratios (R) for each release point is determined by the following relationship.

$$R = \sum_i \frac{C_i}{ECL_i}$$

where:

R = the sum of the ratios for the release point.

C_i = concentration of radionuclide i, μCi per milliliter (ml).

ECL_i = the ECL of radionuclide i, μCi/ml, from 10 CFR Part 20, Appendix B, Table 2, Column 2.

The sum of the ECL ratios must be ≤ 10 following dilution due to the releases from any or all of the release points described above.

The permitted maximum concentration of any single nuclide (including tritium) is unique to each scheduled release, because it is a function of the sum total of radionuclide activity present in the release and the respective ECLs.

- 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

WBN calculation WBNTSR100, provided as Attachment 3 to this enclosure, demonstrates that the WBN radwaste processing systems are capable of controlling releases in compliance with 10 CFR Part 20. Tables 6 and 7 in Enclosure 2 of the June 15, 2015, TVA supplement show aggregate concentration/ECLs that are ≤ 1 if the tritium concentration after dilution is $< 3.26E-04 \mu\text{Ci/cc}$, which is equivalent to releasing approximately 13,145 Ci/yr of tritium. This tritium release represents approximately 55% of the design basis source term. Therefore, an amount representing approximately 45% of the design basis source term would need to be held up. The majority of the tritium comes from the Chemical Volume Control System (CVCS) letdown and primary coolant equipment drains, which accounts for approximately 2,512 gpd. As stated above, approximately 45% of the release volume (i.e., approximately 413,000 gal) would be held up for later release. The Tritiated Water Storage Tank (TWST) has a capacity of 500,000 gal and therefore can hold this volume until more favorable release conditions exist.

Because most WBN liquid releases are performed in a batch mode, a more conservative evaluation is based on instantaneous discharge. As discussed in Enclosure 2 of the June 15, 2015, TVA supplement, the peak RCS tritium concentration is expected to be approximately 2.5 times greater than the average. For the design basis source term, this would result in a peak concentration of approximately $74.5 \mu\text{Ci/gm}$. The table below provides the quarterly waste volume, dilution volume, and radioactive waste volume reported in the annual radioactive effluent release reports (ARERRs) for calendar years 2008 through 2014. The last column provides the average dilution factor for the quarter by taking the ratio of the total volume of water released (waste + dilution) and the radioactive waste volume released. As shown in the table below, more than half of the quarters for calendar years 2008 through 2014 had an average dilution factor greater than $1E+04$, which results in a peak concentration after dilution of $7.5 E-02 \mu\text{Ci/gm}$ and concentration/ECL of 7.5. Consistent with the evaluation in the original DOE topical report and approval in NUREG-1672, the WBN ODCM allows the sum of the concentrations to be less than 10 times the concentration values specified in 10 CFR 20, Appendix B, Table 2, Column 2. For time periods where a dilution factor of $1E+04$ is not achieved, the TWST can be used to holdup those effluents until more favorable release conditions exist.

ARERR Report Period	Total Waste Volume Released (Liters)	Volume of Dilution Water Used (Liters)	Radioactive Waste Volume Released (Liters)	Dilution Factor
2008				
Q1	1.14E+08	1.12E+10	2.36E+06	4.79E+03
Q2	1.50E+08	1.10E+10	1.44E+06	7.74E+03
Q3	1.56E+08	2.43E+10	1.84E+06	1.33E+04
Q4	1.64E+08	1.01E+10	1.10E+06	9.33E+03
2009				
Q1	2.68E+08	6.46E+09	5.13E+05	1.31E+04
Q2	1.82E+08	1.64E+10	1.12E+06	1.48E+04
Q3	2.68E+08	1.57E+10	1.58E+06	1.01E+04
Q4	1.82E+08	5.45E+09	1.03E+06	5.47E+03
2010				
Q1	2.49E+08	8.64E+09	3.47E+05	2.56E+04
Q2	2.87E+08	1.48E+10	1.32E+06	1.14E+04
Q3	2.38E+08	1.77E+10	3.84E+06	4.67E+03
Q4	1.83E+08	1.27E+10	2.65E+06	4.86E+03
2011				
Q1	1.52E+08	5.48E+08	1.16E+06	6.03E+02
Q2	1.87E+08	5.46E+09	1.62E+06	3.49E+03
Q3	2.88E+08	1.90E+10	1.32E+06	1.46E+04
Q4	2.70E+08	9.28E+09	1.44E+06	6.63E+03
2012				
Q1	1.13E+10	3.13E+08	5.95E+05	1.95E+04
Q2	2.31E+09	9.20E+09	7.07E+05	1.63E+04
Q3	2.08E+08	9.55E+09	1.93E+06	5.06E+03
Q4	2.70E+08	1.18E+10	8.62E+05	1.40E+04
2013				
Q1	2.81E+08	2.42E+10	5.95E+05	4.11E+04
Q2	2.39E+08	1.81E+10	8.69E+05	2.11E+04
Q3	1.47E+08	1.14E+10	1.22E+06	9.46E+03
Q4	1.69E+08	2.20E+10	4.06E+06	5.46E+03
2014				
Q1	1.37E+08	1.19E+10	1.57E+06	7.67E+03
Q2	1.77E+08	1.56E+10	1.49E+06	1.06E+04
Q3	9.85E+07	4.93E+10	9.95E+05	4.96E+04
Q4	1.19E+08	1.37E+10	8.18E+05	1.69E+04

- 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

Tables 6 and 7 in Enclosure 2 of the June 15, 2015, TVA supplement show aggregate ECLs that are ≤ 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 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 5.2.7.7.

ARCB RAI 7

Supplement dated June 2015, Enclosure 2, page 7, indicates that the revised design basis source term no longer assumes two failed TPBARs. The source term used to design the reactor facility radwaste systems should include anticipated operational occurrence (AOOs) as well as normal operations. Provide a basis for not including this previously included AOO in the source term assumptions.

TVA Response

The calculations developed for the initial tritium program for the routine effluent realistic source term, the radwaste system design basis source term, and the non-loss of coolant accident

(LOCA) source terms (realistic and design basis) used the conservative assumption of tritium from two TPBAR failures in addition to the tritium permeation.

TVA reevaluated the source term assumptions for the March 31, 2015, LAR. TVA noted that Department of Energy (DOE) document NDP-98-181, Revision 1, "Tritium Production Core (TPC) Topical Report," Section 2.11.2.2 characterized the failure of a TPBAR to be beyond a reasonable design basis consideration and not expected to occur during normal operation or for any Anticipated Operational Occurrence (AOO). TVA has accumulated significant experience with TPBAR irradiation without experiencing a TPBAR failure.

TVA concluded that incorporating an assumption of two TPBAR failures for the routine effluent realistic source term is inconsistent with the regulatory intent of this source term. As such, TVA removed the assumption of two failed TPBARs from this source term. The TPC realistic source term has a tritium component that is conservatively based on an assumed tritium permeation of 5 Ci/yr from 1,900 TPBARs. The source term is conservative in that it assumes: 1) a permeation rate that is ~50% more than the expected permeation rate for the Mark 9.2 TPBAR and 2) a number of TPBARs that is ~6% more than the 1,792 TPBARs requested by the March 31, 2015, LAR.

TVA also concluded that incorporating an assumption of two TPBAR failures for the radwaste system design basis source term is not appropriate, given the definition of an AOO. 10 CFR Part 50, Appendix A defines AOOs as those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. Therefore, TVA removed the assumption of two failed TPBARs from this source term. The TPC radwaste system design basis source term has a tritium component that is conservatively based on an assumed tritium permeation of 10 Ci/yr from 2,500 TPBARs. The source term is conservative in that it assumes: 1) a permeation rate that is almost three times more than the expected permeation rate for the Mark 9.2 TPBAR and 2) a number of TPBARs that is ~40% more than the 1,792 TPBARs requested by the March 31, 2015, LAR. These conservatisms represent a reasonable source term for evaluating the capability (i.e., sizing) of the radwaste system to handle off normal events.

TVA retained the assumption of two TPBAR failures for the non-LOCA accident source term. This source term has a tritium component that is conservatively based on an assumed tritium permeation of 10 Ci/yr from 2,500 TPBARs. The source term is conservative in that it assumes: 1) a permeation rate that is almost three times more than the expected permeation rate for the Mark 9.2 TPBAR and 2) a number of TPBARs that is ~40% more than the 1,792 TPBARs requested by the March 31, 2015, LAR. The assumption of two TPBAR failures is conservative given the DOE failure assessment in NDP-98-181, Revision 1 and the operational experience at WBN. The non-LOCA accident source term provides sufficient margin to bound the TPC for Chapter 15 AOO and Postulated Accident dose consequence evaluations.

ARCB RAI 8

Supplement dated June 2015, Describe the measures (including onsite surveillance and groundwater monitoring) that TVA will employ to insure that Watts Bar operations with this increased H-3 source term will be within the requirements of NRC Bulletin No. 80-10, "CONTAMINATION OF NONRADIOACTIVE SYSTEM AND RESULTING POTENTIAL FOR

UNMONITORED, UNCONTROLLED RELEASE OF RADIOACTIVITY TO ENVIRONMENT," and 10 CFR 20.1406.

TVA Response

Based on a September 28, 2015, conference call with the NRC, TVA understands that ARCB RAI 8 does not request a response to NRC Bulletin 80-10 for WBN. Instead, the RAI is requesting an assessment of the effects of the LAR on previous assessments of the potential for contamination of groundwater or non-radioactive water systems.

TVA provided a summary of the groundwater monitoring program for WBN, Unit 1 in the letter to NRC dated August 4, 2006, "Groundwater Protection - Data Collection Questionnaire for Browns Ferry, Sequoyah, and Watts Bar Nuclear Plants" (ML062280598). The letter was submitted as a result of an industry initiative sponsored by the Nuclear Energy Institute (NEI) to compile baseline information about the status of site programs for monitoring and protecting groundwater and to share that information with NRC.

WBN, Unit 1 is committed to the industry initiative described in NEI 07-07, "Ground Water Protection Initiative," (ML072610036) as documented in UFSAR Section 1.1.3. NEI 07-07 identifies actions to improve utility management and response to instances where the inadvertent release of radioactive substances may result in low but detectible levels of plant-related materials in subsurface soils and water.

The most recent results of the TVA groundwater monitoring program were provided to the NRC in the letter dated May 1, 2015, "Watts Bar Nuclear Plant Unit 1, Annual Radioactive Effluent Release Report - 2014" (ML15121A826).

UFSAR Section 9.2.4 describes the plant features for the Potable and Sanitary Water Systems. UFSAR Section 9.2.4.1.2, "Safety Evaluation," states that the potable water system is not cross-connected with any radioactive system. UFSAR Section 9.2.4.2.3, "Safety Evaluation," states that the sanitary water system does not receive radioactive waste. The NRC review of the Potable and Sanitary Water Systems is documented in NUREG-0847, "Safety Evaluation Report related to the operation of Watts Bar Nuclear Plant, Units 1 and 2" (ML072060490), dated June 1982.

NDP-98-181, Revision 1 addresses the Potable and Sanitary Water Systems as follows:

SRP 9.2.4

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

In NUREG 1672, "Safety Evaluation Report related to the Department of Energy's Topical Report on the Tritium Production Core," dated March 1999, the NRC made the following decision regarding NDP-98-181:

Potable and Sanitary Water Systems (SRP Section 9.2.4)

DOE evaluated the effect of TPBARs on the Potable and sanitary water systems for the reference plant against the guidance of SRP Section 9.2.4 and concludes that there is no effect on this system. The staff agrees with this evaluation.

These conclusions are consistent with the tritium program operational experience at WBN, Unit 1 and remain valid for the proposed operation with 1,792 TPBARs requested by the March 31, 2015, LAR.

For systems that are considered as nonradioactive, but could possibly become radioactive through interfaces with radioactive systems, WBN Chemistry Manual, Chapter 3.01, "System Chemistry Specifications," establishes the following routine sampling/analysis or monitoring program.

Refueling Water Storage Tank	Tritium checked per request (associated with maintenance or troubleshooting)
Component Cooling Water System	Tritium checked per request (associated with maintenance or troubleshooting)
Spent Fuel Pool	Tritium checked quarterly
Feedwater	Tritium checked weekly
Demineralized Water Head Tank	Tritium checked monthly
Condensate Storage Tanks	Tritium checked per request (associated with maintenance or troubleshooting)
Auxiliary Boiler	Tritium checked monthly during operation
Raw Cooling Water	Tritium checked quarterly
Yard Holding Pond	Tritium checked weekly
Low Volume Waste Treatment Pond	Tritium checked weekly
Potable Water System	Tritium checked weekly

The system features, monitoring program, and commitments described in the response to ARCB RAI 8 are not changed by the proposed LAR.

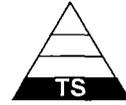
ENCLOSURE 1

**TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1**

TVA Response to NRC Request for Additional Information

Attachment 1

DOE-HDBK-1079-94, "Primer on Tritium Safe Handling Practices"



METRIC

DOE-HDBK-1079-94
December 1994

DOE HANDBOOK

PRIMER ON TRITIUM SAFE HANDLING PRACTICES



U.S. Department of Energy
Washington, D.C. 20585

FSC-6910

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FOREWORD

The *Primer on Tritium Safe Handling Practices* is approved for use by all DOE Components. It was developed as an educational supplement and reference for operations and maintenance personnel involved in tritium handling. This Primer is intended to help operations and maintenance personnel perform their duties safely by increasing their knowledge of tritium handling and control, thus reducing the likelihood of tritium releases and exposures, and by providing basic emergency responses in the case of a tritium release.

The Department of Energy (DOE) Primers are a set of fundamental handbooks on safety-related topics of interest in the DOE Complex. The Primers are written as an educational aid for operations and maintenance personnel. The Primers attempt to supply information in an easily understandable form which will help them perform their duties in a safe and reliable manner. Persons trained in other technical areas may also find the Primers useful as a guide or as a reference source for further investigation.

The DOE Primer series draws heavily upon the subject-specific Primers and training materials previously developed by DOE sites (Savannah River, Rocky Flats, and Mound) and is intended for distribution to all DOE contractors. Information is also drawn from the applicable volumes of the *DOE Fundamentals Handbook* series developed by the DOE Office of Nuclear Safety Policy and Standards. References to other material sources are indicated in the text where applicable and a bibliography is included.

Beneficial comments in the form of recommendations and any pertinent data that may be of use in improving this document should be addressed to

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by using the U.S. Department of Energy Standardization Document Improvement Proposal Form (DOE F 1300.x) appearing at the end of this document or by letter.

Key Words: Hydrogen, Tritium, Half-life, Radiation, Beta particles, Contamination, Health Physics, Monitoring, Bioassay, Metabolism, Emergency Response

OVERVIEW

Tritium Safe Handling Practices was prepared as an information resource for personnel who perform tritium handling functions at DOE facilities. A basic understanding of the properties and hazards associated with tritium will help personnel understand the impact of their actions on the safe and reliable operation of facility systems.

Tritium Safe Handling Practices contains an introduction and sections on the following topics:

- Radiological Fundamentals
- Physical and Chemical Properties of Tritium
- Biological Properties of Tritium
- Tritium Monitoring
- Radiological Control and Protection Practices
- Emergency Response

This Primer is provided as an information resource only, and is not intended to replace any radiation worker or hazardous materials training. The Primer presents the theoretical concepts and good practices that form the basis of safe tritium handling.

Specific references have been cited as footnotes, and a bibliography is available at the end of the Primer. These sources provide further reading and specific guidance. This document contains selected information from the *Health Physics Manual of Good Practices for Tritium Facilities*, which was prepared by EG&G Mound Applied Technologies at the Mound tritium facility. The authors acknowledge their expertise and experience.

**Department of Energy
Handbook**

**PRIMER ON
TRITIUM SAFE HANDLING
PRACTICES**

DOE-HDBK-1079-94
TRITIUM SAFE HANDLING PRACTICES

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INTRODUCTION

This Primer is designed for use by operations and maintenance personnel to improve their knowledge of tritium safe handling practices. It is applicable to many job classifications and can be used as a reference for classroom work or for self-study. It is presented in general terms so that it can be used throughout the DOE Complex.

The information in this Primer should enable the reader to do the following:

- Describe methods of measuring airborne tritium concentration.
- List the types of protective clothing that are effective against tritium uptake from surface and airborne contamination.
- Name two methods of reducing the body dose after a tritium uptake.
- Describe the most common method for determining the amount of tritium uptake in the body.
- Describe the steps to take following an accidental release of airborne tritium.
- Describe the damage to metals that results from absorption of tritium.
- Explain how washing hands or showering in cold water helps reduce tritium uptake.
- Describe how tritium exchanges with normal hydrogen in water and hydrocarbons.

The organization of the Primer is as indicated in the Overview. The following section contains background information on "Radiological Fundamentals." Those familiar with these topics may elect to skip this section and begin reading at the section entitled "Physical and Chemical Properties of Tritium." Additional information about tritium is available from the sources listed in the "Bibliography" section.

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RADIOLOGICAL FUNDAMENTALS

This section provides a review of radiological fundamentals. The reader is assumed to be familiar with this information from radiological worker training. The section discusses hydrogen and its isotopes and describes basic radiological concepts.

Hydrogen and Its Isotopes

Atomic nuclei of a particular element (such as hydrogen or oxygen) have the same number of protons (positively charged), but may have a different number of neutrons (no net charge). Those that have a different number of neutrons are isotopes of that element. Most elements exist in nature in several isotopic forms. For example, hydrogen has one proton. The isotopes of hydrogen either have no neutrons (normal hydrogen, called protium), one neutron (deuterium), or two neutrons (tritium) (Figure 1). Although isotopes of an element have almost the same chemical properties, the nuclear properties can be quite different.

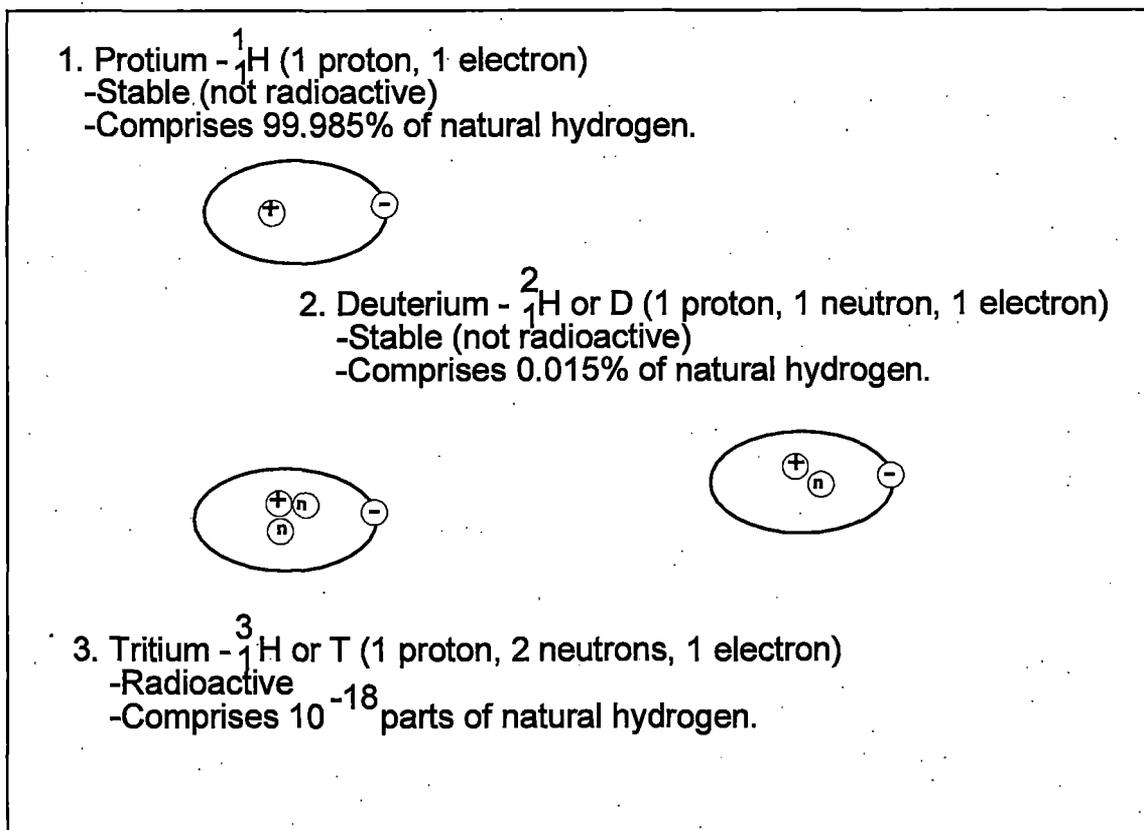


Figure 1 Hydrogen isotopes

Nuclear notation uses the chemical symbol (H for hydrogen) and an arrangement of subscripts and superscripts. The total number of protons and neutrons is shown as a superscript: ^1H for protium, ^2H for deuterium, and ^3H for tritium. The number of protons (which identifies the element) is shown as a subscript. However, the common practice of using H, D and T for these isotopes, respectively, will be followed in this document, except where nuclear reactions are illustrated.

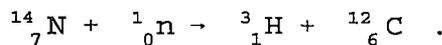
The atomic masses, symbols, and natural abundances of the three isotopes of hydrogen are given in Table 1.

Table 1 Hydrogen isotopes

Symbol		Name	Natural Abundance (%)	Mass (mass units)
Physical	Common			
^1_1H	H	Protium	99.985	1.007825
^2_1H	D	Deuterium	0.015	2.01400
^3_1H	T	Tritium	1×10^{-18}	3.01605

Sources of Tritium

Tritium occurs naturally in the environment. Reactions between cosmic radiation and gases in the upper atmosphere produce most of the world's natural tritium. For example,



Tritium converts into water and reaches the earth's surface as rain. An estimated production rate of 4×10^6 Ci/yr results in a world steady-state natural inventory of $\sim 70 \times 10^6$ Ci.^a

In addition, commercial producers of radioluminescent and neutron generator devices release about 1×10^6 Ci/yr. Atmospheric nuclear test explosions from 1945 to 1975 added about 8×10^9 Ci of tritium to the environment, much of which has since decayed. However, about 5×10^8 Ci remain in the environment, mostly diluted in the oceans. Underground nuclear tests appear to add little tritium to the atmosphere. The nuclear power and defense industries now

a. The curie (Ci) is a unit of activity defined as 3.7×10^{10} disintegrations per second (dps). A more basic unit is 1 dps, which is the definition of the becquerel (Bq). Throughout this Primer, the curie will be used instead of the becquerel.

release $1-2 \times 10^6$ Ci/yr, a small fraction of which comes from light-water reactors. Tritium is also a by-product of light-water and heavy-water nuclear reactor operation. In their coolants, these reactors produce about 500 to 1,000 and 2×10^6 Ci/yr, respectively, for every 1,000 MW(e) of power. Tritium is a fission product within nuclear fuel, generated at a rate of $1-2 \times 10^4$ Ci per year/1000 MW(e). U.S. DOE reactors have produced tritium by the neutron bombardment of lithium (${}^6\text{Li}$).

Stable and Unstable Nuclides

In the lighter elements, the ratio of neutrons to protons in their stable nuclei is usually 1 (one neutron for every proton). For the heavier elements, this ratio gradually increases to about 1.5 (1.5 neutrons for every proton). Although one cannot always predict from its ratio whether an isotope is stable or unstable, the relationship between the number of protons and neutrons is extremely important.

When an isotope is unstable, its nucleus will emit rays or particles or it may split into two different nuclei. Some combinations of neutrons and protons lead to stable nuclei. If there are too many or too few neutrons, the resulting nucleus is not stable. This unstable nucleus tries to become more stable by releasing excess energy. Atoms with unstable nuclei are *radioactive*. The process of nuclei releasing this energy is referred to as *radioactive decay* or *disintegration* (Figure 2). If a nucleus is still unstable after radioactive decay, further decay will occur.

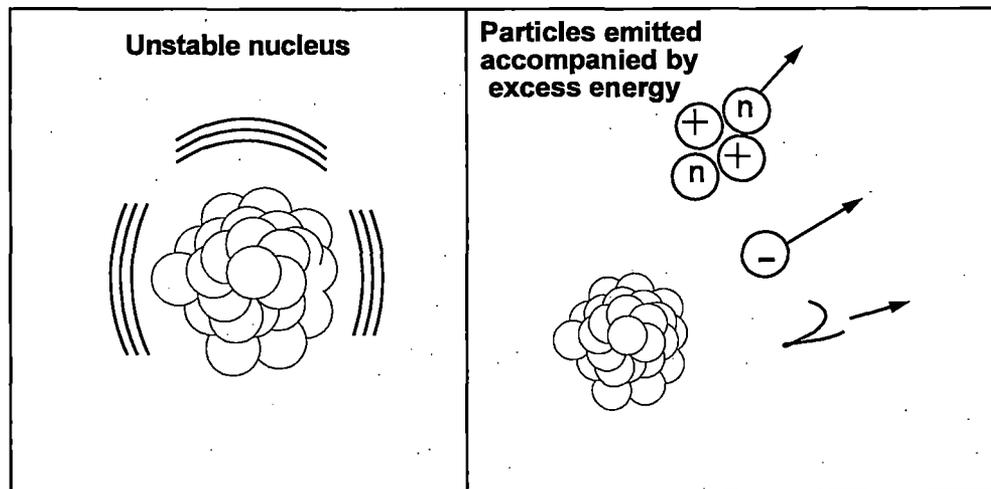


Figure 2 Radioactive decay

Ions and Ionization

Atoms can combine chemically to form molecules. Atoms and molecules are surrounded by orbiting electrons (negatively charged). If the number of electrons equals the total number of protons (positively charged) in the nucleus, the atom or molecule is neutral (uncharged).

Electrically charged atoms or molecules are called *ions*. Ions are either positively or negatively charged, depending on the number of orbiting electrons relative to the number of protons in the nucleus. As shown in Figure 3, ions with more electrons than protons are negatively charged, while ions with more protons than electrons are positively charged. The process of breaking a neutral atom or molecule into electrically charged parts is called *ionization*. This process requires energy. Ionization removes electrons from the atom, or molecule, leaving an ion with a positive charge. The negatively charged electron (which can attach itself to a neutral atom or molecule) and the positively charged ion, are called an ion pair. Radiation that causes ionization is called *ionizing radiation*.

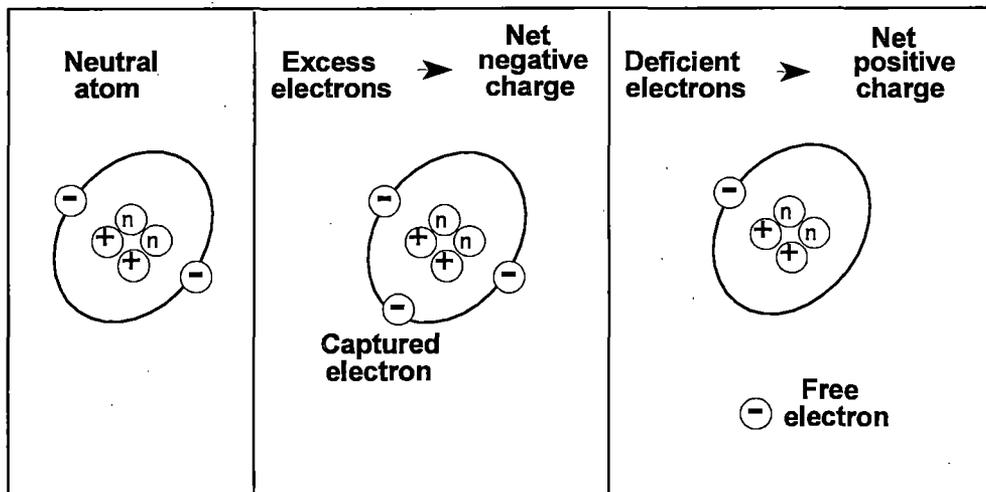


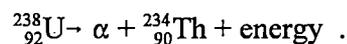
Figure 3 Neutral and ionized atoms

Types of Radiation

There are four basic types of ionizing radiation emitted from nuclei: alpha particles, beta particles, gamma rays, and neutrons.

- **alpha particle (α)**—consists of two protons and two neutrons and is the same as the nucleus of a helium atom (${}^4\text{He}$) (Figure 4). Generally, only the heavy nuclides can emit alpha particles.

A typical example of an α -emitting nuclide is uranium-238:



The mass of an alpha particle is about four times the mass of a single neutron or proton, and has a positive charge of +2 (it has no electrons). This positive charge causes the alpha particle

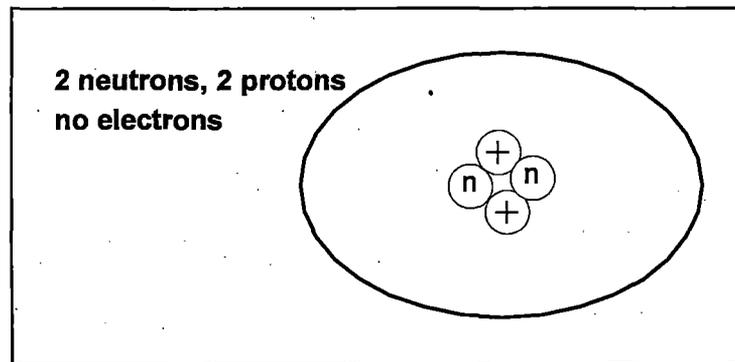


Figure 4 Alpha particle (or nucleus of a helium atom)

to ionize nearby atoms as it passes through body tissue. The strong positive charge and its relatively slow speed (resulting from its large mass) causes the alpha particle to interact strongly with orbiting electrons of atoms and molecules and to lose large amounts of energy in a short distance. This limits the penetrating ability of the alpha particle, making it easy to stop. A few centimeters of air, a sheet of paper, or the outer layer of skin stops alpha particles (Figure 5).

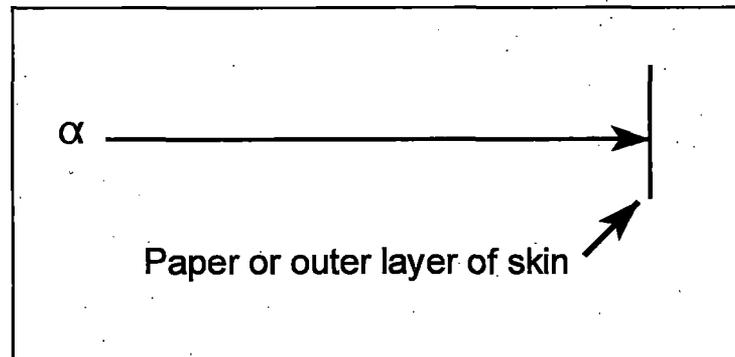


Figure 5 Alpha shielding

Alpha particles are not an external radiation hazard because they are easily stopped by protective clothing or the outer layer of skin. However, if an alpha emitter is inhaled or ingested, it becomes an internal radiation hazard. Because the source is in close contact with body tissue, the alpha particle will dissipate its energy in a short distance of the tissue.

- **beta particle (β)**—is equivalent to an electron except for its source. Beta-emitting nuclides have too many neutrons. A neutron emits a β particle, and the neutron is then converted to a proton. Tritium decay provides a good example of this process:

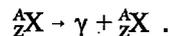


A β particle is identical to an electron, and its mass and charge are the same as those of an electron. As in the case of alpha particles, beta particles ionize atoms by removing electrons from their orbits. This reaction occurs from charged particle interactions or "collisions" with orbiting electrons.

Beta particles penetrate further than alpha particles of the same energy. A high-energy beta particle can penetrate a few centimeters of organic tissue. The higher the energy, the greater the penetrating ability. However, low-energy beta particles of tritium can be shielded by skin, paper, or only about 6 mm of air.

- **gamma ray (γ)**—is emitted when the nucleus of a nuclide releases stored energy without releasing a particle. Many gamma-emitters are found among the products of nuclear fission.

During pure γ emission the nucleus does not emit particles or change its nuclear structure or chemical characteristics. For instance,



Gamma radiation is in the form of electromagnetic waves (or photons). Gamma rays are similar to x-rays, but they differ in their origin and energy. Gamma rays originate within the nucleus, and x-rays originate outside the nucleus.

Gamma rays have a very high penetrating power because they have no charge or mass. Depending on their energy, a stream of gamma rays may penetrate with gradually diminishing intensity through several inches of concrete or similar material. They can be shielded effectively by very dense materials, such as lead and uranium. Gamma rays are a whole-body hazard. That is, because of their penetrating ability, the damage caused by gamma rays is not restricted to any particular body organ.

- **neutron (n)**—may be emitted spontaneously by heavy nuclei during fission or may be emitted during radioactive decay. They are uncharged particles that have mass and a high penetrating ability.

A neutron has about 2,000 times the mass of an electron, but only one-fourth the mass of an alpha particle. Neutrons are difficult to stop because they lack a charge. Neutrons mainly interact with matter by striking hydrogen nuclei or interacting with the nucleus of atoms. These collisions generally cause charged particles or other radiation to be emitted. These particles may then ionize other atoms. Collisions between neutrons and hydrogen nuclei (protons) are effective in stopping or slowing down high-energy neutrons. Neutrons are best shielded by materials with a high hydrogen content, such as water or plastic (see Figure 6).

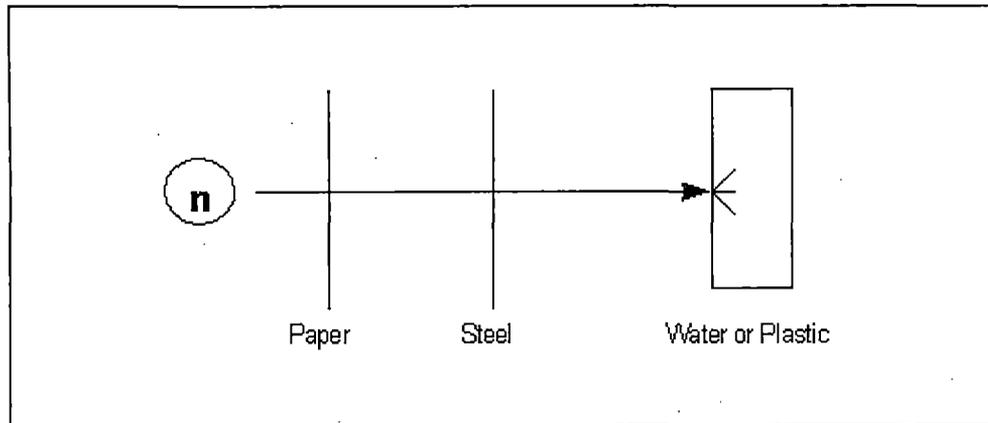


Figure 6 Neutron Shielding

Radioactivity

As radioactive isotopes decay, the number of radioactive nuclei decreases. The time required for half of the nuclei in a sample of a specific radioactive isotope to undergo decay is called its (physical) half-life (Figure 7). Each radioactive isotope has its own characteristic half-life. Radioactive isotopes decay to less than 1% of their original quantity after about seven half-lives.

Half-lives vary widely with different radionuclides, as shown by the following examples:

^{16}N	-	7.35 seconds
^3H	-	12.43 years
^{238}U	-	4.5×10^9 years (4.5 billion years)

The activity of a radioactive isotope sample is defined as the number of nuclei that decay per unit of time.

It has been shown that for a pure radioactive isotope the number of nuclei decaying per unit time (rate of decay) is proportional to the number of nuclei available to decay. If the substance is not being replenished, its activity will decrease accordingly. Therefore, in terms of half-life, the remaining activity after a period of time can be expressed as follows:

$$A_t = A_o \times (1/2)^n \quad \text{or} \quad A_o/A_t = 2^n$$

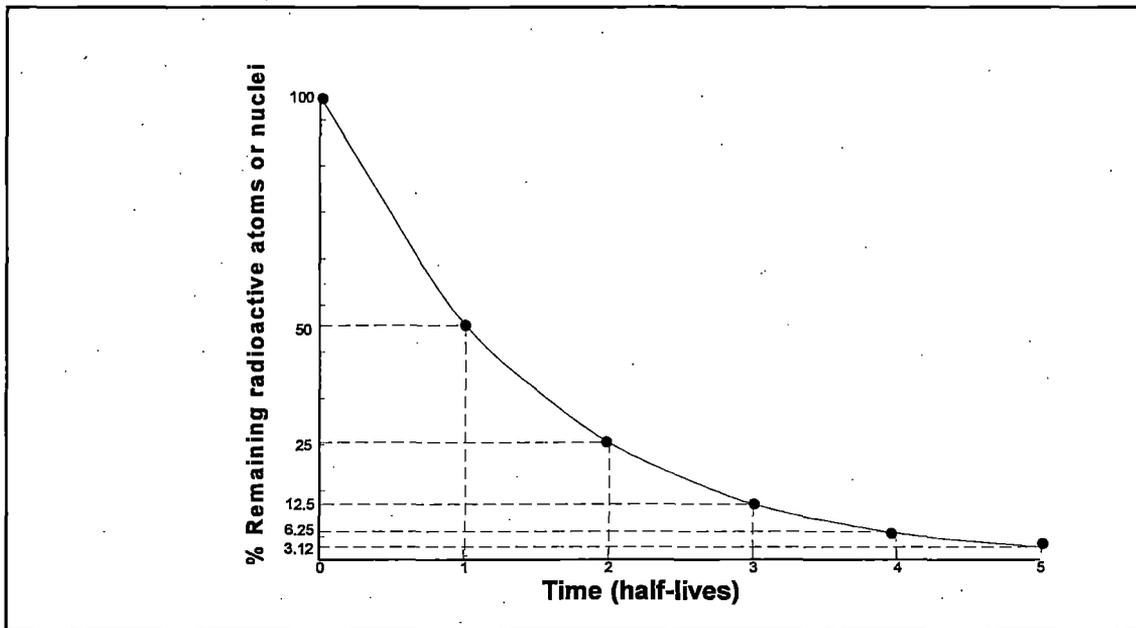


Figure 7 Half-life example

where:

- A_0 = the initial activity (Ci)
 A_t = activity after elapsed time, t (Ci)
 n = number of half-lives during elapsed time, equal to $t/T_{1/2}$, where
 t = elapsed time (time units)
 $T_{1/2}$ = half-life (time units).

For example, assume we have stored 10,000 Ci of tritium. How many curies will be left after 50 years of storage?

$$\begin{aligned}
 A_0 &= 10,000 \text{ Ci} \\
 n &= 50/12.43 = 4.02 \\
 A_t &= A_0 (1/2)^n \\
 A_t &= 10,000 (1/2)^{4.02} \\
 &= 10,000 \times 0.062 \\
 &= 620 \text{ Ci (approximately)}
 \end{aligned}$$

PHYSICAL AND CHEMICAL PROPERTIES OF TRITIUM

This section reviews the nuclear properties of tritium and discusses some of the physical and chemical properties that are important in understanding tritium handling, containment, and contamination control.

Nuclear and Radioactive Properties

Being an isotope of hydrogen, tritium has many of the properties of ordinary hydrogen (such as chemical reactions, permeability, and absorption). Differences may occur because the decaying tritium atoms can speed up (catalyze) reactions of undecayed tritium, or because atoms that have undergone decay have changed into helium atoms (^3He). Additionally, small differences in chemical reaction rates may result from the relative masses of the isotopes.

Some of the useful properties of tritium are listed in Table 2. Note that the properties listed are those of T_2 . The specific activity and power density of HT and DT are approximately one-half those for T_2 . The activity density of HT and DT is exactly one-half that of T_2 .

Table 2 Important nuclear properties of tritium

Half-Life	12.43 yrs
Specific Activity	9,545 Ci/g
Power Density	0.328 W/g
Activity Density	
(T_2 gas, 1 atm, 0°C)	2.589 Ci/cm ³
(T_2 gas, 1 atm, 25°C)	2.372 Ci/cm ³

Penetration Depths of Beta Particles

The penetration and absorption of beta particles in a material are important factors for detecting tritium and understanding the mechanisms by which tritium can degrade materials. A beta particle interacts with matter by colliding with electrons in the surrounding material. In each collision, the

beta particle may lose several electron volts (keV)^b of energy, and the electron is stripped from its atom (ionization) or promoted to an excited state. The beta particle has a finite penetration depth that depends on its energy.

Recall that tritium undergoes beta decay according to the following equation:



The helium daughter (${}^3_2\text{He}$) is stable, but lighter than common helium (${}^4_2\text{He}$). The decay energy is constant (18.6 keV), but is shared between the beta particle and an antineutrino (a tiny particle). The result is that not all beta particles have the same energy. The average energy is 5.7 keV. Consequently, not all tritium betas have the same penetration depth in a given material. Where beta ranges are given, it is customary to list both the highest energy and the average, most representative energy, as listed in Table 3.

Table 3 Penetration depths of tritium betas

Material	E(β) (keV)	Penetration Depth
T ₂ gas, STP ^a	5.7	0.26 cm
T ₂ gas, STP	18.6	3.2 cm
Air, STP	5.7	0.036 cm
Air, STP	18.6	0.45 cm
Water, soft tissue (and oils/polymers of density \approx 1)	5.7	0.42 μm
Water, soft tissue (and oils/polymers of density \approx 1)	18.6	5.2 μm
Stainless steel	5.7	0.06 μm

a. STP = Standard temperature (0°C) and pressure (760 Torr).

b. An electron volt is a small unit of energy used in descriptions of nuclear and chemical reactions. It equals the energy gained by an electron when it moves across a potential of 1 volt.

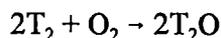
With one unimportant exception, tritium is the weakest beta emitter known. The range of the most energetic tritium beta particles is only about 5 mm in air or 0.005 mm in water or soft tissue. This range makes it a nonhazardous outside the body, but presents a detection problem. Where other radioisotopes can be detected by virtue of their penetrating radiation, tritium has to be introduced directly inside the detector or counter to be measured.

Chemical Properties

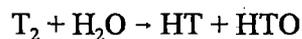
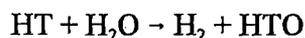
Laboratories that have large quantities of tritium usually handle it in the form of HT. However, at any time the tritium may be stored on metal getter beds (such as titanium, zirconium, or uranium). These beds form weak chemical compounds with hydrogen. Some of the beds are stable in air; and others are not and can only be used in certain atmospheres. The tritium is released (or delivered) by heating the beds to the required temperature.

Laboratories may also handle tritiated gases (such as ammonia and methane) and other compounds. By far the most common of these is HTO, which is formed from HT whenever it is exposed to oxygen or water vapor. The conversion reactions are oxidation and exchange:

oxidation



exchange



These reaction rates are increased by radiation (from nearby tritium at high concentrations), heat, or the presence of metal catalysts (especially palladium or platinum). All chemical reactions involving hydrogen can also be performed with tritium, sometimes at a higher rate if the tritium concentration is high enough to catalyze the reaction. One of the most important reactions occurs when a tritium atom exchanges with a loosely bonded hydrogen atom of an organic molecule. However, where HT is dissolved in water (H_2O), the exchange process is fairly slow because the hydrogen in is tightly bonded and the reaction is not catalyzed.

Contamination

Tritium as HT or HTO will readily adsorb onto the surface of most metals (such as stainless steel, copper, or aluminum), plastics, and rubbers. The tritium will remain fairly close to the surface unless the metal is heated to a high temperature. At room temperature, permeation into these metals is usually extremely slow.

In the case of metal contamination, the tritium remains on or very close to the surface. The contamination can be removed with water or water vapor if the surface is contaminated with HTO or with hydrogen (H_2 or D_2) if the contamination is HT. Heating also speeds up the decontamination process. The initial application of heat to surfaces can also be used to prevent or lessen the contamination by HT or HTO. Metal surfaces exposed to high pressures of HT or HTO for extended periods, especially at high temperatures, may allow enough penetration to cause structural damage to the metal. This is especially true if the decaying tritium causes a buildup of helium within the structure of the metal.

If adsorbed onto hydrogenous material, the tritium will easily permeate into the material. The HTO will move much more rapidly into the bulk material than will HT. The permeation rate varies with the type of material and is accelerated by increasing the temperature. As a result of this movement, plastics and rubbers exposed to tritium (especially as HTO) are readily contaminated deep into the bulk material and are impossible to decontaminate completely. After a period of time, the tritium exchanges with bulk hydrogen and presents little biological risk.

Highly contaminated metal or plastic surfaces may release some of the loosely-bound tritium immediately after exposure to the contaminating tritiated atmosphere or liquid. This is referred to as outgassing. The personnel risk from outgassing tritium is generally much less than that from making unprotected skin contact with the outgassing surface.

BIOLOGICAL PROPERTIES OF TRITIUM

At most tritium facilities, the most commonly encountered forms of tritium are tritium gas (HT) and tritium oxide (HTO). Other forms of tritium may be present, such as metal tritides, tritiated pump oil, and tritiated gases such as methane and ammonia. As noted earlier, deuterated and tritiated compounds generally have the same chemical properties as their protium counterparts, although some minor isotopic differences in reaction rates exist. These various tritiated compounds have a wide range of metabolic properties in humans under similar exposure conditions. For example, inhaled tritium gas is only slightly incorporated into the body during exposure, and the remainder is rapidly removed (by exhalation) following the exposure. On the other hand, tritiated water vapor is readily taken up and retained in the body water. In this Primer, we will address only those compounds likely to be found at DOE laboratories: gaseous tritium, tritiated water, other tritiated species, metallic getters, and other tritiated liquids and gases.

Metabolism of Gaseous Tritium

During a brief exposure to tritium gas, the gas is inhaled and a small amount is dissolved in the bloodstream. The dissolved gas circulates in the bloodstream before being exhaled along with the gaseous waste products (carbon dioxide) and normal water vapor. If the exposure persists, the gas will reach other body fluids. A small percentage of the gaseous tritium is converted to the oxide (HTO), most likely by oxidation in the gastrointestinal tract. Early experiments involving human exposure to a concentration of 9 $\mu\text{Ci/mL}$ resulted in an increase in the HTO concentration in urine of $7.7 \times 10^{-3} \mu\text{Ci/mL}$ per hour of exposure. Although independent of the breathing rate, this conversion can be expressed as the ratio of the HTO buildup to the tritium inhaled as HT at a nominal breathing rate (20 L/min). In this context, the conversion is 0.003% of the total gaseous tritium inhaled. More recent experiments with six volunteers resulted in a conversion of 0.005%. For gaseous tritium exposures, there are two doses: (a) a lung dose from the tritium in the air inside the lung and (b) a whole body dose from the tritium gas that has been converted to HTO. The tritiated water converted from the gas in the body behaves as an exposure to tritiated water.

Intake of gaseous tritium through the skin has been found to be negligible compared with that from inhalation. Small amounts of tritium can enter the skin through unprotected contact with contaminated metal surfaces, which results in organically bound tritium in skin and in urine. Ordinarily this is not a serious problem because surfaces highly contaminated with tritium gas are inaccessible to skin contact. Also, most tritium exposed to air will be converted to the oxide form (water vapor) before the internal surfaces of equipment are handled during maintenance or repair operations.

Metabolism of Tritiated Water

The biological incorporation (uptake) of airborne HTO can be extremely efficient: up to 99% of inhaled HTO is taken into the body by the circulating blood. Ingested liquid HTO is also

almost completely absorbed by the gastrointestinal tract and quickly appears in the blood stream. Within minutes, it can be found in varying concentrations in the organs, fluids, and tissues of the body. Skin absorption of airborne HTO is also important, especially during hot weather, because of the normal movement of water through the skin. For skin temperatures between 30 and 40°C, the absorption of HTO is about 50% of that for HTO by inhalation (assuming an average breathing rate associated with light work, 20 L/min). No matter how it is absorbed, the HTO will be uniformly distributed in all biological fluids within one to two hours. This tritium has a retention that is characteristic of water. In addition, a small fraction of the tritium is incorporated into easily exchanged hydrogen sites in organic molecules. Hence, retention of tritiated water can be described as the sum of several terms: one characteristic of body water, and one or more longer-term components that represent tritium incorporated into organic hydrogen sites.

Metabolism of Other Tritiated Species

Most tritium handled in laboratories is in the form of tritiated gas or tritium oxide. However, tritium handling operations may form other compounds, such as tritiated hydrocarbons and metal tritides. Tritium may also contaminate surfaces and liquids such as pump oil. These materials may present special safe handling problems.

Metallic Getters

Although many metals are commonly used for gettering (chemically combining with) tritium, little information on their metabolic properties is available. Some of these compounds (such as uranium tritide and lithium tritide) are unstable in air. For these, exposure to air produces different results. Uranium tritide, being pyrophoric, releases large quantities of tritiated water; lithium tritide, a hydroxyl scavenger, releases mostly tritium gas.

Tritides of metals (such as titanium, niobium, and zirconium) are stable in air. For particles of these tritides, the primary organ of concern is the lungs. Some of the tritium may leach out in the lung fluids and then be incorporated into the body water. These particles may also produce organically bound tritium from contact with lung tissue, which would further complicate the metabolic process. However, in laboratories where such tritiated metals are handled, the possibility for exposure to airborne particulates of these metals is extremely remote except in accident situations.

Tritiated Liquids

Next to HTO, the most common tritiated liquid is tritiated vacuum pump oil. Experience at DOE facilities has shown that the specific activities of pump oils can easily range from a few mCi/L to a few tens of Ci/L. The wide range in specific activities may result from variations in the tritium concentration and total throughput of tritium. Depending on the history of these pumps, the tritium may be found as HT, HTO, or tritiated hydrocarbons.

Next to pump oils, the next most common group is tritiated solvents. All solvents can be absorbed through the skin and are relatively volatile and toxic. The overall toxicity of tritiated solvents is usually dominated by the chemical nature of the solvent.

Other Tritiated Gases

If tritium is released in a nitrogen- or air-filled glovebox, other tritiated gases may be formed, such as ammonia and methane. The conversion of tritium to tritiated ammonia is small unless the tritium concentration is very high. The toxicity of these gases is not believed to be greater than that of tritium oxide.

Biological Half-Life of HTO

Studies of biological elimination rates of body water in humans date back to 1934 when the body water turnover rate was measured using HDO. Since that time, several additional studies have been conducted with HDO and HTO. A simple average of the data suggests a value of 9.5 days for the measured biological half-life of water in the body with a deviation of $\pm 50\%$. Calculations based on total fluid intake indicate a similar value. This is reasonable because the turnover rate of HTO should be identical to that of body water. In other words, the biological half-life of tritium is a function of the average daily throughput of water.

The biological half-life of HTO has been studied when outdoor temperatures varied at the time of tritium uptake. The data suggest that biological half-lives are shorter in warmer months. For example, the 7.5-day half-life measured in southern Nigeria is not surprising because the mean outdoor temperature there averages 27°C . In contrast, an average 9.5-day half-life was measured in North America, where the mean outdoor temperature averages 17°C . Such findings are consistent with metabolic pathways involving sensible and insensible perspiration. As such, the skin absorption and perspiration pathways can become an important part of body water exchange routes. It is important to note that personnel who are perspiring will have a greater absorption of tritium from contact with tritiated surfaces. For planning purposes, it is customary to use an average half-life of 10 days. However, it is not used to calculate doses from actual exposures.

Prolonged exposures can be expected to affect the biological half-life. Tritium's interaction with organic hydrogen can result in additional half-life components ranging from 21 to 30 days and 250 to 550 days. The shorter duration indicates that organic molecules in the body retain tritium relatively briefly. The longer duration indicates long-term retention by other compounds in the body that do not readily exchange hydrogen or that metabolize more slowly. However, the overall contribution from organically bound tritium is relatively small, that is, less than about 5% for acute exposures and about 10% for chronic exposures. Methods used to compute the annual limits on intake of air and water specify only the body water component and include the assumption of a 10-day biological half-life, as mentioned above.

Bioassay and Internal Dosimetry

Exposure to tritium oxide (HTO) is by far the most important type of tritium exposure. The HTO enters the body by inhalation or skin absorption. When immersed in tritiated water vapor, the body takes in approximately twice as much tritium through the lungs as through the skin. Once in the body, it is circulated by the blood stream and finds its way into fluids both inside and outside the cells.

According to International Commission on Radiological Protection (ICRP) Publication 30, the derived air concentration (DAC)⁸ for tritium gas (HT) and HTO are 540,000 $\mu\text{Ci}/\text{m}^3$ and 21.6 $\mu\text{Ci}/\text{m}^3$, respectively. The ratio of these DACs (25,000) is based on a lung exposure from the gas and a whole body exposure from the oxide. However, as was noted earlier, when a person is exposed to HT in the air, an additional dose actually results: one to the whole body. During exposure to HT, a small fraction of the tritium exchanges in the lung and is transferred by the blood to the gastrointestinal tract where it is oxidized by enzymes. This process results in a buildup of HTO until the HT is removed by exhalation at the end of the exposure. The resultant dose from exposure to this HTO is roughly comparable to the lung dose from exposure to HT. Thus, the total effective dose from an HT exposure is about 10,000 times less than the total effective dose from an equal exposure to airborne HTO. For both HTO and HT exposures, a bioassay program that samples body water for HTO is essential for personnel monitoring at tritium facilities.

Sampling Schedule and Technique

After HTO enters the body, it is quickly distributed throughout the blood system and, within 1 to 2 hours, throughout all water in the body. Once equilibrium is established, the tritium concentration is found to be the same in samples of blood, sputum, and urine. For bioassay purposes, urine is normally used for determining tritium concentrations in body water.

Workers who may be or who have been exposed to tritium are normally required to submit urine samples for bioassay periodically. The sampling period may be daily, biweekly, or longer, depending on the potential for significant exposure.

Special urine samples are normally required after an incident or a work assignment with a high potential for exposure. After a possible exposure, the worker should empty the bladder 1 to 2 hours later. A sample taken after the bladder is emptied should be reasonably representative of the body water concentration. A sample collected before equilibrium is established will not be representative because of dilution in the bladder, or because of initial high concentration in the blood. However, any early sample may still be useful as a sign of the potential seriousness of the exposure.

c. The DAC is defined as that concentration of a gas, which, if a worker were exposed to it for one working year (2,000 hours), would result in an annual dose of 5 rem.

A pure HT exposure is considered as a combination of a lung exposure from the HT and a whole body exposure from HTO. The HTO comes from the conversion of HT dissolved in the blood. The whole body dose can be determined as outlined above by analysis for HTO in the urine. Because the effective dose equivalents from the lung and whole body exposures are about equal, the total effective dose can be obtained conservatively by multiplying the HTO whole body dose by 2. However, in general, this is too conservative because a release of pure tritium gas with less than 0.01% HTO is highly unlikely. With only a slight fraction (~0.1%) of HTO in the air, the total effective dose is essentially the HTO whole body dose determined by bioassay.

As noted above, tritium-labeled molecules in the skin result from contact with metal surfaces contaminated with HT. This form is associated with a longer half-life. Lung exposure to airborne metal tritides may also cause unusual patterns of tritium concentrations in body water because of the slow release of tritium to the blood stream. If such exposures are possible at the facility, it is good practice to follow the elimination data carefully and to look for organically bound tritium in the urine.

The results of the bioassay measurements and their contribution to the worker's dose and general health must be shared with the worker in a timely fashion.

Dose Reduction

The committed dose following an HTO exposure is directly proportional to the biological half-life, which in turn is inversely proportional to the turnover rate of body water. This rate varies from individual to individual. Such things as temperature, humidity, work, and drinking habits may cause rate variations. Although the average biological half-life is 10 days, it can be decreased by simply increasing fluid intake, especially diuretic liquids such as coffee, tea, beer, and wine. Even though the half-life may be easily reduced to 4 to 5 days in this way, a physician must be consulted before persons are placed on a regimen that might affect their health. Chemical diuretics require medical supervision because the resultant loss of potassium and other electrolytes can be very serious if they are not replaced. Such drastic measures can result in a decrease in half-life to 1 to 2 days. Even more drastic is the use of peritoneal dialysis or a kidney dialysis machine, which may reduce the half-life to 13 and 4 hours, respectively. Such extreme techniques should be used only in life-threatening situations involving potential committed dose equivalents that would exceed about 100 rem without any treatment. Based on a 10-day half-life, the committed dose for an intake of 1 mCi of HTO is approximately 63 mrem.

Individuals whose urine concentrations exceed established limits should stop work that involves possible exposure to radiation, whether from tritium or other sources. Work restrictions are suggested or imposed to make certain that the annual dose limits for workers are not exceeded. The operating group may impose stricter limits on their staff than those imposed by the health physics group. Depending on the number of workers available and the importance of the work to be done, doses can be managed to safe levels (from 5 to 100 $\mu\text{Ci/L}$ in urine).

Results of bioassay sampling should be given to workers who have submitted samples as soon as they are available. The results may be posted, or the workers may be notified personally. Moreover, the results must be kept in the workers' radiation exposure records or medical files. Like any other radiation exposure, any dose in excess of the limits specified by applicable regulations must be reported to DOE.

TRITIUM MONITORING

The tritium monitoring system at a tritium handling facility is critically important to its safe operation. Operators and others at the facility need to be informed of the status of the processes, the development of any leaks in the primary or secondary containments, or of any releases to the room or environment so that protective measures and corrective action may be taken quickly. The location and degree of surface contamination are equally important to prevent accidental uptakes of tritium by personnel.

In this section, the various techniques used to monitor for tritium in gases (including air), in liquids, and on surfaces will be discussed.

Air Monitoring

Fixed ionization chamber instruments are the most widely used instruments for measuring gaseous forms of tritium in laboratory and process monitoring applications. Portable ionization chamber instruments are also used to control contamination and to supplement fixed instrument measurements. Such simple devices require only an electrically polarized ionization chamber, suitable electronics, and a method for moving the gas sample through the chamber—usually a pump. Chamber volumes typically range from a tenth to a few tens of liters, depending on the required sensitivity. The output is usually given in units of concentration (typically $\mu\text{Ci}/\text{m}^3$) or, if a commercial electrometer or picoammeter is used, in current units that must be converted to those of tritium concentration. The following rule-of-thumb can be used to convert current to concentration: $10^{15} \times \text{current (amps)}/\text{chamber volume (liters)} = \text{concentration } (\mu\text{Ci}/\text{m}^3)$. For real-time tritium monitoring, the practical lower limits of sensitivity range from 0.1 to 10 $\mu\text{Ci}/\text{m}^3$. External background radiation or the presence of radon can lower the sensitivity of the instrument.

For measurements of low concentrations, sensitive electrometers are needed. For higher concentrations ($>1 \text{ mCi}/\text{m}^3$ for example), the requirements on the electronics can be relaxed, and smaller ion chambers may be used. Smaller chambers also need less applied voltage. Because of a greater ratio of surface area to volume, residual contamination in the chamber is more likely and is called "memory." This residual contamination elevates the background chamber current. Response times for higher level measurements can be made correspondingly shorter. However, because small chambers and chambers operated at low pressures may have significant wall effects, the above rule-of-thumb may not apply. Such instruments would have to be calibrated to determine their response.

Although most ionization chambers are the flow-through type that require a pump to provide the flow, a number of facilities use "open window" or "perforated wall" chambers. These chambers, which employ a dust cover to protect the chamber from particulates, allow the air or gas to penetrate through the wall to the inside chamber without the need for a pump. These instruments are used as single point monitors to monitor rooms, hoods, gloveboxes, and ducts.

Differential Air Monitoring

Because HTO is more toxic than HT (10,000 to 25,000 times greater), it may be desirable to know the relative amounts of each species following a significant release into a room or to the environment. In the case of stack monitoring, discrete samples of the stack effluent should be taken using bubblers or desiccants with a catalyst for oxidizing the HT. Another technique for differential monitoring uses a desiccant cartridge in the sampling line of an ionization chamber monitor. The result is a measurement of the HT concentration. Without the cartridge, the total tritium concentration is measured. Subtraction of HT from the total produces the HTO concentration. The technique may be used with two instruments or one instrument in which the desiccant cartridge is automatically switched in and out of the sampling line.

Another technique uses a semipermeable membrane tube bundle in the sampling line to remove the HTO (preferentially over the HT), which is directed to an HTO monitor. After removing the remaining HTO with another membrane dryer, the sampled air is directed to the HT monitor. Although this technique is slower than the one requiring a desiccant cartridge, it does not require a periodic cartridge replacement. Furthermore, it can be adapted to measure tritium in both species in the presence of noble gases or other radioactive gases by adding a catalyst after the HTO dryers, followed by additional membrane dryers for the HTO. However, because of its slow response, it is more suitable for effluent or stack monitoring than for room monitoring. Because significant releases into a room are quite rare, it is easier to treat any such release as one of HTO than use complicated techniques for continuous differential monitoring.

Discrete Air Sampling

Discrete sampling differs from real-time monitoring in that the sampled gas (usually air) must be analyzed for tritium content (usually by liquid scintillation counting). The usual technique is to flow the sampled air through either a solid desiccant (molecular sieve, silica gel, or Drierite) or water or glycol bubblers. For low-flow rates (about 0.1 to 1 L/min), bubblers may be used. Bubblers are more convenient for sampling, but are less sensitive than the solid desiccant cartridges if the water in the desiccant is recovered by heating. Glycol or water may be used, but glycol is preferred for long-term sampling. In any case, the collected water is then analyzed for HTO. For differential monitoring of HTO and HT, a heated catalyst (usually a palladium sponge) is used between the HTO desiccant cartridge or bubblers and the HT cartridge or bubblers. This is currently the preferred method for monitoring stacks for reporting purposes. In a different arrangement, palladium is coated on the molecular sieve in the HT cartridge to oxidize the HT into HTO, which is then absorbed by the molecular sieve. However, this technique is used primarily for environmental monitoring.

Another technique for sampling HTO in room air is to use a "cold finger" to freeze HTO out of the air. An alcohol and dry ice mixture in a stainless steel beaker works well. To

determine the concentration, the relative humidity must be known. Another sampling technique is to squeeze a soft plastic bottle several times to introduce the air (containing the HTO) into the bottle. A measured quantity of water is then introduced, and the bottle is capped and shaken. In a minute or less, essentially all the HTO is taken up by the water, which is then analyzed.

Other techniques involve placing a number of vials or other small specially designed containers of water, cocktail, or other liquid in selected locations in the area being monitored. After a period of time (usually a number of days), the liquid in the containers is analyzed. The result is qualitative (for open containers) to semiquantitative (for specially designed containers).

Process Monitoring

Ionization chambers are typically used for monitoring stacks, rooms, hoods, glove boxes, and processes. The outputs can be used to sound alarms, activate ventilation valves, activate detritiation systems, and perform other functions. In general, it can be expected that stack, room, and hood monitors will require little nonelectronic maintenance (i.e., chamber replacement because of contamination). Under normal circumstances, the chambers are constantly flushed with clean air and are not exposed to high tritium concentrations. However, glove box monitors can be expected to eventually become contaminated, especially if exposed to high concentrations of HTO. Process HT monitor backgrounds can also be expected to present problems if a wide range of concentrations (4 to 5 orders of magnitude) are to be measured.

Mass spectrometers, gas chromatographs, and calorimeters are the main instruments used for process monitoring. Because of their relative insensitivities, these instruments cannot detect tritium much below a few parts per million (Ci/m³). For this reason, the analytical results and the related health physics concerns must be interpreted carefully. It is not uncommon to find that samples showing no trace of tritium when analyzed on a mass spectrometer may actually have a concentration of several curies of tritium per cubic meter. In spite of their contamination problems, ionization chamber instruments are useful for measuring these lower concentrations and for providing instant indications of changing concentrations that are not possible with the more sophisticated instruments.

Surface Monitoring

Any material exposed to tritium or a tritiated compound has the potential of being contaminated. Although it is difficult to quantify tritium contamination levels, several methods are available to evaluate the extent of contamination, including smear surveys and off-gassing measurements. Good housekeeping and work practices are essential in maintaining contamination at acceptable levels.

For health or safety implications, an indication of loose, removable tritium contamination is more valuable than a measurement of the total surface contamination. Loose tritium can be

transferred to the body by skin contact or inhalation if it becomes airborne. As a result, loose contamination is routinely monitored by smears, which are wiped over a surface and then analyzed by liquid scintillation or proportional counting.

The smears are typically small round filter papers used dry or wet (with water, glycol, or glycerol). Wet smears are more efficient in removing tritium, and the results are more reproducible, although the papers are usually more fragile when wet. However, results are only semiquantitative, and reproducibility within a factor of 2 agreement (for wet or dry smears) is considered satisfactory. Ordinarily, an area of 100 cm² of the surface is wiped with the smear paper and quickly placed in a vial with about 10 mL of liquid scintillation cocktail, or 1 or 2 mL of water with the cocktail added later. The paper must be placed in liquid immediately after wiping because losses from evaporation can be considerable, especially if the paper is dry. The efficiency of the liquid scintillation cocktail is only slightly affected by the size of the swipe. Foam smears are also available commercially. These smears dissolve in most cocktails and do not interfere significantly with the normal counting efficiency.

Smears may be counted by gas-flow proportional counting. However, because of the inherent counting delays, tritium losses before counting can be significant. Moreover, counting efficiencies may be difficult to determine and may vary greatly from one sample to the next. Another drawback is potential contamination of the counting chamber when counting very "hot" smears. For all of these reasons, a liquid scintillation spectrometer is the preferred system.

An effective tritium health physics program must specify the frequency of routine smear surveys. Each facility should develop a routine surveillance program that may include daily smear surveys in laboratories, process areas, step-off pads, change rooms, and lunchrooms. In many locations within a facility, weekly or monthly routine smear surveys may be sufficient. The frequency should be dictated by operational experience and the potential for contamination. In addition to the routine survey program, special surveys should be made following spills or on potentially contaminated material being transferred to a less controlled area to prevent the spread of contamination from controlled areas.

The surface contamination levels acceptable for the release of materials from radiological areas may be found in the DOE *Radiological Control Manual* and DOE Order 5400.5.

Tritium Probes

In general, the total tritium contamination on a surface can be measured only by destructive techniques. When tritium penetrates a surface even slightly, it becomes undetectable because of the weak energy of its beta particles. With open-window probes operated in the Geiger Mueller (GM) or proportional regions, it is possible to measure many of the betas emitted from the surface. Quantifying that measurement in terms of the total tritium present is difficult because the history of every exposure is different. Consequently, the relative amounts of measurable and unmeasurable tritium are different.

Such monitoring probes are used to survey areas quickly before more careful monitoring by smears, or to monitor the smears themselves while in the field.

The probe must be protected carefully from contamination. When monitoring a slightly contaminated surface after monitoring a highly contaminated one, contamination of the probe can be an immediate problem. Placing a disposable mask over the front face of the probe can reduce, but never eliminate this contamination, particularly if the tritium is rapidly outgassing from the surface. Sensitivity of the instrument depends on many factors, but should be about 10^3 to 10^4 dpm/cm².

For highly contaminated surfaces (>1 mCi/100 cm²), a thin sodium iodide crystal or a thin-window GM tube can be used to measure the characteristic and continuous x-rays (Bremsstrahlung) emitted from the surface as a result of the interaction of the beta particle with the surface material.

Off-Gassing Measurements

Off-gassing can be measured using one of two methods. The simplest method is to "sniff" the surface for airborne tritium using a portable or fixed tritium monitor. The most reliable method, however, uses a closed-loop system of known volume and a flow-through ionization chamber monitor. By placing the sample inside the volume and measuring the change in concentration over time, tritium off-gassing rates can be determined accurately on virtually any material. The initial off-gassing rate is the required value because the equilibrium concentration may be reached quickly in a closed volume, especially if the volume is small because of recontamination by the airborne tritium.

The uptake of tritium from off-gassing materials is difficult to predict. Off-gassing tritium that is readily measured indicates contaminated equipment that should not be released for uncontrolled use.

Liquid Monitoring

Liquid is almost universally monitored by liquid scintillation counting. The liquid must be compatible with the cocktail. Certain chemicals can degrade the cocktail. Others may retain much of the tritium; still others result in a high degree of quenching. In addition, samples that contain peroxide or that are alkaline may result in chemiluminescence that can interfere with measuring. Such samples should first be neutralized before counting. Chemiluminescence and phosphorescence both decay with time. Phosphorescence, activated by sunlight or fluorescent lighting, decays in the dark in a few minutes (fast component) to several days (slow component). Chemiluminescence, the result of chemical interaction of sample components, may take days to decay at room temperatures, but takes only hours to decay at the cold temperatures of a refrigerated liquid scintillation spectrometer. Distillations may be necessary for some samples.

For rather "hot" samples, as may be the case for vacuum pump oils, Bremsstrahlung counting may be useful. This technique may also be useful for active monitoring of "hot" liquids. Liquids may be monitored actively with scintillation flow cells, which are often made of plastic scintillator material or of glass tubing filled with anthracene crystals. However, both types are prone to memory effects that result from tritium contamination. In addition, flow cells are also prone to contamination by algae or other foreign material that can quickly degrade their counting efficiency.

RADIOLOGICAL CONTROL AND PROTECTION PRACTICES

Airborne Tritium

Tritium released to room air moves readily with normal air current. The room or building ventilation system should be designed to prevent the air from being carried to uncontaminated areas, such as offices or other laboratories where tritium is not allowed. For that reason, differential pressure zoning is commonly used, and released tritium is directed outside through the building stack. In some newer facilities where the large quantities of tritium are being handled, room air cleanup systems are available for emergency use. Following a significant release, the room ventilation system is effectively shut down, the room is isolated, and cleanup of room air is begun.

Secondary Containment

The most important control for preventing a release of tritium to the room atmosphere is the use of containment around the source of tritium. This containment usually takes the form of a glove box, which is then a secondary containment if the tritium is already contained within the process plumbing, which is the primary containment. Even if the tritium is on the outside surface of a piece of equipment and located inside the glove box, through popular usage, the box is still referred to as the secondary containment.

Glove boxes used for tritium work typically are made of stainless steel or aluminum and use gloves made of butyl, neoprene, or Hypalon. Windows are made of glass or Lexan. In order to reduce the amount of tritium released to the atmosphere, glove boxes where significant quantities of tritium are handled incorporate detritiation systems that process the glove box atmosphere and remove the tritium. These detritiation systems, including the room cleanup systems mentioned above, convert released HT to HTO and collect the HTO on a molecular sieve for later recovery or burial. Newer systems use metal getters that recover HT without resorting to oxidation. These getters, which can only be used in certain glove box atmospheres, can be heated to release and recover the HT easily.

The atmosphere in the glove box may be air, nitrogen, argon, or helium, depending on the type of activity in the box. Even in boxes with inert gas atmospheres, small amounts of moisture and oxygen exist. Any release of tritium gas in the box will eventually be converted to the oxide. As a result, the oxide will slowly diffuse through the gloves and contaminate their outside surfaces. For that reason, personnel using glove boxes that have had tritium releases are required to wear one or more additional pairs of disposable gloves when working in the glove box.

Glove box monitors are used to alert personnel of a release in the box and may be used to activate a cleanup system or to increase the rate of the cleanup process. With releases of tritium in the box, the monitor chamber will eventually develop a memory from

contamination, mainly by HTO. Heated monitor chambers are useful in minimizing contamination by HTO.

The relative pressure of the glove box atmosphere is normally kept negative in order to prevent the gloves from hanging outside the box where passersby may brush against them and to prevent tritium from escaping into the room should a leak develop in the glove box. However, outward permeation of HTO through the gloves and inward permeation of room moisture are not affected by the pressure inside the glove box.

Temporary Enclosures

At times, maintenance or repair work is done on equipment that cannot be moved into a glove box or fume hood and that has a high potential to release tritium. For these activities a temporary box ("tent"), may be constructed over the equipment, and an existing cleanup system installed to process the air. Alternatively, if the tritium at risk is not significant, the enclosing atmosphere may be purged to the stack. If the enclosure is small, gloves and glove ports may be fitted to the side of the enclosure. For larger enclosures entry may be required. In such cases, personnel must work in air-supplied suits inside the enclosure.

Protection by Local Ventilation

In spite of the greater protection afforded by glove boxes, fume hoods are commonly used at tritium facilities for handling or storing material with low quantities of tritium or with low-level contamination. Limits are generally imposed on the quantities used or stored in these hoods.

Fume hoods are also used to protect personnel at the outside door of glove-box pass boxes where materials are passed into and out of the boxes. Ideally, any tritium released in a hood from outgassing or a leaky container, for instance, is routed to the hood's exhaust duct. However, turbulence may occur at the hood entrance, resulting in backwash and possible contamination of personnel if the face velocity is not adequate for the design of the hood, the activities in the hood, or the local conditions (such as traffic in front of the hood). No hood should be used that has not been thoroughly surveyed and judged acceptable for tritium use.

For small operations local ventilation is commonly provided at the work site through a flexible ventilation duct ("elephant trunk") directed to the room exhaust system. The exhaust of these ducts is generally directed to the building ventilation exhaust system, which of itself may be adequate to supply the needed air flow for the duct without help from an additional in-line blower.

Flexible ducts can provide adequate ventilation during maintenance in a glovebox with a panel removed. In this application, a flexible duct can be connected to a gloveport before the panel is removed, and then the work can proceed safely.

Supplied-Air Respirators

In general, only supplied-air respirators are effective in preventing inhalation of airborne tritium. Two types of air-supplied respirators are available: self-contained breathing apparatus (SCBA) and full-face supplied air masks.

An SCBA, consisting of a full-face mask fed by a bottle of compressed air carried on the worker's back, provides excellent protection against HTO inhalation. Because the mask provides no protection against absorption by most of the skin, the SCBA is normally reserved for emergency use only. The protection factor of 3 or more afforded by the SCBA may be adequate for some applications. An SCBA can be used as an added precaution during certain maintenance or operations that experience has shown should not result in the release of significant amounts of HTO. Nevertheless, the potential for exposure is real, and the SCBA gives the worker time to leave the area if necessary before a skin exposure occurs.

Full-face supplied-air masks are also available. Because the air is normally supplied by a fixed-breathing-air system, they are not practical for many emergency situations and, consequently, are not as popular as SCBAs.

Supplied-Air Suits

Because of the inherent disadvantages associated with respirators and other breathing apparatus, supplied-air plastic suits that completely enclose the body are often used by facilities that handle large quantities of tritium. Although they afford reasonably complete body protection, they are slow to don and cumbersome to wear. For these reasons, they are not favored for rescue work where time and mobility are important considerations. For certain maintenance operations outside of glove boxes with a high degree of risk, supplied-air suits may be quite useful.

For tritium work, supplied-air suits are constructed of materials that have acceptable permeation protection against HTO and provide good tear and abrasion resistance. Because of the closed environment, and the additional background noise caused by the flow of air into the suits, communication between personnel may require special equipment or methods.

Protection from Surface Contamination

Experience at tritium laboratories has shown that many tritium exposures to personnel occur as a result of contact with highly contaminated surfaces. Sudden and significant releases of airborne tritium occur mostly as the less toxic form HT and are quickly detected by portable or

strategically placed, fixed tritium monitors. The result is that the exposure and uptake of airborne tritium are minimized. (Heavy-water reactors, of course, present a more significant risk of exposure to tritiated water vapor than to tritium gas.) The presence and degree of contamination may be unknown until measurements are made. Consequently, the importance of routine and special monitoring surveys for surfaces that personnel might contact cannot be overestimated.

Protective clothing worn by workers is one of the most important aspects of an effective health physics program. Because tritium can be absorbed easily through the skin or by inhalation, personnel protective equipment must protect against both exposure routes. The following paragraphs describe protective measures and equipment.

Protective Clothing

Lab Coats and Coveralls

Lab coats and coveralls (fabric barriers) are worn in most tritium facilities. Lab coats are routinely worn to protect personal clothing. Coveralls are sometimes worn for added protection instead of a lab coat when the work is unusually dusty, dirty, or greasy. The protection afforded by lab coats and coveralls is minimal (except for short exposures) when tritium is airborne, but they are more effective in preventing skin contact with contaminated surfaces.

Disposable water-proof and water-resistant lab coats and coveralls have been tested at various laboratories. They are not popular for everyday use because of the cost and excessive discomfort inflicted on the worker. Most facilities prefer using ordinary open-weave fabrics for lab coats and coveralls and using an approved laundry for contaminated clothing. Some facilities have chosen to use disposable paper lab coats and coveralls, exchanging the costs associated with a laundry for the costs associated with replacement and waste disposal.

Shoe Covers

Although shoe covers provide protection against the spread of contamination and exposure, the routine use of shoe covers in a tritium facility is usually weighed against actual need. Shoe covers can offer both a degree of personnel protection and control over the spread of contamination on floors. However, in modern facilities where tritium is largely controlled by the use of secondary containment, shoe covers may not be required. Such facilities can easily maintain a clean laboratory environment by the use of regular smear surveys and good housekeeping. Using liquid-proof shoe covers until spills are cleaned up should be considered following spills of tritium-contaminated liquids and solids to prevent the spread of local contamination.

Gloves

In most operations, the hands and forearms of workers are vulnerable to contact with tritium surface contamination. The proper use and selection of gloves are essential.

Many factors should be considered in selecting the proper type of glove. These include chemical compatibility, permeation resistance, abrasion resistance, solvent resistance, glove thickness, glove toughness, glove color, shelf life, and unit cost. Gloves are commercially available in butyl rubber, neoprene, polyvinyl chloride (PVC) plastics, latex, etc.

The most common gloves found in tritium laboratories are the light-weight, disposable short glove (usually PVC or latex) used for handling lightly contaminated equipment. Depending on the level of contamination, such gloves may be changed frequently (every 10–20 minutes), a second pair may be worn, or heavier gloves may be used instead. When using gloves for this purpose, the work should be planned so that contaminated gloves do not spread contamination to surfaces that are being kept free of contamination.

When working in a glove box using the box gloves, disposable gloves are worn to prevent uptake of HTO contaminating the outside of the box gloves. Again, depending on the level of contamination, more than one additional pair may be required, one of which may be a longer, surgeon's length, glove.

In spite of all the precautions normally taken, workers may occasionally be contaminated with tritium. The skin should be decontaminated as soon as possible after any potential skin exposure to minimize absorption into the body. Effective personal decontamination methods include rinsing the affected part of the body with cool water and soap. If the entire body is affected, the worker should shower with soap and water that is as cool as can be tolerated. Cool water keeps the pores of the skin closed and reduces the transfer of HTO across the skin. The importance of washing the affected skin as soon as possible after contamination cannot be over-emphasized. Figure 8^d illustrates the effect of speed on reducing the uptake and the resultant dose. Even if gloves are worn when handling contaminated equipment or when working in contaminated glove box gloves, it is good practice to wash the hands after removing the gloves.

d. W. R. Bush, *Assessing and Controlling the Hazard from Tritiated Water*, AECL-4150, Atomic Energy of Canada LTD., Chalk River, Ontario, 1972.

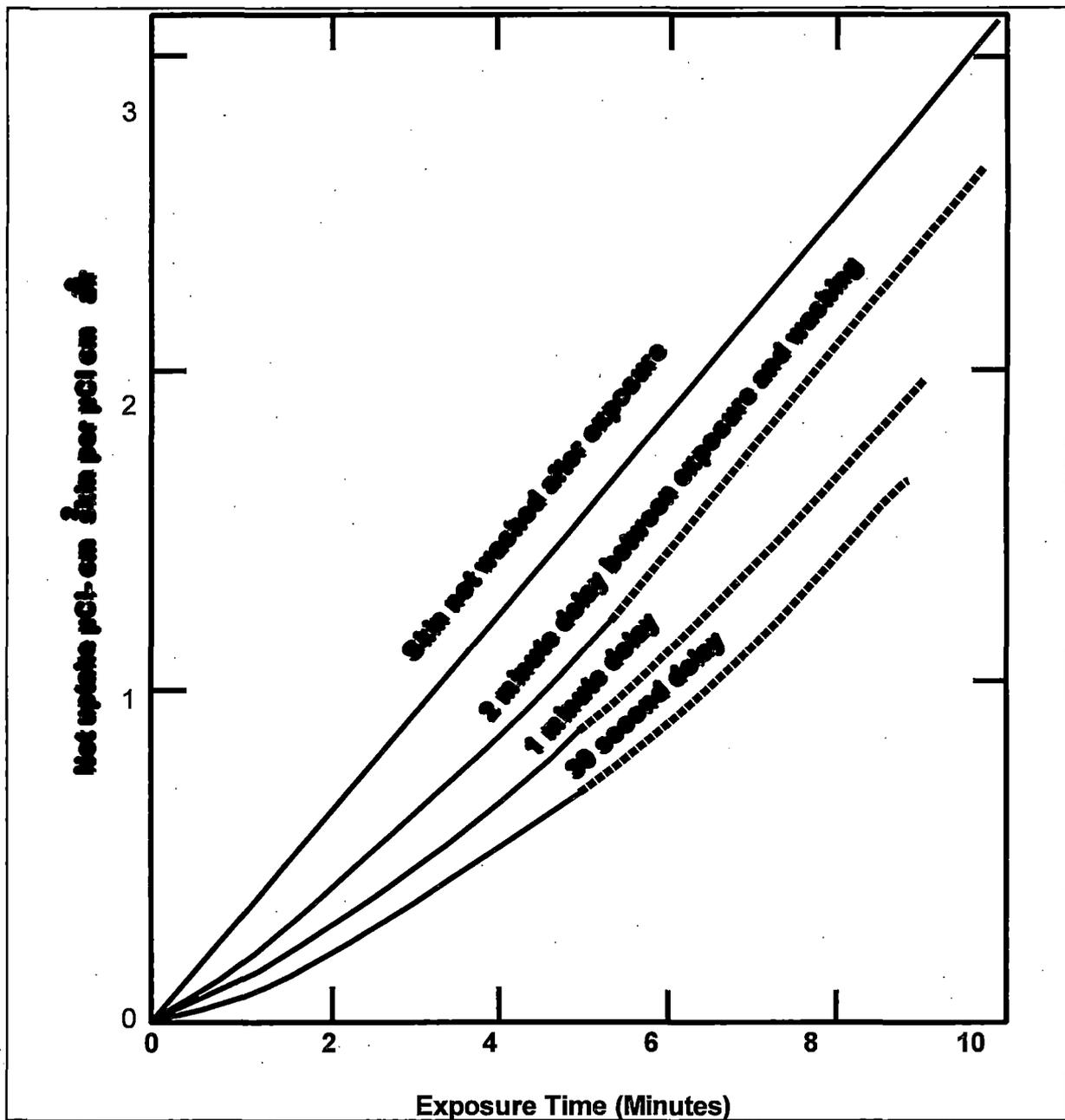


Figure 8 Reducing HTO uptake by washing after exposure to HTO vapor.

EMERGENCY RESPONSE

It is important to examine the history of accidents that have occurred in tritium facilities and to consider foreseeable unplanned events in order to minimize or mitigate their effects or to prevent their taking place at all. When an accident does occur, requirements for reporting accidents must be followed.

Facilities that handle significant quantities of radioactive material must have a site-specific emergency plan. All radiological workers at the site must be familiar with certain aspects of this plan. In addition, job assignments involving radiological hazards are typically covered by procedures and work permits that include steps for emergency situations that may arise during the course of the work. Radiological workers must be familiar with these procedures or be accompanied by a radiological control technician (RCT) to provide guidance in case of an emergency.

Emergency Steps to Take

The initial steps to be undertaken following a serious accident must always include the following:

- Warning others in the vicinity
- Evacuating the laboratory if an airborne release has occurred
- Requesting any necessary assistance
- Giving urgent first aid in the event of serious injuries (This should take priority over problems that arise from contamination)
- Starting personnel decontamination procedures
- Submitting urine samples following the schedule outlined for nonroutine samples.

Decontamination of Personnel

Personnel should be decontaminated by the following procedure:

- Remove clothing thought to be contaminated
- Wash hands with soap and cool water
- Wash other parts of the body (such as face, hair, and arms) that may have been exposed to tritium, or immediately shower with cool water and soap

- If mouth-to-mouth resuscitation must be given to a contaminated victim, the victim's mouth should first be wiped with a damp cloth.

Decontamination of Surfaces

Following a tritium spill involving a liquid with high specific activity, the area may have to be isolated and other protective measures taken before cleaning up the liquid. Monitoring for possible airborne tritium must be started to determine the need for respiratory protection or skin protection. After the spill has been cleaned up, residual contamination will remain. Depending on the level of contamination, any further steps needed to prevent the spread of contamination and reduce the level to an acceptable value should be determined.

Following a release of tritium gas, surfaces would not be expected to be heavily contaminated. If tritiated water vapor is released, the contamination may be greater, depending on the amount and activity of the released vapor. In any case, smear and air surveys will be used to determine the course of action needed to control and reduce the contamination safely.

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U.S. Department of Energy, *Occurrence Reporting and Processing of Operations Information*, DOE Order 5000.3B, January 19, 1993.

U.S. Department of Energy, *Radiation Protection of the Public and the Environment*, DOE Order 5400.5, Change 2, January 7, 1993.

CONCLUDING MATERIAL

Review activities:

DOE Facilities

ANL-W, BNL, EG&G Idaho,
EG&G Mound, EG&G Rocky Flats,
LLNL, LANL, MMES, ORAU, REECo,
WHC, WINCO, and WSRC.

DOE Program Offices

AD, DP, EH, EM, ER, NP, NS, RW.

DOE Field Offices

AL, CH, ID, NV, OR, RL, SR, OAK, RF.

Preparing activity:

DOE EH-63
Project Number 6910-0036

ENCLOSURE 1

**TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1**

TVA Response to NRC Request for Additional Information

Attachment 2

**WBNNAL3003, Revision 5, "Reactor Coolant and Secondary Side Activities
in Accordance with ANSI/ANS-18.1-1984"**

NPG CALCULATION COVERSHEET/CTS UPDATE

<u>REV 0 EDMS/RIMS NO.</u> B45 860107 235		<u>CTS TYPE:</u> Calculation	<u>EDMS TYPE:</u> CALCULATIONS (NUCLEAR)	<u>EDMS ACCESSION NO (N/A for REV. 0)</u> 977 130808 806	
Calc Title: Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984					
<u>ORG</u>	<u>PLANT</u>	<u>BRANCH</u>	<u>NUMBER</u>		<u>CUR REV</u>
NUC	WBN	NTB	WBNNAL3003		004
<u>CALC ID</u>					<u>NEW REV</u> 005
CTS UPDATE ONLY <input type="checkbox"/> (Verifier and Approval Signatures Not Required)			No CTS Changes <input type="checkbox"/> (For calc revision, CTS has been reviewed and no CTS changes required)		
<u>UNITS (check one)</u> 0 <input checked="" type="checkbox"/> 1 <input type="checkbox"/> 2 <input type="checkbox"/> 3 <input type="checkbox"/>		<u>SYSTEMS</u> N/A		<u>UNIDS</u> N/A	
<u>DCN.EDC.N/A</u> DCN 61599		<u>APPLICABLE DESIGN DOCUMENT(S)</u> N/A			<u>CLASSIFICATION</u> E
<u>QUALITY RELATED?</u> Yes <input checked="" type="checkbox"/> No <input type="checkbox"/>	<u>SAFETY RELATED?</u> (If yes, QR = yes) Yes <input checked="" type="checkbox"/> No <input type="checkbox"/>	<u>UNVERIFIED ASSUMPTION</u> Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	<u>SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS?</u> Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	<u>DESIGN OUTPUT ATTACHMENT?</u> Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	<u>SAR/TS and/or ISFSI SAR/CoC AFFECTED?</u> Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>
<u>CALCULATION NUMBER REQUESTOR</u> Name: _____ PHONE: _____		<u>PREPARING DISCIPLINE</u> N	<u>VERIFICATION METHOD</u> Design Review	<u>NEW METHOD OF ANALYSIS</u> <input type="checkbox"/> Yes <input checked="" type="checkbox"/> No	
<u>PREPARER (PRINT NAME AND SIGN)</u> Aleksandar Milicevic <i>[Signature]</i>		<u>DATE</u> 06/28/2013	<u>CHECKER (PRINT NAME AND SIGN)</u> Barry C. Schwartz <i>[Signature]</i>		<u>DATE</u> 6/28/2013
<u>VERIFIER (PRINT NAME AND SIGN)</u> Barry C. Schwartz <i>[Signature]</i>		<u>DATE</u> 6/28/2013	<u>APPROVAL (PRINT NAME AND SIGN)</u> <i>[Signature]</i> PATCH		<u>DATE</u> 8/7/13
<u>STATEMENT OF PROBLEM/ABSTRACT</u>					
<p>The methodology of ANSI/ANS-18.1-1984 was followed to calculate reactor coolant activities except for 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.</p> <p>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.</p> <p>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.</p>					
<u>MICROFICHE/EFICHE</u> Yes <input type="checkbox"/> No <input checked="" type="checkbox"/> <u>FICHE NUMBER(S)</u>					

NPG 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 3.7 μ Ci/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

NPG 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	<p>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.</p> <p>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.</p> <p>See DCN 61599 for SAR/Tech Spec impact determination.</p> <p>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</p> <p>The revision of all the pages has been changed to revision 5 and changes to the calculation are indicated by change bars.</p>

NPG CALCULATION TABLE OF CONTENTS		
Calculation Identifier: WBNNAL3003		Revision: 005
TABLE OF CONTENTS		
SECTION	TITLE	PAGE
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NPG CALCULATION VERIFICATION FORM			
Calculation Identifier	WBNNAL3003	Revision	005
Method of verification used:			
1.	Design Review	<input checked="" type="checkbox"/>	
2.	Alternate Calculation	<input type="checkbox"/>	Verifier B. C. Schwartz Date 6/28/2013
3.	Qualification Test	<input type="checkbox"/>	<i>B. C. Schwartz</i>
Comments:			
<p>I have reviewed Calculation WBNNAL3003 Revision 5 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 5, 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.</p>			

NPG COMPUTER INPUT FILE STORAGE INFORMATION SHEET			
Document	WBNNAL3003	Rev. 005	Plant: WBN
Subject: Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984			
<input type="checkbox"/> Electronic storage of the input files for this calculation is not required. Comments:			
<input checked="" type="checkbox"/> 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 <i>Calculation WBNNAL3003, Rev. 5.doc</i>)			
<input type="checkbox"/> Microfiche/eFiche			



Calculation No.	WBNNAL3003	Rev: 005	Plant: WBN	Page: 8
Subject:	Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984			

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.

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. These TPC tritium permeation rates are based on assumption 6.

Assumptions

- 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).
- 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.
- 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.
- A peak concentration of 2.5 $\mu\text{Ci/g}$ 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 $\mu\text{Ci/g}$ before dilution occurs in a normal cycle, i.e. no scrams or interruptions (ref.13).
- 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.
- A realistic and design basis TPC tritium permeation rate of 5 Ci/TPBAR/yr and 10 Ci/TPBAR/yr, respectively, is assumed.
Technical justification: The realistic permeation rate of 5 Ci/TPBAR/yr is acceptable because it bounds the observed permeation rate. The design basis permeation rate of 10 Ci/TPBAR/yr provides an additional factor of 2 margin and is therefore reasonable, but conservative and bounding.

Special Requirements/Limiting Conditions
none



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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 a realistic and design basis average tritium concentration in the reactor coolant. The realistic average tritium concentration is 5.1 $\mu\text{Ci/g}$ for 704 TPBARs and 12.0 $\mu\text{Ci/g}$ for 1900 TPBARs. This is roughly based on WBN having an average tritium concentration during normal operation of 1.0 $\mu\text{Ci/g}$ 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. The design basis average tritium concentration is 29.8 $\mu\text{Ci/g}$ for 2500 TPBARs and is also based on WBN having an average tritium concentration during normal operation of 1.0 $\mu\text{Ci/g}$ and projecting an approximate 29.8 times increase above the tritium production in a non-TPC core.

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)



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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 ft³/lb.

c) Reference 9 gives the weight of water in a steam generator as 4.745E7 g which translates to $4 \times 4.745E7 \text{ g} / 453.59 \text{ g/lb} = 4.184E5 \text{ lb}$. 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 3.7E4 lb/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 30,000 lb/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.



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- 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 ANSI-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)

$$\text{Lambda} = \ln(2) / (\text{Half Life})$$

$$R_{1i} = \frac{\text{FB} + (\text{FD} - \text{FB}) Y}{\text{WP}}$$

$$R_{2,3,6} = \frac{(\text{FD}) (\text{NB}_{2,3,6}) + (1 - \text{NB}_{2,3,6}) [\text{FB} + \text{FA} (\text{NA}_{2,3,6})]}{\text{WP}}$$

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

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

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

$R_{n6} = 0$ (this value comes directly from ANSI/ANS-18.1-1984)

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

$\text{NB}_2 = .99$ (this value comes directly from ANSI/ANS-18.1-1984)

$\text{NB}_3 = 0.5$ (this value comes directly from ANSI/ANS-18.1-1984)

$\text{NB}_6 = 0.98$ (this value comes directly from ANSI/ANS-18.1-1984)

$\text{NA}_2 = 0$ (this value comes directly from ANSI/ANS-18.1-1984)

$\text{NA}_3 = 0.9$ (this value comes directly from ANSI/ANS-18.1-1984)

$\text{NA}_6 = 0.9$ (this value comes directly from ANSI/ANS-18.1-1984)

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

$\text{NS}_2 = 0.01$ (this value comes directly from ANSI/ANS-18.1-1984)

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

$\text{NS}_5 = 1.0$ (this value comes directly from ANSI/ANS-18.1-1984)

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

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

$\text{NX}_3 = 0.5$ (this value comes directly from ANSI/ANS-18.1-1984)



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

$r_{n2,nc} = 0.17$ (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)

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

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

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



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TPC Average Concentration and TPBAR Failure: methodology and values from ref. 15
Tritium Sources = TPBAR releases + IFBA releases + Non-TPC sources

Realistic Source Term

$$\text{TPBAR releases} = \#\text{TPBARs} * 5 \text{ Ci/TPBAR/yr}$$

$$\text{IFBA releases} = 40 \text{ Ci/yr}$$

$$\text{Non-TPC sources} = 870 \text{ Ci/yr}$$

TPC coolant average concentration is the non-TPC average concentration (1 $\mu\text{Ci/g}$) multiplied by the ratio of the tritium sources for the TPC core to the normal tritium sources

$$C_{\text{TPC}} = (1 \mu\text{Ci/g}) (\#\text{TPBARs} * 5 + 40 + 870) / (870)$$

$$(1 \mu\text{Ci/g}) (704 * 5 + 40 + 870) / 870 = 5.1 \mu\text{Ci/g for 704 TPBARs}$$

$$(1 \mu\text{Ci/g}) (1900 * 5 + 40 + 870) / 870 = 12.0 \mu\text{Ci/g for 1900 TPBARs}$$

Increase in coolant average concentrations due to TPBAR failure:

$$\text{mass of RCS} = 5.4\text{E}5 \text{ lb} * 453.9 \text{ g/lb} = 2.45\text{E}8 \text{ g}$$

$$I_{\text{TPBARs}} = 11,600 \text{ Ci}$$

$$I_{\text{RCS}} = \text{RCS inventory prior to failure} = \text{average concentration} * \text{RCS mass}$$

$$5.1 \mu\text{Ci/g} * 2.45\text{E}8 \text{ g} = 1249.5 \text{ Ci for 704 TPBARs}$$

$$12.0 \mu\text{Ci/g} * 2.45\text{E}8 \text{ g} = 2940.0 \text{ Ci for 1900 TPBARs}$$

1 TPBAR Failure

$$C_{\text{new}} = (11,600 + 1249.5) \text{ Ci} / 2.45\text{E}8 \text{ g} = 52.45 \mu\text{Ci/g for 704 TPBARs}$$

$$C_{\text{new}} = (11,600 + 2940.0) \text{ Ci} / 2.45\text{E}8 \text{ g} = 59.35 \mu\text{Ci/g for 1900 TPBARs}$$

2 TPBAR Failure

$$C_{\text{new}} = (2 * 11,600 + 1249.5) \text{ Ci} / 2.45\text{E}8 \text{ g} = 99.79 \mu\text{Ci/g for 704 TPBARs}$$

$$C_{\text{new}} = (2 * 11,600 + 2940.0) \text{ Ci} / 2.45\text{E}8 \text{ g} = 106.69 \mu\text{Ci/g for 1900 TPBARs}$$

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

Design Basis Source Term

$$\text{TPBAR releases} = \#\text{TPBARs} * 10 \text{ Ci/TPBAR/yr}$$

$$\text{IFBA releases} = 40 \text{ Ci/yr}$$

$$\text{Non-TPC sources} = 870 \text{ Ci/yr}$$

TPC coolant average concentration is the non-TPC average concentration (1 $\mu\text{Ci/g}$) multiplied by the ratio of the tritium sources for the TPC core to the normal tritium sources

$$C_{\text{TPC}} = (1 \mu\text{Ci/g}) (\#\text{TPBARs} * 10 + 40 + 870) / (870)$$

$$(1 \mu\text{Ci/g}) (2500 * 10 + 40 + 870) / 870 = 29.8 \mu\text{Ci/g for 2500 TPBARs}$$

Increase in coolant average concentrations due to TPBAR failure:

$$\text{mass of RCS} = 5.4\text{E}5 \text{ lb} * 453.9 \text{ g/lb} = 2.45\text{E}8 \text{ g}$$

$$I_{\text{TPBARs}} = 11,600 \text{ Ci}$$

$$I_{\text{RCS}} = \text{RCS inventory prior to failure} = \text{average concentration} * \text{RCS mass}$$

$$29.8 \mu\text{Ci/g} * 2.45\text{E}8 \text{ g} = 7301.0 \text{ Ci for 2500 TPBARs}$$

1 TPBAR Failure

$$C_{\text{new}} = (11,600 + 7301.0) \text{ Ci} / 2.45\text{E}8 \text{ g} = 77.15 \mu\text{Ci/g for 2500 TPBARs}$$

2 TPBAR Failure

$$C_{\text{new}} = (2 * 11,600 + 7301.0) \text{ Ci} / 2.45\text{E}8 \text{ g} = 124.49 \mu\text{Ci/g for 2500 TPBARs}$$

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



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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).



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REACTOR COOLANT ACTIVITIES (continued)

	half life	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



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SECONDARY SIDE ACTIVITIES

Nuclider	removal rate	adjust. factor f'	adjust. factor f''	nominal		nominal	
				secondary water ANSI-18.1 uCi/gm	secondary water WBN uCi/gm	secondary steam ANSI-18.1 uCi/gm	secondary steam WBN uCi/gm
<u>Class 1</u>							
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
<u>Class 2</u>							
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
<u>Class 3</u>							
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
<u>Class 4</u>							
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).



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SECONDARY SIDE ACTIVITIES (continued)

Nuclider	removal rate	adjust. factor f'	adjust. factor f''	nominal		nominal	
				secondary water ANSI-18.1 uCi/gm	secondary water WBN uCi/gm	secondary steam ANSI-18.1 uCi/gm	secondary steam WBN uCi/gm
<u>Class 5</u>							
H-3	0.00E+00	1.00E+00	1.00E+00	1.0E-03	1.00E-03	1.0E-03	1.00E-03
<u>Class 6</u>							
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



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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	Secondary Water	Secondary Steam
	WBN uCi/gm	WBN uCi/gm	WBN uCi/gm
<u>Class 1</u>			
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
<u>Class 2</u>			
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
<u>Class 3</u>			
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
<u>Class 4</u>			
N-16	4.00E+01	1.29E-06	1.29E-07



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Nuclide	Reactor Coolant	Secondary Water	Secondary Steam
<u>Class 5</u>	uCi/gm	uCi/gm	uCi/gm
H-3	1.00E+00	1.00E-03	1.00E-03

Realistic Source Term

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''

Design Basis Source Term

TPC 2500	2.98E+01'	2.98E-02''	2.98E-02''
1 rod	7.72E+01	7.72E-02''	7.72E-02''
2 rod	1.24E+02	1.24E-01''	1.24E-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



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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.



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



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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
Realistic Source Term 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
Design Basis Source Term TPC 2500						2.98E+01'
1 rod						7.72E+01
2 rod						1.24E+02



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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.



Calculation No. WBNNAL3003	Rev: 005	Plant: WBN	Page: 24
Subject: Reactor Coolant and Secondary Side Activities in Accordance with ANSI/ANS-18.1-1984			

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
Realistic Source Term							
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'
Design Basis Source Term							
TPC-2500					2.98E-02'		2.98E-02'
1 rod					7.72E-02'		7.72E-02'
2 rod					1.24E-01'		1.24E-01'

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



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Appendix A (continued)
Steam Generator Replacement Results

Nuclide	removal	adjust.	adjust.	nominal	secondary	nominal	secondary
	rate	factor	factor	secondary	water	secondary	steam
	r	f'	f''	ANSI-18.1	WBN	ANSI-18.1	WBN
				uCi/gm	uCi/gm	uCi/gm	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.



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

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.

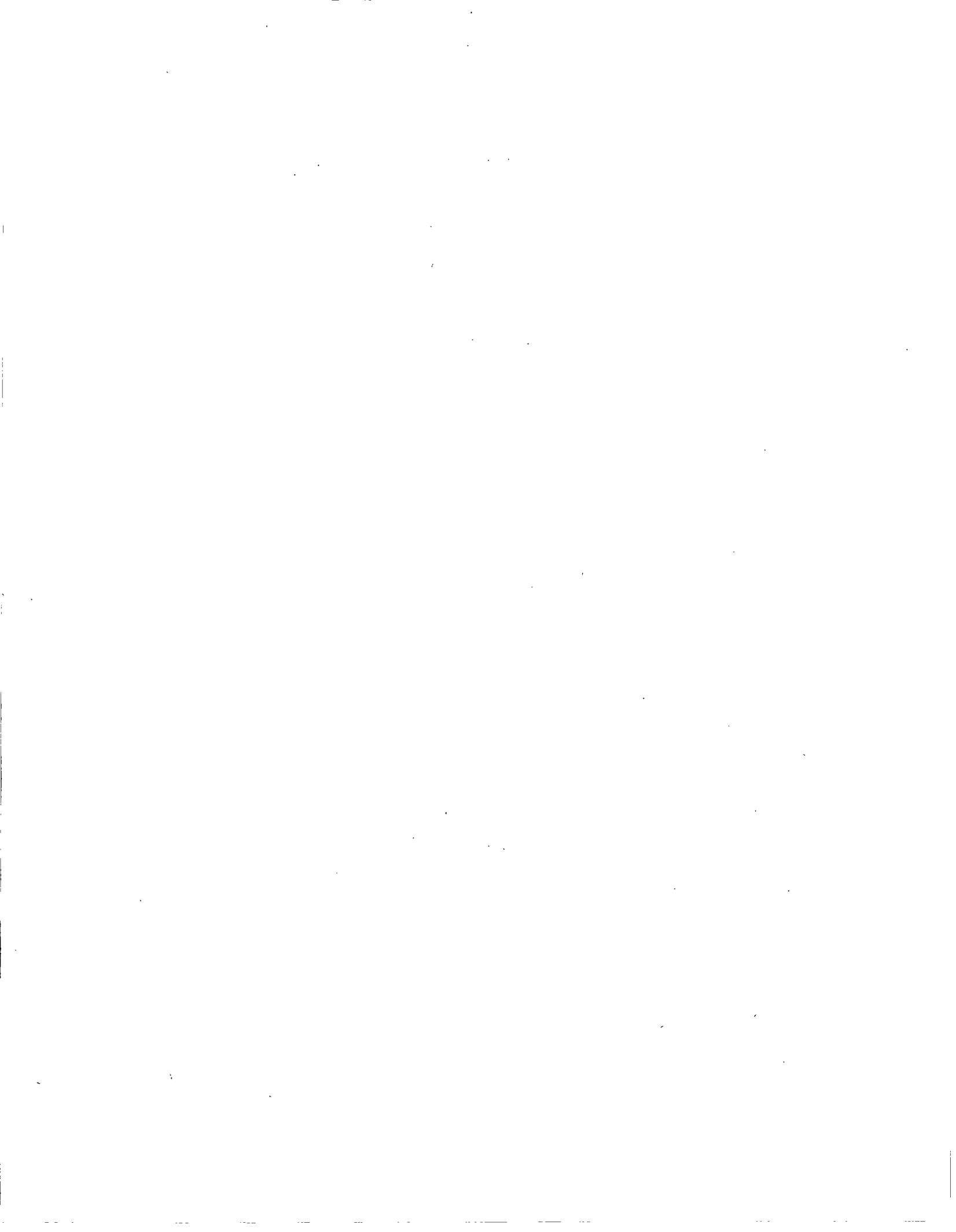
Topic: The reason for the deletion of several isotopes from ANSI/ANS-18.1-1984 isotope listing.

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.

Pages 28 to 49 of Attachment 2 contain a copy of American National Standard *ANSI/ANS-18.1 -1984*, "Radioactive Source Term for Normal Operation of Light Water Reactors," and have been redacted due to copyright restrictions.



ENCLOSURE 1

**TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
UNIT 1**

TVA Response to NRC Request for Additional Information

Attachment 3

WBNTSR100, Revision 12, "Design Releases to Show Compliance with 10CFR20"

NPG CALCULATION COVERSHEET / CTS UPDATE

REV 0 EDMS/RIMS NO. B26950110310		CTS TYPE: Calculation		EDMS TYPE: CALCULATIONS (NUCLEAR)		EDMS ACCESSION NO (N/A for REV. 0) 771 130718 804	
Calc Title: Design Releases to Show Compliance with 10CFR20							
	ORG	PLANT	BRANCH	NUMBER	CUR REV	NEW REV	
CALC ID	NUC	WBN	NTB	WBNTSR100	011	012	
CTS UPDATE ONLY <input type="checkbox"/> (Verifier and Approval Signatures Not Required)				NO CTS CHANGES <input type="checkbox"/> (For calc revision, CTS has been reviewed and no CTS changes required)			
UNIT (check one) 0 <input checked="" type="checkbox"/> , 1 <input type="checkbox"/> , 2 <input type="checkbox"/> , 3 <input type="checkbox"/>		SYSTEMS N/A		UNIDS N/A			
DCN.EDC.N/A DCN 61599		APPLICABLE DESIGN DOCUMENT(S) N/A				CLASSIFICATION EO	
QUALITY RELATED? Yes <input checked="" type="checkbox"/> No <input type="checkbox"/>	SAFETY RELATED? (If yes, QR = yes) Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	UNVERIFIED ASSUMPTION Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	SPECIAL REQUIREMENTS AND/OR LIMITING CONDITIONS? Yes <input checked="" type="checkbox"/> No <input type="checkbox"/>		DESIGN OUTPUT ATTACHMENT? Yes <input checked="" type="checkbox"/> No <input type="checkbox"/>	SAR/TS and/or I/SESL SAR/CoC AFFECTED Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	
CALCULATION NUMBER REQUESTOR Name: N/A		PREPARING DISCIPLINE PHONE: N/A		VERIFICATION METHOD N	NEW METHOD OF ANALYSIS Design Review <input type="checkbox"/> Yes <input checked="" type="checkbox"/> No		
PREPARER (PRINT NAME AND SIGN) Mehran A. Mohammadian		DATE 7/16/13	CHECKER (PRINT NAME AND SIGN) Aleksandar Milicevic		DATE 7/16/13		
VERIFIER (PRINT NAME AND SIGN) Aleksandar Milicevic		DATE 7/16/13	APPROVAL (PRINT NAME AND SIGN) SOV		DATE 7/17/13		
STATEMENT OF PROBLEM/ABSTRACT							
<p>Problem: 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).</p> <p>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 T1534 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.</p> <p>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.</p> <p>The presence of a Tritiated Water Storage Tank (TWST) has no affect on the results or conclusions of this calculation.</p> <p>The case of operating with a tritium production core (TPC) is also analyzed in this calculation. The gaseous releases are based on the design basis tritium (H-3) source term. For liquid releases, the maximum allowable liquid concentration of H-3 is determined. The maximum allowable liquid concentration of H-3 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 maximum allowable liquid concentration of tritium (H-3) released to the environment is 3.26E-04 µCi/cc. H-3 concentrations above 3.26E-04 µCi/cc released to the environment may result in the 10CFR20 limit being exceeded.</p>							
MICROFICHE/EFICHE		Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>		FICHE NUMBER(S)			

TVAN CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER WBNTR-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 WBNTR-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 pages 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 releases of tritium was changed in TI-594 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 WBNTR-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 Release to Show Compliance with 10CFR20	
Revision No.	DESCRIPTION OF REVISION
7	<p>Revision 7 of this calculation was created to add/update Unit 2 applicability. This calculation is applicable to Unit 2 based on the following:</p> <ul style="list-style-type: none"> • 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 Core/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. <p>Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy.</p> <p>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.</p> <p>FSAR AND TECHNICAL SPECIFICATIONS HAVE BEEN REVIEWED AND ARE NOT AFFECTED BY THIS REVISION OF THE CALCULATION.</p> <p>Reviewer: <i>R. White</i> 7-1-08</p> <p>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: 28 Pages</p>

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 WORKING~~ 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 TEM ARE NOT IMPACTED BY THIS REVISION.

Reviewer: *H. David Knight II* 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

← = PSAR SECTION 11.3 (SPECIFICALLY TABLES 11.3-B2 AND 11.3-B.C) 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	<p>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.</p> <ul style="list-style-type: none"> • 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. <p>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.</p> <p>Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy.</p> <p>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.</p> <p>The effect of Unit 2/dual unit operation on Unit 1 margins has been reviewed with no impact.</p> <p>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.</p> <p>Reviewer: <i>Meerik</i> 7/26/2011 J. H. KAENIK</p> <p>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</p> <p>Total number of pages in this revision including Attachments: 44 pages (Rev. 8) + 14 pages (Rev. 9) - 9 page (Rev. 9) = 49</p> <p>Appendix A: 4 pages Attachment 1: 1 page</p>

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	<p>Revision 10 of this calculation was created to incorporate updates to predecessor calculations</p> <ul style="list-style-type: none"> - 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 <p>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.</p> <p>Affected engineering judgments and assumptions were reviewed and (1) were found to be adequate, or (2) were revised as necessary to ensure adequacy.</p> <p>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.</p> <p>The affect of Unit 2/dual unit operation on Unit 1 margins has been reviewed with no impact.</p> <p>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</p> <p>Reviewer: <u>D.G.Fickey</u></p> <p>Pages added: 2d, 24, 25, 3a, 3b Pages revised/replaced: 1, 1a, 3-23, Attachment I Pages deleted: i, ia, ii, iii, 1b, 3a, 3b, 12a, 13a, 12b, 13a, 14a, 15a, 16a, 17a, 18a, 19a, 20a, Appendix A (4 pages)</p> <p>Total number pages in this revision: 49 pages (Rev.9)+5 pages(Rev.10)-21 page (Rev.10)=33 pages Attachment 1: 1 page</p>
11	<p>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</p> <p>Pages added: none Pages deleted: none Pages changed: 1, 1a, 2d, 3, 5, 25 Total Number of Pages: 33</p>

NPG CALCULATION RECORD OF REVISION	
CALCULATION IDENTIFIER WBNTSR100	
Title Design Releases to Show Compliance with 10CFR20	
Revision No.	DESCRIPTION OF REVISION
12	<p>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.</p> <p>The TPC 1 rod and 2 rod failure cases were removed from this revision because they do not reflect normal plant operation.</p> <p>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.</p> <p>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 $\mu\text{Ci/gm}$ 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 $\mu\text{Ci/gm}$. The units on the concentration column were changed to $\mu\text{Ci/cc}$.</p> <p>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.</p> <p>Pages replaced in Revision 12 contain an updated page header for Revision 12. All unaffected pages retain their original headers.</p> <p>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.</p> <p>See DCN 61599 for SAR/Tech Spec determination.</p> <p>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)</p>

NPG CALCULATION VERIFICATION FORM	
Calculation Identifier WBNTR100	Revision 12
Method of verification used:	
1. Design Review <input checked="" type="checkbox"/>	Verifier <u><i>Aleksandar Milicevic</i></u> Date <u>7/16/13</u> Aleksandar Milicevic
2. Alternate Calculation <input type="checkbox"/>	
3. Qualification Test <input type="checkbox"/>	
Comments: I have reviewed Calculation WBNTR100 Revision 12 and have found the portions of the calculation revised to be technically adequate. In conducting the verification of the portions of WBNTR100 revised as part of Revision 12, 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.	

NPG CALCULATION TABLE OF CONTENTS		
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Subject: Design Releases to Show Compliance with 10CFR20			

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.III.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 $\mu\text{Ci/gm}$ 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 effluent releases of tritium (H-3) are based on the design basis source term. For liquid releases of H-3, the maximum allowable liquid concentration of H-3 is back-calculated based on the available margin for the liquid release scenarios.

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 design basis source term (2500 tritium-producing burnable absorber rods (TPBARs) each with an average H-3 permeation rate of 10 Ci/TPBAR/yr) is used for the gaseous releases.

Technical Justification: Gaseous releases follow the tritium production rate, which is linear, so the average H-3 permeation rate is appropriate. 2500 TPBARs at 10 Ci/TPBAR/yr is assumed because it yields the largest, most conservative-source terms in ref.5.

Special Requirements/Limiting Conditions

The maximum allowable liquid concentration of tritium (H-3) released to the environment is $3.26\text{E-}04 \mu\text{Ci/cc}$. H-3 concentrations above $3.26\text{E-}04 \mu\text{Ci/cc}$ released to the environment may result in the 10CFR20 limit being exceeded.



Calculation sheet

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Subject: Design Releases to Show Compliance with 10CFR20			

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 $\mu\text{Ci/gm}$ I-131 dose equivalent is calculated by multiplying the equivalence factor from ref. 14 (11.348) and 0.265 $\mu\text{Ci/g}$ 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. ($\mu\text{Ci/gm}$)	Design RCS Conc. ($\mu\text{Ci/gm}$)	Des/Exp Ratio	Tech. Spec. 0.265 $\mu\text{Ci/gm}$ I-131 Equivalent ($\mu\text{Ci/gm}$)	Tech. Spec/ Design Ratio ($\mu\text{Ci/gm}$)
Kr-85m	1.71E-01	2.10E+00	12.28		
Kr-85	2.66E-01	8.80E+00	33.08		
Kr-87	1.81E-01	1.20E+00	7.45		
Kr-88	3.00E-01	3.70E+00	12.33		
Xe-131m	6.54E-01	1.80E+00	2.81		
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	6.04		
Xe-135	8.04E-01	6.30E+00	8.97		
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	0.877	0.752
I-133	1.49E-01	4.00E+00	26.85	0.448	0.112
I-134	3.84E-01	8.00E-01	1.85	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-138	9.08E-04	1.50E-01	165.20		
Cs-137	9.76E-03	1.50E+00	153.22		
Cr-51	3.26E-03	8.60E-04	0.29		
Mn-54	1.88E-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		
Sn-88	1.47E-04	3.30E-03	22.45		
Sr-90	1.28E-05	1.70E-04	13.49		
Sr-91	1.02E-03	1.90E-03	1.88		
Y-90	1.26E-05	2.00E-04	15.87		
Y-91	5.47E-08	6.10E-03	1115.17		
Zr-95	4.10E-04	7.00E-04	1.71		
Nb-95	2.85E-04	6.90E-04	2.34		
Mo-99	6.76E-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 010.



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Subject: Design Releases to Show Compliance with 10CFR20			

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/m³ (ref.4). To correct the units, the formula used is:

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

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.

Using a TPC with 2500 tritium-producing burnable absorber rods (TPBARs) with an average tritium permeation rate of 10 Ci/TPBAR/yr, referred to as the design basis source term (Assumption 2), the additional amount of tritium, based on the methodology established in TI-534, is (10 Ci/TPBAR/yr * 2500 TPBARs) 25,000 Ci/yr. Of this amount, 10% is available for gaseous release, so the additional amount of gaseous tritium for a TPC is (10% of 25,000 Ci/yr) 2500 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 (TPC) consists of the tritium value activity above added to the non-TPC's expected release value.



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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	1.78E-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)	2.64E+03	1.00E+00	2.64E+03	9.12E-10	1E-7	9.12E-03	9.60E-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)						3.20E-01	6.32E-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.



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Table 3: Gas Releases, Unfiltered Containment Vent Option

Nuclide	Exp. Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (μ Ci/cc)	10CFR20 (ECL, μ Ci/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	4.32E-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)	2.64E+03	1.00E+00	2.64E+03	9.12E-10	1E-7	9.12E-03	9.60E-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)						3.33E-01	6.57E-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.



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Table 4: Gas Releases, Continuous Filtered Containment Vent Option

Nuclide	Exp. Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (μ Ci/cc)	10CFR20 (ECL, μ Ci/cc)	Single Unit	Dual Unit
						Operation C/ECL	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)	2.64E+03	1.00E+00	2.64E+03	9.12E-10	1E-7	9.12E-03	9.60E-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)						2.79E-01	5.49E-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.



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III. Liquid Releases

The expected annual liquid release is taken from WBNTR093 (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 WBNTR093 = 16141.654 gal/day. Note: the WBNTR093 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:

$$[\mu\text{Ci/cc}] = [\text{Ci/yr}] * (1e6 \mu\text{Ci/Ci}) / ((16141.654 + 2.88E7 \text{ gal/day}) * 8.34 \text{ lb/gal} * 453.59 \text{ g/lb} * 365 \text{ day/yr})$$

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 $\mu\text{Ci/gm}$ 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, the liquid H-3 concentration peaks due to boron dilution requirements (feed and bleed). There is more dilution toward the end of the cycle compared to the beginning of the cycle. The competing effects of tritium production, due to TPBAR permeation and increasing primary coolant boron concentration, and dilution from the feed and bleed process lead to a peak in H-3 concentration at some point during the cycle. Therefore, using the average H-3 concentration is not conservative because there are times during the cycle that the calculated H-3 releases would be smaller than the actual H-3 releases. The expected peak H-3 concentration would result in the 10CFR20, App. B limit being exceeded, so the maximum liquid H-3 concentration that can be tolerated is determined (see Table 12 for back-calculation of H-3 concentrations). The value that appears in the following tables for H-3 (TPC) is the most limiting liquid H-3 release concentration as calculated in Table 12.



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Table 5: Liquid Release, No Processing

Nuclide	Exp.Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (µCi/cc)	10CFR20 (ECL, µCi/cc)	Single Unit	Dual Unit	Tech Spec	
						Operation Design C/ECL	Operation Design C/ECL	Single Unit Operation 0.265 µCi/cc	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)	N/A	N/A	N/A	3.26E-04*	1E-3	3.26E-01	3.26E-01		
Total						3.16E+00	6.32E+00	1.77E+00	3.55E+00
Total (TPC)						3.45E+00	6.58E+00	2.07E+00	3.82E+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.

* Value is the limiting liquid H-3 effluent concentration for a TPC calculated in Table 12. No expected release value was required.



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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 $1E-6 \mu\text{Ci/gm}$. From WBNNAL3003 (ref.5), the activity of some of the isotopes in the secondary side is greater than $1E-6 \mu\text{Ci/gm}$. 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:

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Table 6: Liquid Release, Mobile Demineralizer Processing

Nuclide	Exp.Rel. (Ci/yr)	Des/Exp Ratio	Design (Ci/yr)	Design (μ Ci/cc)	10CFR20 (ECL, μ Ci/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)	N/A	N/A	N/A	3.26E-04*	1E-3	3.26E-01	3.26E-01
Total						3.56E-01	7.11E-01
Total (TPC)						6.50E-01	9.74E-01

* 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.

* Value is the limiting liquid H-3 effluent concentration for a TPC calculated in Table 12. No expected release value was required.



Calculation sheet

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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 WBNTSR093 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 = $5E-7$ μ Ci/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 from 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 (16,141.854 - 3400 + 377,280 = 390,021.654). The concentration is calculated as follows:

$$[\mu\text{Ci/cc}] = [\text{Ci/yr}] * (1e6 \mu\text{Ci/Ci}) / ((390021.654 \text{ gpd} + 2.88E7 \text{ gal/day}) * 8.34 \text{ lb/gal} * 453.59 \text{ g/lb} * 365 \text{ day/yr})$$

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.

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

Nuclide	LRW (Ci/yr)	³ SGB Ci/yr Scaled to 0.26 Ci	Des/Exp Ratio	Des (Ci/yr)	Liquid (μCi/cc)	Liquid	Single Unit Operation C/ECL	Dual Unit Operation C/ECL
						10CFR20 (ECL, μCi/cc)		
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	3.66E-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)	N/A		N/A	N/A	3.26E-04*	1E-3	3.26E-01	3.26E-01
Total							3.50E-01	7.01E-01
Total (TPC)							6.45E-01	9.65E-01

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

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

* Value is the limiting liquid H-3 effluent concentration for a TPC calculated in Table 12. No expected release value was required.



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

Nuclide	LRW (Ci/yr)	² SGB Ci/yr Scaled to 4.40 Ci	Des/Exp Ratio	Des (Ci/yr)	Liquid (μCi/cc)	Liquid	Single Unit Operation C/ECL	Dual Unit Operation C/ECL
						10CFR20 (ECL, μCi/cc)		
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)	N/A		N/A	N/A	3.26E-04*	1E-3	3.26E-01	3.26E-01
Total							3.68E-01	7.36E-01
Total (TPC)							6.63E-01	1.00E+00

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

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

* Value is the limiting liquid H-3 effluent concentration for a TPC calculated in Table 12. No expected release value was required.



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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.

$$[\mu\text{Ci/cc}] = [\text{Ci/yr}] * (1e6 \mu\text{Ci/Ci}) / ((390,021.654 \text{ gpd}) * 8.34 \text{ lb/gal} * 453.59 \text{ g/lb} * 365 \text{ day/yr})$$

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

Nuclide	Des (Ci/yr)	Liquid	Liquid	Single Unit	Dual Unit
		Des (μ Ci/cc)	10CFR20 (ECL, μ Ci/cc)	Operation C/ECL	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	2.10E-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)	N/A	3.26E-04*	1E-3	3.26E-01	3.26E-01
Total				2.62E+01	5.25E+01
Total (TPC)				2.42E+01	4.81E+01

*Design Release from Table 7

* Value is the limiting liquid H-3 effluent concentration for a TPC calculated in Table 12. No expected release value was required.



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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 WBNTSR093. The concentration is calculated as follows in the third column:

$$[\mu\text{Ci/cc}] = [\text{Ci/yr}] * (1e6 \mu\text{Ci/Ci}) / ((377,280 \text{ gpd}) * 8.34 \text{ lb/gal} * 453.59 \text{ g/lb} * 365 \text{ day/yr})$$

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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	SGD at	SGD at	Liquid 10CFR20 (ECL, $\mu\text{Ci/cc}$)	Single Unit	Dual Unit
	LLD (Ci/yr)	LLD ($\mu\text{Ci/cc}$)		Operation C/ECL	Operation Liquid C/ECL
Br-84	8.15E-04	1.57E-09	4E-4	3.91E-08	7.83E-08
I-131	1.20E-02	2.31E-08	1E-8	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	8.60E-08	7E-8	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	6.33E-03	1.67E-02
Cs-136	4.74E-04	9.10E-10	6E-8	1.52E-04	3.03E-04
Cs-137	5.21E-03	1.00E-08	1E-8	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	2.38E-10	1E-5	2.38E-05	4.71E-05
Co-58	1.84E-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-8	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	1.83E-05	3.65E-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-06	7.11E-11	2E-3	3.65E-08	7.11E-08
Y-91	2.15E-06	4.13E-12	8E-6	5.16E-07	1.03E-06
Y-93	1.26E-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	2.18E-10	3E-5	7.20E-06	1.44E-05
Mo-99	2.68E-03	4.98E-09	2E-5	2.48E-04	4.96E-04
Tc-99m	1.16E-03	2.29E-09	1E-3	2.29E-08	4.58E-08
Ru-103	3.17E-03	6.06E-09	3E-5	2.03E-04	4.06E-04
Ru-106	3.78E-02	7.27E-08	3E-6	2.42E-02	4.85E-02
Tc-129m	7.98E-05	1.53E-10	7E-8	2.19E-05	4.38E-05
Tc-129	2.52E-03	4.85E-09	4E-4	1.21E-05	2.42E-05
Tc-131m	5.63E-04	1.08E-09	6E-8	1.35E-04	2.70E-04
Tc-131	3.39E-04	6.50E-10	8E-5	8.12E-08	1.62E-05
Tc-132	6.81E-04	1.31E-09	9E-8	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	6E-6	2.06E-03	4.11E-03
Ce-141	8.24E-05	1.20E-10	3E-5	3.99E-06	7.98E-06
Ce-143	1.04E-03	2.00E-09	2E-5	8.89E-05	2.00E-04
Ce-144	1.64E-03	3.14E-09	3E-8	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 ($\mu\text{Ci/cc}$) which would result in a 10 ECL release. The concentration is calculated as follows in the third column:

$$[\mu\text{Ci/cc}] = [\text{Ci/yr}] * (1e6 \mu\text{Ci/Ci}) / ((377,280 \text{ gpd}) * 8.34 \text{ lb/gal} * 453.59 \text{ g/lb} * 365 \text{ day/yr})$$

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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 ($\mu\text{Ci/yr}$)	Max App. I ($\mu\text{Ci/cc}$)	Liquid 10CFR20 (ECL, $\mu\text{Ci/cc}$)	Single Unit	Dual Unit	Normalized 10 C/ECL	Max $\mu\text{Ci/cc}$ for 10 ECL
				Operation	Operation		
Br-84	1.38E-02	2.65E-08	4E-4	6.62E-05	1.32E-04	4.55E-04	1.92E-07
I-131	2.03E-01	3.91E-07	1E-8	3.91E-01	7.81E-01	2.68E+00	2.68E-08
I-132	4.86E-01	9.34E-07	1E-4	9.34E-03	1.87E-02	6.42E-02	6.42E-08
I-133	9.82E-01	1.12E-08	7E-6	1.60E-01	3.19E-01	1.10E+00	7.87E-08
I-134	4.23E-01	8.12E-07	4E-4	2.03E-03	4.06E-03	1.38E-02	5.88E-08
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-08
Cs-134	6.81E-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	1.78E-02	1.06E-07
Cs-137	8.82E-02	1.69E-07	1E-6	1.89E-01	3.78E-01	1.18E+00	1.16E-06
Na-24	2.88E-01	6.15E-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	8.04E-05	1.73E-04	5.94E-04	2.87E-07
Mn-54	1.13E-02	2.18E-08	3E-5	7.20E-04	1.44E-03	4.85E-03	1.49E-07
Fe-55	8.49E-03	1.63E-08	1E-4	1.83E-04	3.26E-04	1.12E-03	1.12E-07
Fe-59	2.08E-03	3.99E-09	1E-5	3.89E-04	7.80E-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	1.66E-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	8.01E-07	1.20E-08	4.13E-08	8.27E-09
Y-91	3.94E-05	6.88E-11	8E-6	8.73E-08	1.75E-05	6.00E-05	4.80E-10
Y-93	2.16E-02	4.18E-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	2.86E-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	8.38E-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-08
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	2.28E-03	4.57E-03	1.57E-02	1.28E-07
Te-131	5.73E-03	1.10E-08	8E-5	1.37E-04	2.75E-04	9.46E-04	7.56E-08
Te-132	1.15E-02	2.21E-06	9E-6	2.48E-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	3.39E-08	2E-5	1.86E-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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The expected peak H-3 concentration in the primary coolant would result in the 10CFR20 limits being exceeded. Therefore, the maximum liquid H-3 concentration that can be tolerated is determined by considering the available liquid release margin remaining after all other isotopes have been accounted for. The "Dual Unit Operation Liquid C/ECL" totals from Tables 5 - 9 are used as a starting point because Dual Unit Operation values bound Single Unit Operation values. The remaining margin is calculated based on the following relationship, which accounts for the non-TPC H-3 so that a single H-3 release concentration may be obtained:

$$(C/ECL)_{\text{Remaining}} = 1.00E+00 - ((C/ECL)_{\text{Dual}} - (C/ECL)_{\text{non-TPC,H-3}})$$

where $(C/ECL)_{\text{Remaining}}$ is the available margin for the C/ECL fraction that does not contain H-3, $(C/ECL)_{\text{Dual}}$ is the "Dual Unit Operation Liquid C/ECL" total from Tables 5 - 9, $(C/ECL)_{\text{non-TPC,H-3}}$ is the Dual Unit C/ECL fraction attributable to non-TPC H-3 from Tables 5 - 9. If the $(C/ECL)_{\text{Remaining}}$ is negative, the combination of other radioactive liquid releases has already exceeded the 10CFR20 limits (Tables 5 and 9) and further calculations are not applicable.

The release concentration, which accounts for case-specific dilution, can be back-calculated by multiplying the remaining C/ECL margin as follows:

$$C_{\text{H-3,Released}} = (C/ECL)_{\text{Remaining}} * ECL_{\text{H-3}}$$

where $C_{\text{H-3,Released}}$ is the concentration of released H-3 for a TPC under case-specific dilution scenarios that would result in the 10CFR20, App. B limit being reached, and $ECL_{\text{H-3}}$ is the 10CFR20, App. B ECL for liquid releases of H-3 to the public in $\mu\text{Ci/cc}$.

Using the above two equations and the results from Tables 5 - 9 as explained above, Table 12 determines the concentration of released H-3 for a TPC with case-specific dilution parameters that would result in the 10CFR20, App. B limit being reached.

Table 12: Maximum Liquid H-3 Concentration Tolerable for Dual Unit Operation with a TPC

Table	Dual Unit Operation (C/ECL)	Non-TPC H-3 (C/ECL)	Remaining C/ECL Fraction (no H-3) (C/ECL)	H-3 ECL ($\mu\text{Ci/cc}$)	Maximum Liquid H-3 Concentration ($\mu\text{Ci/cc}$)
5 - Liquid Release, No Processing	3.55E+00	6.30E-02	-2.49E+00	1E-3	N/A
6 - Liquid Release, Mobile Demineralizer Processing	7.11E-01	6.30E-02	3.52E-01	1E-3	3.52E-04
7 - Direct SGBD Release/SGBD at LLD with 20,000 gpm Dilution	7.01E-01	6.22E-02	3.61E-01	1E-3	3.61E-04
8 - Direct SGBD Release/SGBD at Max App. I with 20,000 gpm Dilution	7.36E-01	6.22E-02	3.26E-01	1E-3	3.26E-04
9 - Direct SGBD Release/SGBD at LLD without 20,000 gpm Dilution	5.25E+01	4.65E+00	-4.68E+01	1E-3	N/A

From Table 12, the most limiting liquid H-3 concentration that could be released to the environment and still meet the 10CFR20, App. B limit is $3.26E-04 \mu\text{Ci/cc}$, which is associated with Table 8 (Direct SGBD Release/SGBD at Max App. I with 20,000 gpm Dilution). Therefore, the following statement is made a special requirement/limiting condition of this calculation:

The maximum allowable liquid concentration of tritium (H-3) released to the environment is $3.26E-04 \mu\text{Ci/cc}$. H-3 concentrations above $3.26E-04 \mu\text{Ci/cc}$ released to the environment may result in the 10CFR20 limit being exceeded.



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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 (0.320)	0.623 (0.632)	(with containment purge) - 2
	0.324 (0.333)	0.648 (0.657)	(with unfiltered containment venting) - 3
	0.270 (0.279)	0.540 (0.549)	(with continuous filtered containment venting) - 4
Liquid	3.16 (3.45*)	6.32 (6.58*)	(unprocessed cond.demin.waste) - 5
	1.77 (2.07*)	3.55 (3.82*)	(unprocessed cond.demin. iodine at 0.265 tech spec limits) - 5
	0.356 (0.650*)	0.711 (0.974*)	(processed cond.demin. waste) - 6
	0.350 (0.645*)	0.701 (0.965*)	(no SGB processing, SGB release limited to LLD) - 7
	0.368 (0.663*)	0.736 (1.00*)	(no SGB processing, SGB rel. limited to max App.I) - 8
	26.2 (24.2*)	52.5 (48.1*)	(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

* TPC value is based on the limiting liquid H-3 effluent concentration calculated in Table 12.

Maximum concentrations for no 20,000 gpm dilution, SGB only release path:
 for 10 ECL release: 5.81E-05 μ Ci/cc - table 11
 for 1 ECL release: 5.81E-06 μ Ci/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 μ Ci/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 μ Ci/gm (Table 1) but the technical specification is 0.265 μ Ci/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 μ Ci/cc gross gamma (Table 11), or even better, be restricted to the LLD (5E-7 μ Ci/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 μ Ci/cc gross gamma or less.

The concentration/ECL fractions due to implementing a TPC in Unit 1 are determined in this calculation. Gaseous effluent releases of H-3 are based on the design basis source term. Conclusions for the gaseous releases are unchanged for a TPC.

For liquid releases of H-3, the maximum allowable liquid concentration of H-3 that can be released to the environment before exceeding the 10CFR20, App. B limit is back-calculated based on the available margin for the liquid release scenarios. This is made a special requirement/limiting condition of the calculation: **The maximum allowable liquid concentration of tritium (H-3) released to the environment is 3.26E-04 μ Ci/cc. H-3 concentrations above 3.26E-04 μ Ci/cc released to the environment may result in the 10CFR20 limit being exceeded.** Maintaining the H-3 concentration released to the environment lower than this value will ensure that the liquid radwaste system meets the requirements of 10CFR20.



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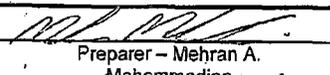
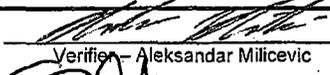
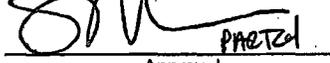
References

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 R5 "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
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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

Attachment A
NPG Calculation Design Output

EDMS T71130718804

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NPG CALCULATION DESIGN OUTPUT			
Calculation Identifier: WBNTR100	Engineering Change Document DCN 61599		
Revision 12			
Calculation Title: Design Releases to Show Compliance with 10CFR20			
 Preparer – Mehran A. Mohammadian	7/16/13 Date	 Verifier – Aleksandar Milicevic	7/16/13 Date
 Checker – Aleksandar Milicevic	7/16/13 Date	 Approval	8/6/13 Date
<p>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.</p> <p>The entire calculation WBNTR100 Rev. 12 is considered design output.</p> <p>A special requirement/limiting condition has been added to WBNTR100 Rev. 12:</p> <p>The maximum allowable liquid concentration of tritium (H-3) released to the environment is 3.26E-04 µCi/cc. H-3 concentrations above 3.26E-04 µCi/cc released to the environment may result in the 10CFR20 limit being exceeded.</p>			

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY WATTS BAR NUCLEAR PLANT UNIT 1

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

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. TVA will replace the containment isolation thermal relief check valves on the Watts Bar, Unit 1 supply lines to the containment for the Component Cooling Water System and Essential Raw Cooling Water System with simple relief valves and will replace one Watts Bar, Unit 1 Component Cooling Water System return line thermal relief check valve (1-CKV-70-698) with a simple relief check valve prior to increasing the number of TPBARs loaded in the reactor core above 704.
2. TVA will replace the WBN, Unit 1 upper compartment cooler cooling coils with fully qualified cooling coils to ensure ERCW System integrity during design basis events prior to increasing the number of TPBARs loaded in the reactor core above 704.
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 μ Ci/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 associated 0.01 μ Ci/ml (i.e., 10 μ Ci/g) action level will be specified by RCI-137, Table 3.1. In addition, RCI-137, Table 3.1 will be simplified as follows.

TRITIUM ACTION LEVELS					
Process Tritium Concentration (μ Ci/ml)	DAC, DAC-hrs	Mode of Exposure	Tritium Survey Requirements	Recommended 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
≥ 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	US NRC Regulatory Guide 8.32

				contact with air (such as a fuel pool).	
	≥ 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 shall be requested for any employee who exceeds this limit	RCDP-7, basis is 10 mrem/week, which is easily detected and verified by bioassay.

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

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