Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

June 11, 2002

TVA-SQN-TS-00-06 10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 AND 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 00-06, SUPPLEMENTAL RESPONSE REGARDING RADIOLOGICAL IMPACT (TAC NOS. MB2972 AND MB2973)

Reference: TVA letter to NRC dated September 21, 2001, "Sequoyah Nuclear Plant (SQN) - Units 1 and 2 - Revision of Instrumentation Measurement Range, Boron Concentration Limits, Reactor Core Limitations, and Spent Fuel Pool Storage Requirements for Tritium Production Cores (TPCs) - Technical Specification (TS) Change No. 00-06"

TVA submitted TS Change 00-06 to NRC by the referenced letter to propose changes to the SQN TSs that will accommodate the production of tritium. The enclosure to this letter provides supplemental information regarding the radiological impacts associated with irradiating Tritium Producing Burnable Absorber Rods. This information is being provided to assist in the review and approval of the proposed TS Change 00-06.

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There are no new commitments contained in this letter and the proposed TS change in the Referenced 1 letter is not altered by the enclosed information.

This letter is being sent in accordance with NRC RIS 2001-05. If you have any questions about this response, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 11th day of June

Notary Public

My Commission Expires Oct. 9, 2002

Enclosure

ENCLOSURE

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 and 2
DOCKET NOS. 327 AND 328

SUPPLEMENTAL RADIOLOGICAL IMPACT INFORMATION FOR TECHNICAL SPECIFICATION (TS) CHANGE 00-06

REVIEW OF RADIOLOGICAL CONSIDERATIONS FOR PRODUCTION OF TRITIUM AT SEQUOYAH NUCLEAR PLANT

TENNESSEE VALLEY AUTHORITY

JUNE 10, 2002

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Background

The U.S. Department of Energy (DOE) and the Tennessee Valley Authority (TVA) have agreed to cooperate in a program to produce tritium for the National Security Stockpile by irradiating Tritium Producing Burnable Absorber Rods (TPBARs) at TVA's Watts Bar Nuclear Plant (WBN) and, to the extent necessary, at TVA's Sequoyah Nuclear Plant (SQN).

The environmental impacts of producing tritium at WBN and SQN were assessed in a FINAL ENVIRONMENTAL IMPACT STATEMENT (EIS) FOR THE PRODUCTION OF TRITIUM IN A COMMERCIAL LIGHT WATER REACTOR (DOE/EIS - 0288, March 1999) prepared by DOE. TVA was a cooperating agency in the preparation of this EIS, and adopted the EIS in accordance with 40 CFR 1506.3(c) of the Council on Environmental Quality regulations. TVA's *Record of Decision* (ROD*) and Adoption of the Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor* was published in the Federal Register at 65 Fed. Reg. 26259 (May 5, 2000). In addition to the DOE EIS and TVA's ROD, a Tritium Production Core (TPC) Topical Report (NDP-98-181, Rev. 1) was prepared by DOE to address the safety and licensing issues associated with incorporating TPBARs in a PWR. The Nuclear Regulatory Commission's (NRC) Standard Review Plan (SRP) (NUREG-0800) was used as the basis for evaluating the impact of the TPBARs on a reference plant. The NRC reviewed the TPC Topical Report and issued a Safety Evaluation Report (SER) (NUREG-1672) to support plant-specific licensing of TPBARs in a PWR.

Radiological Impact Consideration

TVA conducted a review of the EIS and TVA ROD with a particular focus on evaluating the radiological impacts associated with the irradiation of TPBARs at SQN. This review addressed both the onsite and offsite potential radiological impacts of tritium production.

TVA's review utilized the EIS as the basis for the specific evaluations and analyses performed for SQN. Extensive analyses and evaluations of the environmental impacts of a CLWR incorporating TPBARS were documented in the EIS. This review did not reproduce the evaluations presented in EIS that showed no impact of TPBAR utilization in a CLWR. However, each section of the EIS was reviewed to ensure that the "no impact" conclusion continued to be valid for SQN. Plant-specific evaluations (and analyses if required) were performed for SQN using the equations and values given in the Sequoyah Updated Final Safety Analysis Report and Offsite Dose Calculation Manual. In addition, the review included identifying any significant differences between the EIS and the SQN Tritium Program license amendment associated with the TPBARs and assessed them for impacts. The noted differences are discussed in each section of this review, as appropriate, with the EIS values indicated in ***bold italics***.

CONCLUSION

Upon examination of the documents and the analyses described above, the review determined that there were no significant radiological impacts associated with the SQN tritium production program identified for either the plant staff or the offsite population. All calculated, radiological consequences continue to remain well below NRC regulatory requirements and acceptance criteria.

Radiological Impacts of the Proposed Changes

Tritium

Tritium is a radioactive isotope of hydrogen with a half-life of 12.323 years. Tritium undergoes beta decay, with a maximum energy of 18.6 KeV. The average energy is 5.7 KeV. This low energy limits the maximum range of a tritium beta to about 6 millimeters in air and 0.0042 millimeters in soft tissue. Therefore, the primary radiological significance of exposure to tritium is in the form of internal exposure.

Tritium Source Terms

Regarding tritium sources, in a non-TPC, the production of tritium in the Reactor Coolant System (RCS) is primarily the result of three processes:

- Ternary fission,
- Boron activation, and
- Lithium activation.

A review of Westinghouse Pressurized Water Reactors benchmark tritium data[1] indicates a nominal production/release tritium value of about 870 Ci/y/unit. This nominal value is consistent with the 854 Ci/y unit average tritium effluent total (TABLE 1 below) observed over the past five years (1997 – 2001) at WBN and SQN.

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

TABLE 1 Station Annual Liquid and Gaseous Tritium Effluents (Curies)[2]

SQN	LIQUID	GAS	TOTAL	GAS %
1997	1559.00	45.29	1604.29	2.82%
1998	1905.00	83.72	1988.72	4.21%
1999	998.00	34.26	1032.26	3.32%
2000	2832.40	62.65	2895.05	2.16%
2001	1323.60	40.16	1423.33	2.82%
STATION MEAN	1735.51	53.22	1788.73	2.98%
UNIT MEAN	867.76	26.61	894.37	2.98%
EIS REPORTED VALUE	714	25	739	
WBN	**LIQUID**	**GAS**	**TOTAL**	**GAS %**
1997	639.20	2.56	641.76	0.40%
1998	712.58	7.45	720.03	1.03%
1999	368.43	8.58	377.01	2.28%
2000	1116 00	14.70	1130.70	1.30%
2001	933.31	61.71	995.02	6.20%
STATION MEAN	753.90	19.00	772.90	2.46%
UNIT MEAN	753.90	19.00	772.90	2.46%
EIS REPORTED VALUE	639	5.60	644.6	
TVA	**LIQUID**	**GAS**	**TOTAL**	**GAS %**
PWR UNIT MEAN	829.81	24.07	853.88	2.82%
EIS CALCULATED MEAN VALUE	676.5	15.3	691.8	

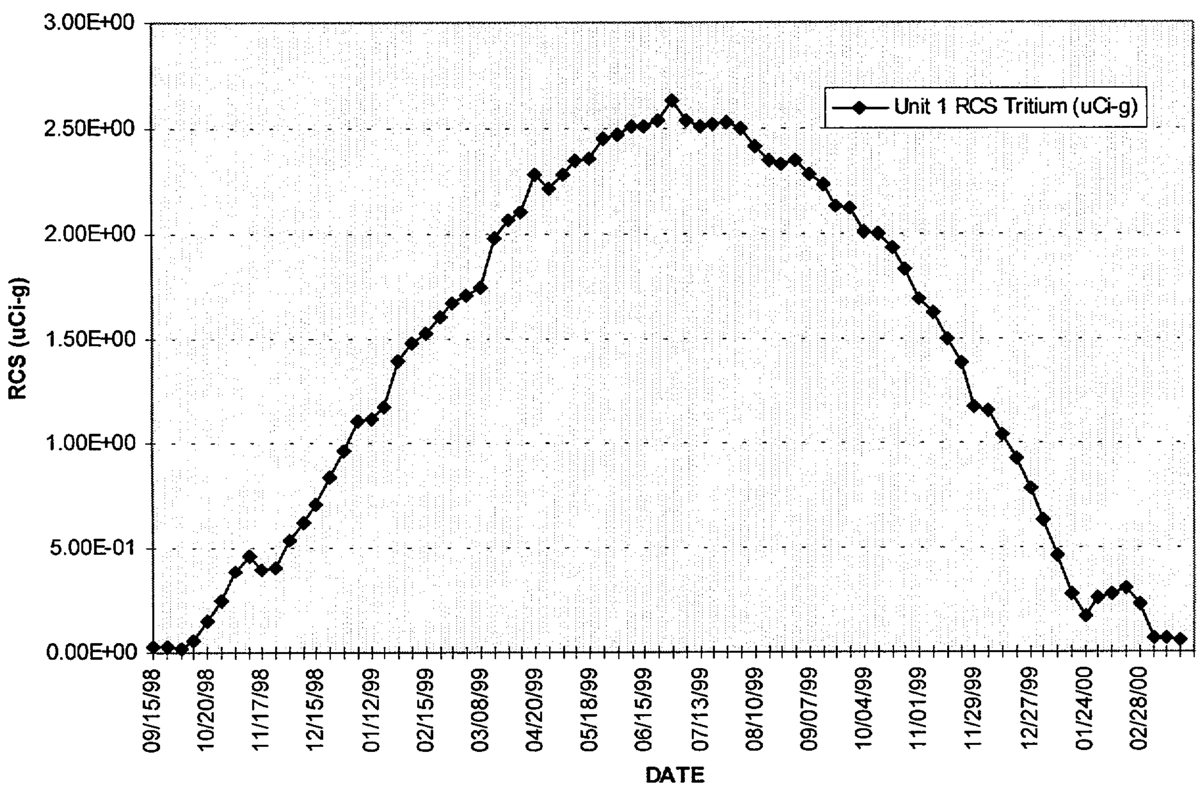
When reviewing station annual tritium effluents, it is important to recognize that plants such as WBN and SQN operate with 18-month fuel cycles which tend to generate more tritium early in the core cycle, owing to higher initial boron concentrations and/or burnable poisons and Integral Fuel Burnable Absorber rods that are required for reactivity control. This results in increasing concentration of tritium in the RCS during the first half of the fuel cycle when discharges from the RCS are relatively small since the amount of feed and bleed necessary to reduce the RCS boron

[2] Station Annual Radioactive Effluent Release Reports.

concentration is minimal. However, as the boron concentration is reduced and additional feed and bleed of the RCS is necessary to accommodate boron removal, the amount of primary coolant that is removed increases and the RCS tritium concentrations are reduced over the latter parts of the cycle.

The RCS tritium concentrations from SQN Unit 1 Cycle 10 are shown below in FIGURE 1 as an example. WBN Unit 1 Cycle 3 demonstrated a similar pattern.

FIGURE 1 Sequoyah Unit 1 Cycle 10 RCS Tritium Concentration Vs Time



TPBARs are designed and fabricated to retain as much tritium as possible within the TPBAR. Since the TPBAR produced tritium is chemically bonded within the TPBAR, virtually no tritium is available in a form that could permeate through the TPBAR cladding. However, it is assumed that while operating with a TPC, some of the tritium inventory in the TPBARs may permeate the cladding material and be released to the primary coolant. The design goal for this permeation process is less than 1,000 Ci per 1,000 TPBARs per year as a core average. Thus, a single TPBAR may release more than 1 Ci/year, but the total release for 1,000 TPBARs will be less than 1,000 Ci/year. As the TPC will contain up to 2,256 TPBARs at SQN, the total design basis tritium input from the maximum number of TPBARs is 2,256

Ci/y into the RCS. The design basis sources of tritium for the RCS, on a fuel cycle basis, are summarized in TABLE 2 below.

TABLE 2 Design Basis Sources of Tritium In The Primary Coolant For The SQN Tritium Production Core Operating Cycle (2,256 TPBARs)

Tritium Source	Curies[3]
Tritium Producing Burnable Absorber Rods	3,384 (design basis value, actual value will be developed based on operating experience)
Ternary Fission	1,770 (design basis value, actual value is estimated to be 350)
Integral Fuel Burnable Absorbers	40
Control Rods	95
Coolant soluble boron	460
Coolant soluble lithium	176
Deuterium	4
Total Design Basis Tritium	5,929

Along with the maximum design basis TPBAR permeation release, a potential release scenario involves the failure of one or more of the TPBARs. Although unlikely, it has been assumed that two TPBARs under irradiation would fail and the entire inventory of tritium would be released to the primary coolant. At the end of the operating cycle, the maximum available tritium in a single TPBAR is calculated to be about 11,600 Ci. While the occurrence of one or two failed TPBARs is considered to be beyond that associated with reasonable design basis considerations, the assumption of two failed TPBARs is discussed in the EIS.

The SQN TPC projected tritium source term values are summarized in TABLE 3 below:

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

TABLE 3 TPC Projected Tritium RCS Source Term Values

RCS Tritium Sources	Estimated Annual Tritium Release to RCS (Ci)	Estimated Peak RCS Tritium Concentration (μCi/g)
Non-TPC with nominal tritium release	870[4]	≈ 2.5[5]
Non-TPC with design tritium release	1,826[6]	≈ 5.3
TPC with nominal tritium release and design basis permeation from 2,256 TPBARs	3,126	≈ 9.0[7]
Abnormal Operations		
TPC with nominal tritium release, design basis permeation from TPBARs and one TPBAR failure having instantaneous release at end of operating cycle	14,726	≈ 53[7]
TPC with nominal tritium release, design basis permeation from TPBARs and two TPBAR failures having instantaneous release at end of operating cycle	26,326	≈ 105[7]

TVA has performed an evaluation (see **Tritium Impacts on Station Operation**, below) and determined that for normal TPBAR operation (permeation only), TVA will maintain normal RCS feed and bleed operation for boron control throughout the cycle. Primary coolant discharge volumes with a TPC will therefore be comparable with current plant practice. The maximum tritium level in the RCS is anticipated to be approximately 9 μCi/g.

[4] Ibid., Westinghouse Electric Company.

[5] Observed RCS tritium values Watts Bar Unit 1 Cycle 3 and Sequoyah Unit 1 Cycle 10.

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

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

Tritium Impacts on Station Operation

Normal Operation

Site-specific data collected by Chemistry during recent extended operating cycles (Watts Bar Unit 1 Cycle 3 and Sequoyah Unit 1 Cycle 10) have provided useful data to estimate the impact from tritium production on station radiological conditions. The RCS maximum tritium levels noted during the extended operating cycles were ≈ 2.5 μCi/g with a cycle RCS tritium mean of ≈ 1.0 μCi/g. The TVA experienced end of cycle (pre-flood up) RCS tritium values have typically been in the 0.1 - 0.3 μCi/g range for both WBN and SQN. The post-flood up tritium values have typically been in the mid 10^{-2} μCi/g range. The extended cycle tritium peak RCS tritium values of ≈ 2.5 μCi/g have resulted in containment peak tritium Derived Air Concentration (DAC)-fractions of <0.15 for both WBN and SQN with a containment average DAC-fraction of about 0.08. It is understood that containment tritium DAC values are a function of the RCS tritium activity, the transfer of tritium from the RCS to the containment atmosphere (leak rate), and the turnover/dilution of the containment atmosphere through periodic and continuous containment venting and purging.

The projected tritium release to the RCS with a TPC containing TPBARs releasing tritium at the design maximum rate will result in about a factor of four increase over the current tritium production rate, that is,

Ratio = (TPC) 3,126 Ci/yr/ (Nominal Core) 870 Ci/yr = 3.6.

By extrapolation it has been calculated that with no modifications to TVA's current boron-control feed and bleed methodologies, the design basis RCS maximum tritium values will approximate 9 μCi/g with a cycle mean of ≈ 3.6 μCi/g. These values would indicate an estimated containment peak tritium DAC-fraction of ≈ 0.6 and an average containment tritium DAC-fraction of about 0.3. The design basis estimated containment average tritium DAC-fraction equates to an effective dose rate of about 0.7 mrem/h.

As previously discussed, the primary radiological significance of exposure to tritium is in the form of internal exposure and a potential hazard arises when personnel are exposed to open processes that have been wetted with tritiated liquids. Therefore, the design features of the plant that deal with contamination and airborne radioactivity control such as drain and ventilation systems are potentially challenged. TVA has concluded there will be minimal impact on estimated annual Total Effective Dose Equivalent (TEDE) values. TVA, using the site-specific data collected during recent extended operating cycles, has evaluated the additional deep-dose equivalent to select station personnel during TPBAR consolidation and the additional committed effective dose equivalent from possible increased tritium airborne activity in containment.

TVA's current estimate of the TPBAR cycle work scope includes pre-cycle preparation activities, post cycle removal and handling activities, TPBAR consolidation (including equipment setup and disassembly) and shipping activities, and the processing, packaging, and shipping of the irradiated components for an estimated total of 2,500 man-hours in a 1 mrem/hour radiation field.

The EIS estimated the incremental annual station dose (adjusted for 2,256 TPBARs) to be 0.79 rem and did not include estimated exposure for TPBAR handling and consolidation activities.

TVA estimates that on a TPC basis, this additional TEDE is about 1.7 rem per year for TPBAR handling and consolidation activities (2.5 rem per TPC cycle) and 1.5 rem per year for the additional committed effective dose equivalent from possible increased tritium airborne activity in containment. This estimated additional 3.2 rem per year is an increase of 2.2% of the current SQN unit dose assessment of 145 rem[8], an amount that remains bounded by the station dose assessment of record. Given this small additional increase, this value is considered insignificant.

Abnormal Operation

In order to consider the unlikely event of TPBAR abnormal operation, TVA established two tritium RCS action levels: > 9 μCi/g and > 15 μCi/g. The lower action level requires more frequent sampling (once/day) to monitor the RCS tritium levels. In the unlikely event that the higher action level is exceeded, TVA will take further action to minimize the onsite and offsite radiological impacts of abnormal RCS tritium levels. These actions may include, but not be limited to, initiating actions to determine cause, more frequent tritium monitoring of RCS as well as other potentially impacted areas such as containment, increased feed and bleed of the RCS to reduce the tritium concentration, and the temporary onsite storage of tritiated liquids to ensure that the discharge concentration limits are met. The action levels described above will be used in response to any abnormal increases of the tritium levels in the RCS.

The EIS estimated the incremental station dose associated with the failure of two TPBARS to be 8.2 rem. TVA considers the additional station dose from the unlikely failure of two TPBARs to be bounded by this estimate.

[8]Sequoyah Nuclear Plant, *Updated Final Safety Analysis Report (UFSAR).*

Tritium Impacts on Public Dose

Normal Operation

Using the revised TPC source terms, the offsite radiation incremental tritium doses calculated for releases of radionuclides in liquid and gaseous effluents during normal two unit operation are summarized in TABLE 4. This table also lists the values previously estimated in the EIS.

TABLE 4 Annual Projected Impact of TPC on Effluent Dose to Maximally Exposed Members of the Public and Total Public Dose

	Current Core	EIS Data	Revised TPC
Normal Site Operations		Two Unit X 2,256 TPBARS Incremental Tritium Dose	Two Unit X 2,256 TPBARS Incremental Tritium Dose
Annual Radioactive Gaseous Emissions			
Maximally Exposed Individual (mrem)	8.76	0.066	0.26
50-mile Population Dose (Rem)	13.78	0.722	1.100
Annual Radioactive Liquid Emissions (			
Maximally Exposed Individual (mrem)	0.480	0.00722	0.000
50-mile Population Dose (Rem)	7.000	1.850	2.600

The NRC annual effluent per unit exposure guidelines are 15 mrem to the maximally exposed individual for radioactive gaseous emissions and 3 mrem to the maximally exposed individual for radioactive liquid emissions. The above Table demonstrates that the calculated SQN station effluent doses are already well below the NRC acceptance criteria, so the small increase in the reactor coolant activity from the tritium and resultant environmental releases would have a negligible effect on the offsite doses, which continue to remain well below the NRC's acceptance criteria. These values are considered to be insignificant.

Abnormal Operation

In addition to the maximum design basis TPBAR permeation release, a potential release scenario involves the failure of one or more of the TPBARs. It has been conservatively assumed that two TPBARs under irradiation would fail and the entire inventory of tritium would be released to the primary coolant. At the end of the operating cycle, the maximum available tritium in a single

TPBAR is calculated to be about 11,600 Ci. While the occurrence of one or two failed TPBARs is considered to be beyond that associated with reasonable design basis considerations, the assumption of two failed TPBARs is discussed in the EIS.

For analysis purposes, it is assumed that this entire inventory of tritium is released to the environment. Even in this extreme case, the doses from liquid and airborne effluent release would remain below applicable ODCM limits, and tritium release concentrations would remain within 10 CFR 20 and ODCM release limits.

The offsite radiation doses calculated for releases of radionuclides in liquid and gaseous effluents for two unit TPC operation with a single unit abnormal operation are summarized in TABLE 5 below. This table also lists SQN's regulatory established radioactive effluent design objectives and the values previously estimated in the EIS.

TABLE 5 Projected Impact of Two TPBAR Failure on Effluent Dose to Maximally Exposed Members of the Public and Total Public Dose

	Current Core	EIS Data	Revised TPC	NRC Annual Effluent per Unit Exposure Guideline
Abnormal Operations		***Two TPBAR Failure Incremental Tritium Dose***	Two TPBAR Failure Incremental Tritium Dose	
Annual Radioactive Gaseous Emissions (per unit)				
Maximally Exposed Individual (mrem)	4.38	***0.36***	1.46	15.00
50-mile Population Dose (Rem)	6.89	***3.67***	12.31	NA
Annual Radioactive Liquid Emissions (per site)				
Maximally Exposed Individual (mrem)	0.480	***0.037***	0.04	3.00
50-mile Population Dose (Rem)	7.00	***9.19***	14.000	NA

The above table demonstrates that the calculated SQN station effluent doses are already well below the NRC acceptance criteria, so that even in the unlikely abnormal event of two TPBARs failing, the increase in the reactor coolant activity from the tritium and resultant environmental releases would have a negligible effect on the offsite doses, which continue to remain well below the NRC's acceptance criteria. These values are therefore considered to be insignificant.

Solid Radioactive Waste

For normal TPC operations, the additional solid waste associated with TPCs that TVA will need to handle will be the base plate and thimble plug assemblies that remain after consolidation. TVA will consolidate and temporarily store these items on-site. Offsite shipment and ultimate disposal is assumed in accordance with agreements between TVA and DOE. The estimated activity inventory associated with these additional irradiated components[9] (96 base plates and 48 thimble plugs per TPC) when adjusted to reflect measured dose rate from BP-263, Base Plate with 24 Thimble Plugs following 113 day decay adjusted to 180 days (WBN Survey 010201 #2) is 4,052 curies per cycle (180 day post irradiation decay) or an average of 2,701 curies per year. The estimated disposal volume of this additional solid waste is 50 cubic feet per TPC operating cycle or an average of 33.3 cubic feet per year. ***The EIS estimated this annual volume to be 15 cubic feet per year.*** This additional volume is an insignificant increase in the SQN annual estimated solid waste (UFSAR), from 43,550 cubic feet per year to 43,565 cubic feet per year.

TVA's current estimate of the TPBAR cycle work scope includes pre-cycle preparation activities, post cycle removal and handling activities, TPBAR consolidation (including equipment setup and disassembly) and shipping activities, and the processing, packaging, and shipping of the irradiated components for an estimated total of 2,500 man-hours in a 1 mrem/hour radiation field. ***The EIS estimated the incremental annual station dose (adjusted for 2,256 TPBARs) to be 0.79 rem and did not include estimated exposure for TPBAR handling and consolidation activities.*** TVA estimates that on a TPC basis, this additional TEDE is about 1.7 rem per year for TPBAR handling and consolidation activities (2.5 rem per TPC cycle). This estimated additional 1.7 rem per year is an increase of 1.1% of the current SQN unit dose assessment of 145 rem[10], an amount that remains bounded by the station dose assessment of record. Given this small additional ManRem increase for TPBAR handling, consolidation, processing, packaging, and shipping activities, the impact of the increased curies associated with the irradiated components is considered insignificant.

For abnormal TPC operation, where increased feed and bleed operation may be used to reduce tritium levels in the RCS, the increased resins that may result from the increased feed and bleed operation will be stored at TVA in suitable containers. Offsite shipment and ultimate disposal will be according to established agreements between TVA and DOE. The amount of increase associated with abnormal TPC operation is estimated to be an additional 600 Ci and an additional 30 cubic feet. This additional volume is an insignificant increase in the SQN annual estimated solid waste (UFSAR), from 43,550 cubic feet per year to 43,580 cubic feet per year.

Spent Fuel Generation and Storage

The EIS assessed the environmental impact from the storage of additional spent fuel associated with the production of 3,400 TPBARs. The number of additional fresh fuel bundles per cycle due to tritium production was set a 56. The SQN license amendment establishes 2,256 as the maximum number of TPBARs per cycle. This level of TPBAR irradiation will require approximately 16 additional fresh fuel bundles per cycle. Thus, the tritium production additional spent fuel generation environmental impact is bounded by the EIS impact assessment.

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

[10] Sequoyah Nuclear Plant, *Updated Final Safety Analysis Report (UFSAR).*

Tritium Impacts on Station Accident Analysis

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

Condition I:	Normal Operation and Operational Transients
Condition II:	Faults of Moderate Frequency
Condition III:	Infrequent Faults
Condition IV:	Limiting Faults

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

TPBARs were designed to withstand the rigors associated with category I through IV events, therefore, no TPBAR failures are predicted to occur during design-basis accidents except for a large break loss of cooling accident (LBLOCA) or a fuel handling accident.

Radiological Consequences of Accidents

The analysis of thyroid, beta-skin, and gamma whole body doses, resulting from events leading to fission product release, are discussed below. To appropriately account for the radiological consequences of the increased tritium in the TPC, TVA has included calculated Total Effective Dose Equivalent (TEDE) values in the accident analysis. In those instances where the event was previously analyzed in the EIS, these data are included for comparison. TVA's review confirmed that no significant environmental impact would occur as the result of any design basis accident. The calculated station offsite accident doses will be below the NRC's acceptance criteria.

Loss of AC Power

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage conservatively assumed to be equal to the alternate repair criteria limit of 3.7 gpm per steam generator. This analysis incorporates conservative assumptions associated with both a pre-existing iodine spike and the situation in which the event triggers an iodine spike.. In addition failure of two TPBARs was assumed yielding an RCS Tritium level of 98.4 μCi/cc. The calculated offsite doses are well within the 10 CFR 100 limits[11].

[11] Westinghouse Electric Company, TVA-01-234, Radiological Consequences Of Accidents For The Sequoyah Nuclear Plant Units 1 And 2 Addressing Implementation Of Tritium Producing Burnable Absorber Rods (TPBARS).

TVA's review confirmed that no significant environmental impact would occur as the result loss of normal AC power to the plant auxiliaries. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well within the NRC's acceptance criteria.

Waste Gas Decay Tank (WGDT) Failure

The gaseous waste processing system is designed to remove fission product gases from the reactor coolant. The system consists of a closed loop with waste gas compressors, waste gas decay tanks for service at power and other waste gas decay tanks for service at shutdown and startup.

The maximum amount of waste gases stored occurs after a refueling shutdown at which time the gas decay tanks store the radioactive gases stripped from the reactor coolant.

A waste gas decay tank (GDT) is assumed to develop a leak immediately after a reactor shutdown in which the reactor coolant noble gas inventory has been stored in the tank. Activity is released to the outside atmosphere without any credit for filtration. The noble gas and iodine activity contained in the GDT is assumed to be unchanged from the existing analysis reported in the FSAR. In addition, consideration is included of tritium in the GDT.

This accident was described in the EIS as the non-LOCA Design Basis Accident. The offsite radiation doses calculated for the unexpected and uncontrolled release of the tank's contents are summarized in TABLE 6 below. This table also lists SQN's regulatory established applicable accident design dose limits and the values previously estimated in the EIS. ***The EIS assumed that the tritium content of the WGDT would be approximately 345 curies at the time of the postulated failure.*** In the current analysis TVA has assumed that the tritium content of the WGDT would include curies from normal operations and additionally would contain curies of tritium from the failure of two TPBARs for a total of 2,450 curies of tritium released at the time of the postulated tank rupture.

The consequences have been analyzed and it has been determined that the 30 day Low Population Zone offsite doses would be well within the 10 CFR 100 limits. For the 2-hour Exclusion Area Boundary /Site Boundary offsite dose to be less than the recommendation of NUREG-0800 (0.500 rem), the Xe-133 equivalency will be administratively controlled to ensure that the content of the waste gas decay tank rupture/release will not exceed regulatory requirements.[12].

TVA's review confirmed that no significant environmental impact would occur for a WGDT rupture accident. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant

[12] Ibid., Westinghouse Electric Company.

environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well within the NRC's acceptance criteria.

TABLE 6 Radiological Consequences of a Non-LOCA Design Basis Accident (rem)

	Current Core	EIS Data Adjusted for 2,256 TPBARs	Revised TPC	NRC 10 CFR 100 Exposure Limits
2 Hour Exclusion Area Boundary				
Thyroid	.039	***0.0124***	0.039	300
Beta-Skin	4.7	***Not Reported Separately, included in Whole Body Value***	4.7	300
Whole Body (Gamma)	1.77	***0.000722***	1.77	25
TEDE	1.78	***Not Analyzed***	1.86	
30-Day Low Population Zone				
Thyroid	0.0047	***0.00147***	0.0047	300
Beta-Skin	0.56	***Not Reported Separately, included in Whole Body Value***	0.56	300
Whole Body (Gamma)	0.2115	***0.000088***	0.2115	25
TEDE	0.22	***Not Analyzed***	0.23	

Loss of Coolant Accident

Loss-of-coolant accidents (LOCAs) are accidents that would result from the loss of reactor coolant at a rate more than the capability of the reactor coolant makeup system. LOCAs could occur from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (RCS). Large breaks are defined as breaks in the reactor coolant pressure boundary having a cross-sectional area greater than or equal to 1.0 ft^2. For conservatism, the entire contents of the end of life 2,304 (1.2 grams of tritium (11,600 curies)) TPBARs are assumed by TVA to be released into containment. The SER assumed 0.9 grams of tritium (8,700 curies) and the ***EIS assumed 1 gram of tritium (9,640 curies)***.

The total tritium released into containment assumption (for 2,304 TPBARs) was 2.0E+7 curies in the SER, ***2.2E+7 curies in the EIS***, and 2.7E+7 curies in the current calculation.

In accordance with General Design Criterion 19, the control room ventilation system and shielding have been designed to limit the whole body gamma dose during an accident period to 5 rem, the thyroid dose to 30 rem and the beta skin dose to 30 rem.

This accident was described in the EIS as the LOCA Design Basis Accident. The offsite radiation doses calculated for the loss of reactor coolant at a rate more than the capability of the reactor coolant makeup system are summarized below. ***The EIS did not analyze the dose to the Control Room Operator***. TABLE 7 also lists SQN's regulatory established applicable accident design dose limits and the values previously estimated in the EIS.

The 30 day Low Population Zone and 2-hr Exclusion Area Boundary /Site Boundary LOCA offsite doses were calculated to be below the 10 CFR 100 limits of 25 rem gamma, 300 rem beta, and 300 rem thyroid. The control room operator doses resulting from a LOCA were calculated to be below the 10 CFR 50 App. A, GDC 19 limits of 5 rem gamma, 30 rem beta, and 30 rem thyroid[13]. These data are shown on TABLE 7 below.

TVA's review confirmed that no significant environmental impact would occur for the LOCA Design Basis Accident. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well below the NRC's acceptance criteria.

[13] SQNAPS3-067 OFFSITE AND CONTROL ROOM OPERATOR DOSES DUE TO A MHA LOCA WITH MAXIMUM ALLOWABLE ANNULUS INLEAKAGE

TABLE 7 Radiological Consequences of a Design Basis Large Break LOCA (rem)

	Current Core	EIS Data Adjusted for 2,256 TPBARs	Revised TPC	NRC 10 CFR 100 Exposure Limits
2 Hour Exclusion Area Boundary				
Thyroid	145.1	*145*	139.7	300
Beta-Skin	4.522	*Not Reported Separately, included in Whole Body Value*	4.664	300
Whole Body (Gamma)	7.682	*12.2*	8.265	25
TEDE	9.875	*Not Analyzed*	10.23	
30-Day Low Population Zone				
Thyroid	27	*27.0*	25.96	300
Beta-Skin	1.341	*Not Reported Separately, included in Whole Body Value*	1.34	300
Whole Body (Gamma)	1.459	*2.9*	1.527	25
TEDE	1.923	*Not Analyzed*	2.084	
Control Room				NRC 10 CFR 50 Exposure Limits GDC 19
Thyroid	5.876	*Not Analyzed*	5.642	30
Beta-Skin	5.789	*Not Analyzed*	5.655	30
Whole Body (Gamma)	1.126	*Not Analyzed*	1.138	5
TEDE	1.330	*Not Analyzed*	2.186	

Main Steam Line Failure Outside of Containment

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the Reactor Coolant System to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage in each of the intact steam generators assumed to be equal to the Technical Specification limit of 150 gallons per day per steam generator. This analysis incorporates conservative assumptions associated with the alternate steam generator tube plugging criteria (3.7 gpm leakage in the faulted steam generator) and with both a pre-existing iodine spike and the situation in which the event triggers an iodine spike. In addition, failure of two TPBARs was assumed yielding an RCS Tritium level of about 98 μCi/gram.

It has been determined that the offsite doses due to a Main Steam Line Break with a pre-existing iodine spike are well within the 10 CFR 100 offsite dose guidelines. The offsite doses due to a Main Steam Line Break with an accident initiated iodine spike are a small fraction the 10 CFR 100 offsite dose guidelines (i.e. 30 rem thyroid and 2.5 rem whole body). The control room operator doses are less than the 10 CFR 50, Appendix A, GDC 19 limits (30 rem thyroid, 30 rem skin of the whole body, and 5 rem whole body)[14].

TVA's review confirmed that no significant environmental impact would occur for a potential Main Steam Line Failure Accident. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well below the NRC's acceptance criteria.

Steam Generator Tube Failure

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. A conservative analysis of the postulated steam generator tube rupture assumes the loss of offsite power and hence involves the release of steam from the secondary system. A conservative analysis of the potential offsite doses resulting from this accident is presented including an updated thermal and hydraulic analysis. This analysis incorporates conservatively updated assumptions associated with break flow flashing fractions and with both a pre-existing iodine spike and the situation in which the event triggers an iodine spike.

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[14] Westinghouse Electric Company, TVA-01-234, Radiological Consequences Of Accidents For The Sequoyah Nuclear Plant Units 1 And 2 Addressing Implementation Of Tritium Producing Burnable Absorber Rods (TPBARS).

In addition failure of two TPBARs was assumed yielding an RCS Tritium level of about 98 uCi/gram.[15].

TVA's review confirmed that no significant environmental impact would occur for a potential Steam Generator Tube Failure Accident. The calculated station offsite accident doses will be below the NRC's acceptance criteria.

Fuel Handling Accidents (FHA)

A fuel assembly is assumed to be dropped and damaged during refueling. Activity released from the damaged assembly is released to the outside atmosphere through either the containment purge system or the fuel-handling building ventilation system to the plant vent.

It is assumed that all of the fuel rods in the equivalent of one fuel assembly are damaged to the extent that all their gap activity is released. Also, the assembly inventory is based on the assumption that the subject fuel assembly has been operated at 1.7 times core average power. The damaged fuel assembly is assumed to be one with 24 TPBARs which are also assumed to be damaged. Although the release of tritium to the water pool is expected to take place relatively slowly, it is conservatively assumed that the tritium release occurs immediately.

The accident is defined as dropping of a spent fuel assembly resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations. The analysis considers an FHA occurring in containment with activity passing through the Purge Air Exhaust filters, and an FHA occurring in the fuel handling area of the Auxiliary Building with activity passing through the Auxiliary Building Gas Treatment System filters. The FHA is assumed to occur at 100 hours after shutdown. All the activity is assumed to be released over a two hour period per Safety Guide 25. For the TPC this analysis conservatively assumes that 24 TPBARs are located within the dropped spent fuel assembly and that they rupture and exchange their tritium with the water in the spent fuel pool. Data from Pacific Northwest National Laboratory[16] indicate that the total tritium activity released from 24 TPBARs into water of <200°F would not exceed 84,890 curies. This analysis assumes that the 84,890 curies of tritium are released to the environment over a two hour period.

An FHA occurring in containment results in the largest off site doses.

The 30 day Low Population Zone and 2-hr Exclusion Area Boundary /Site Boundary FHA offsite doses for the containment and fuel handling area of the auxiliary building were calculated to be

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[15] Ibid., Westinghouse Electric Company.

[16] TTQP-1-109 Rev 4. January 2001. *UNCLASSIFIED TPBAR RELEASES, INCLUDING TRITIUM.* Pacific Northwest National Laboratory, Richland, Washington.

well within the 10 CFR 100 limits of 25 rem gamma, 300 rem beta, and 300 rem thyroid[17]. These data are shown on TABLE 8 below.

This accident was described in the EIS as the TPBAR Handling Accident. It included the offsite dose from only the tritium component. The EIS analysis assumed that the entire tritium contents (231,360 curies) of the 24 TPBARs were released to the environment. For tritium only, the EIS calculated the maximally exposed offsite individual to receive an exposure of 0.028 rem and the average individual in the 50 mile Emergency Planning Zone a dose of 0.000014 rem.

TVA's review confirmed that no significant environmental impact would occur for a potential Fuel Handling Accident. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well below the NRC's acceptance criteria.

TABLE 8 Radiological Consequences of a Fuel Handling Accident (rem)

	Current Core	EIS Data	Revised TPC	NRC 10 CFR 100 Exposure Limits
2 Hour Exclusion Area Boundary (Containment)				
Thyroid	51.8	*Not Analyzed*	48.9	300
Beta-Skin	0.54	*Not Analyzed*	0.52	300
Whole Body (Gamma)	0.187	*Not Analyzed*	0.184	25
TEDE	1.77	*Not Analyzed*	2.03	
30-Day Low Population Zone (Containment)				
Thyroid	8.4	*Not Analyzed*	8.0	300
Beta-Skin	0.086	*Not Analyzed*	0.084	300
Whole Body (Gamma)	0.0303	*Not Analyzed*	0.0297	25
TEDE	0..29	*Not Analyzed*	0.33	

[17] Westinghouse Electric Company, TVA-01-234, Radiological Consequences Of Accidents For The Sequoyah Nuclear Plant Units 1 And 2 Addressing Implementation Of Tritium Producing Burnable Absorber Rods (TPBARS).

Rod Ejection Accident

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. This accident is bounded by the loss-of-coolant accident, discussed above.

TVA's review confirmed that no significant environmental impact would occur for a potential Rod Ejection Accident. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well below the NRC's acceptance criteria.

Failure of Small Lines Carrying Primary Coolant Outside Containment

The analysis of the environmental consequences included the offsite and control room operator dose due to Emergency Core Cooling System (ECCS) leakage outside containment following a LOCA. It has been determined that the offsite doses due to post LOCA ECCS leakage into the Auxiliary building are less than 10 CFR 100 limits of 25 rem gamma, 300 rem beta, 300 rem inhalation and 25 rem TEDE. The control room operator doses are less than the 10 CFR 50, App. A, GDC 19 limits of 5 rem gamma, 30 rem beta, 30 rem inhalation. It has been concluded that the ECCS leakage post LOCA will not contribute significantly to the total doses[18].

TVA's review confirmed that no significant environmental impact would occur due to Emergency Core Cooling System (ECCS) leakage outside containment following a LOCA Accident. Because the calculated station offsite accident doses are already well below the NRC acceptance criteria, any small increase in activity from the tritium and resultant environmental releases would have a negligible effect on the offsite accident doses, which would continue to remain well below the NRC's acceptance criteria.

[18] SQNSQS20210 OFFSITE AND CONTROL ROOM DOSE DUE TO AN ECCS LEAK OUTSIDE CONTAINMENT FOLLOWING A LOCA Revision 0