TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

MAY 2 1 2002

10 CFR 50.90

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

Gentlemen:

In the Matter of	)	Docket No.50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING TRITIUM PRODUCTION - INTERFACE ISSUE NUMBER 5 - CONTROL ROOM HABITABILITY SYSTEMS (TAC NO. MB1884)

The purpose of this letter is to provide information regarding NUREG1672, Interface Issue Number 5, "Control Room Habitability Systems," that was requested by NRC via letter dated May 8, 2002.

Initial information related to this interface issue was supplied by TVA on May 1, 2001, and with the license amendment request dated August 20, 2001. Enclosure 1 to this letter provide both the request and the TVA response. Enclosure 2 provides a summary of each of the accident analyses requested in NRC question 1.1.

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A003

There are no regulatory commitments made by this letter. If you have any questions about this letter, please contact me at (423) 365-1824.

Sincerely,

Rebecca M Mays
for P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures
cc: See page 3

Subscribed and sworn to before me
on this 21st day of May 2002

E. Jeannette Long
E. Jeannette Long
Notary Public

My Commission expires May 21, 2005

cc (Enclosures):

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ENCLOSURE 1
WATTS NUCLEAR PLANT (WBN)
INTERFACE ITEM NUMBER 5
CONTROL ROOM HABITABILITY SYSTEMS

NRC REQUEST

1. Section 2.15.6 of Enclosure 4, "Westinghouse Report NDP-00-0344," of Tennessee Valley Authority's (TVA's) August 20, 2001, amendment request addresses radiological consequences of various design basis accidents affected by the addition of tritium-producing burnable absorber rods to the Watts Bar Reactor. The U.S. Nuclear Regulatory Commission (NRC) staff needs additional information to make the requisite finding that consequences of the accidents are consistent with regulatory criteria. Please refer to Regulatory Information Summary 2001-019, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests," for a more complete discussion of the staff's expectations in regard to analysis descriptions. Please provide the information requested below or give us a specific reference if you have already docketed some of this information. For each accident analyzed:

   1.1 Provide a tabulation of all analysis inputs and assumptions used in offsite and control room habitability analyses in sufficient detail to enable the staff to evaluate the appropriateness of these data and, if deemed necessary, to perform confirmatory calculations.

   1.2 Describe any analysis methodology or modeling that is different from that previously approved by the NRC in a licensing action for WBN. Provide a justification for each change. This includes any changes in the determination of atmospheric dispersion values (χ/Q) for offsite or control room intake.

TVA RESPONSE:

Information related to Main Steam Line Break (MSLB) and Steam Generator Tube Rupture (SGTR) analyses have been previously provided as part of TVA submittals involving Outside Diameter Stress Corrosion Cracking (ODSCC), which was subsequently approved via NRC Safety Evaluation Report dated February 26, 2002, and the Iodine Spiking Technical Specification Change TS-01-12 submitted January 14, 2002. It is TVA's understanding that the iodine spiking change is currently under review and tracked by TAC Number MB3831. For ease of review, the input assumptions for both of these analyses have been provided herein.

It should also be noted that TVA License Amendment Request for Tritium Production was submitted on August 20, 2001, prior to issuance in October 2001 of Regulatory Information Summary 2001-019, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests."

For input assumptions regarding each of the analyses listed below, see Enclosure 2. Offsite/control room analyses inputs and assumptions for the Loss of AC Power, Waste Gas Decay Tank (WGDT) Failure, Loss of Coolant Accident (LOCA), MSLB, SGTR, Fuel Handling Accidents (FHA), Rod Ejection Accident (Consequences bounded by LOCA), and Failure of Small Lines Carrying Primary Coolant Outside Containment accidents have not changed for tritium production core (TPC) except for the following:

1) Initial Source Term: Core and single assembly sources were redeveloped taking into account TPC parameters. Based on specific accident, various cores were analyzed, including transition and equilibrium cores.

2) Because of the availability of updated analysis techniques TVA decided that the analysis of potential radiological consequences would be assessed using more up-to-date versions of the applicable models. ARCON96[1] was used for many of the control room χ/Q as described in Enclosure 2. The χ/Q values are based on the fact that ARCON96 provides more realistic meteorological factors which are used by various computer codes to define atmospheric pathways of radioactive effluents

[1] NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," May 1997.

and estimates of changes in concentrations as a result of atmospheric dispersion.

3) In addition to addressing the NRC regulatory requirements and dose guidelines, TVA has included calculated Total Effective Dose Equivalent (TEDE) in the accident analysis to appropriately account for the radiological consequences of the increased tritium in the Tritium Production Core. The TEDE values were calculated for informational purposes only and do not replace the whole body and thyroid dose guidelines currently in the WBN licensing basis.

4) TVA also used dose conversion values from Federal Guidance Report Number (FGR) 11[2] and 12[3]. The previous models used International Commission of Radiological Protection (ICRP)-2, dated 1970. ICRP-30 (FGR 11 and 12) provides more current dose conversion factors published by the International Commission of Radiological Protection and are used throughout the nuclear industry. Inhalation doses are now based on ICRP-30 dose conversion factors instead of ICRP-2.

5) 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 curies. 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 has been included in the MSLB, Loss of AC Power, SGTR, and WGDT accident analysis.

<u>NRC REQUEST</u>

1.3 Please provide a substantiated basis for the assumed 51 cfm control room unfiltered in-leakage. Justify your "positive pressure means no in-leakage" assumption in light of industry experience on this subject.

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[2] Federal Guidance Report No. 11, "Limiting Values Of Radionuclide Intake And Air Concentration And Dose Conversion Factors For Inhalation, Submersion, And Ingestion", EPA-520/1-88-020, U.S. EPA, Washington, DC.

[3] Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", EPA 402-R-93-081, U.S. EPA, Washington, DC., 1993.

TVA RESPONSE:

The irradiation of TPBARs does not impact analysis assumptions regarding control room in-leakage. The issue of Main Control Room Habitability and General Design Criteria (GDC) 19 compliance was addressed in Sections 1.5.5 and 2.15.6.3 of Enclosure 4 of the Watts Bar Nuclear Plant License Amendment Request (LAR). The data in the LAR table demonstrates that operation with a Tritium Production Core will result in compliance with GDC 19.

In accordance with GDC 19, the control room ventilation system and shielding have been designed to limit deep dose equivalent during an accident period to 5 rem. Thyroid dose is limited to 30 rem and beta skin dose should not exceed 30 rem. To appropriately account for the radiological consequences of the increased tritium in the Tritium Production Core, TVA has included calculated Total Effective Dose Equivalent (TEDE) in the accident analysis for information.

It should be noted that as tritium is unaffected (unfiltered) by the particulate and iodine filtration provided in the ventilation system, its concentration within the Control Room is therefore, only minimally impacted by changes in Control Room unfiltered in-leakage.

With respect to the issue of control room unfiltered in-leakage, TVA Watts Bar has previously provided information in a letter dated August 22, 2001. The following additional discussion is intended to supplement this information:

TVA performed a test in 1995 of identified vulnerabilities (i.e., vulnerabilities are potential unfiltered in-leakage paths, in this case the battery room exhaust duct) and measured the in-leakage at 1.8 CFM. TVA also conducted a review of the control room habitability zone (CRHZ) and determined there were no other vulnerabilities. Because the control room HVAC equipment is located within the CRHZ, it cannot contribute to unfiltered in-leakage. TVA also took into account the impact of pneumatically operated equipment and ingress/egress of personnel. The total unfiltered in-leakage is then documented in a TVA calculation (the document which includes the test data, pneumatic air leakage, and ingress/egress). This calculation closely parallels the COMPONENT TEST process defined in Nuclear Energy Institute (NEI) 99-03, *Control Room Habitability Assessment Guidance*, but did not follow NEI 99-03 exactly because the NEI document had not been issued at the time the unfiltered in-leakage at WBN was determined. The calculation establishes 51 CFM as the steady state unfiltered in-leakage for WBN. This 51 CFM includes 12.2 CFM of margin. TVA notes that NEI has previously provided NRC with information that supports the use of a component test as a valid method of determining unfiltered in-leakage. Note that TVA did not simply calculate unfiltered in-leakage, but used test data along with reviews of the CRHZ to document the unfiltered in-leakage into the most appropriate vehicle (i.e., a calculation) for plant records.

The issue involving the quantification of in-leakage is currently being assessed by the NRC with input from the nuclear industry. The issuance of draft Regulatory Guides and a draft Generic Letter along with NEI 99-03 are expected to address and resolve this issue. As such, this issue is independent of the irradiation of TPBARs and for Watts Bar Nuclear Plant is being handled by TAC No. MB3831. The open TAC Number MB3831 involving Technical Specification Change TS-01-12, "Iodine Spiking Technical Specification Change, submitted January 14, 2002, and or the Generic Letter, are more appropriate vehicles to resolve this issue.

<u>NRC REQUEST</u>

1.4 **The NRC's letter of April 25, 2002, contained a second RAI on Interface Issue 7, questioning the ability of the TPBAR consolidation canister load-handling equipment to meet single failure criteria. If TVA does not address this under Interface Issue 7, please provide an analysis of the offsite and control room doses resulting from a dropped consolidation canister 300 TPBARs to demonstrate that the event is, in fact, bounded by the drop of a single assembly containing 24 TPBARs.**

<u>TVA RESPONSE:</u>

The TPBAR container movement equipment is adequately designed to preclude damage to consolidation canister contents and the analysis of doses resulting from damage to 300 TPBARs is not required. This issue is discussed in detail in TVA's response to the NRC RAI on Interface Issue 7, "Light Load Handling System" dated May 21, 2002.

NRC REQUEST

2. Section 1.5.5 of Enclosure 4 to TVA's letter of August 20, 2001, discusses control room habitability. The discussion appears to be limited to the emergency core cooling system leakage component of the loss of coolant (LOCA). You refer to Table 2.15.6-2 as the basis for TVA's conclusion that General Design Criterion 19 (GDC-19) will continue to be met. We note that the language of GDC-19 is not restricted to LOCAs, but applies to all accidents. Please explain how TVA's conclusion that meeting GDC-19 addresses all of the design-basis radiological accidents considered in the Watts Bar Updated Final Safety Analysis Report. If TVA based this conclusion on the LOCA being the limiting accident, please justify this conclusion addressing the following:

   2.1 The impact of accident-specific differences in release point configuration (e.g., upwind direction and distance, release point height, diffuse or point source, etc.)

   2.2 The impact of accident-specific differences in the activation of control room protective features inherent delays associated with these differences (e.g., instantaneous safety injection signal versus radiation monitor alarm.)

   2.3 The impact of accident-specific differences in source terms on monitor response and isolation delay in reaching set point if actuation is based on radiation monitor response.

   2.4 Mode dependent engineered safety feature operability for the fuel-handling accident versus at-power accidents, including the impact this may have on control room unfiltered in-leakage.

TVA RESPONSE:

Section 1.5 of Enclosure 3 of the Watts Bar License Amendment Request was developed to address each of the NRC Interface Items as identified in NUREG-1672, "Safety Evaluation Report Related to the Department of Energy's Topical Report on the Tritium Production Core." Section 1.5.5 addresses Interface Item 5, on the Control Room Habitability Systems. NUREG-1672, Section 2.6.1 discussed the impact of Emergency Core Cooling System (ECCS) leakage on Main Control Room Habitability and required a plant specific assessment of ECCS impacts. The discussion in LAR Section 1.5.5 is limited to discussion of this subject and the associated accident (LOCA) and the impact on GDC-19 criteria for Main Control Room Habitability.

TVA concurs that GDC-19 applies to the spectrum of potential accident scenarios in the WBN Updated Final Safety Analysis Report (UFSAR). These scenarios are individually addressed in section 2.15.6 of Enclosure 4 of the WBN Licensed Amendment Request dated August 20, 2001. It is TVA's determination that a failure of small lines carrying primary coolant outside containment results in the greatest dose for beta and gamma, the MSLB is limiting for thyroid, and a spent fuel pool FHA is limiting for TEDE. Therefore, TVA meets the requirements of GDC-19 for all postulated events.

<u>NRC REQUEST</u>

3. **TVA analyzed the total effective dose equivalent as well as the whole body and thyroid doses to determine radiological consequence from TPBARs being in the reactor. TVA's submittal of August 20, 2001, does not appear to request that the total effective dose equivalent (TEDE) dose quantity and its associated dose criteria will replace the whole body and thyroid dose guidelines currently in the Watts Bar licensing basis. Please confirm our understanding that future design basis accident radiological analyses, intended to demonstrate compliance with regulatory criteria, will continue to assess whole body and thyroid doses, as well as TEDE, for tritium.**

<u>TVA RESPONSE:</u>

In addition to addressing the NRC regulatory requirements and dose guidelines, TVA has included calculated TEDE in the accident analysis to appropriately account for the radiological consequences of the increased tritium in the Tritium Production Core. The TEDE values were calculated for informational purposes only and do not replace the whole body and thyroid dose guidelines currently in the WBN licensing basis. Future design basis accident radiological analyses, which are intended to demonstrate compliance with regulatory criteria, will continue to assess whole body and thyroid doses and will contain informational data regarding TEDE.

ENCLOSURE 2
WATTS NUCLEAR PLANT (WBN) UNIT 1
INTERFACE ITEM NUMBER 5
INPUT ASSUMPTION SUMMARIES FOR SELECTED ANALYSES

LOSS OF AC POWER

A calculation has been performed by TVA to determine offsite doses and the control room operator doses due to a Loss of AC Power for WBN Unit 1 operation with a Tritium Production Core (TPC). The calculation determined that a fuel handling accident would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, General Design Criteria (GDC)-19 limit.

The calculation used TVA computer codes Source Transport Program (STP) Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which is used to determine 30-day Low Population Zone (LPZ) and 2-hour Exclusion Area Boundary (EAB) offsite. The calculation incorporates new atmospheric dispersion coefficients (χ/Q) values as determined by ARCON96.

Design Input:

1. The calculation was performed for the realistic case and 1 percent (%) failed fuel case. The 1% failed fuel case is a conservative analysis that utilizes a factor of 8 as a multiplier to the realistic case. This factor causes the inventories to exceed the technical specification values (which would have a multiplier of 7.965). The 1% failed fuel is based on the realistic case being 0.125% failed fuel, therefore 8 times 0.125% = 1 %.

2. Models are taken from Main Steam Line Break (MSLB) calculation, except for the source terms and χ/Q's. The amount of steam released to the environment due to the loss of AC power is provided below:
   0 - 2 hours (hrs) 625,000 pounds (lbs.)
   2 - 8 hrs 959,000 lbs.

Assumptions:

1. The secondary side source consists of expected/realistic radionuclide activity levels for a reactor based on ANSI/ANS

18.1-1984. The 2 TPBAR failure source term is used for each case.

2. The following are the ARCON96 χ/Q values used in the computer code models:

Offsite : 30 day LPZ:	Control Room:
0-2 hr - 1.41E-4 2-8 hr - 6.68E-5 8-24 hr - 4.59E-5 1-4 days - 2.04E-5 2 hr EAB - 6.07E-4	0-2 hr - 4.03E-3 2-8 hr - 3.35E-3 8-24 hr - 2.27E-4 1-4 day - 1.81E-4 4-30 day - 1.45E-4

WASTE GAS DECAY TANK (WGDT)

A calculation was performed by TVA to determine offsite doses and the control room operator doses due to a rupture of a waste gas decay tank for WBN Unit 1 operation with a TPC. The calculation determined that a waste gas decay tank rupture would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, GDC-19 limit.

The calculation used TVA computer codes STP Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which is used to determine 30-day LPZ and 2-hour EAB offsite.

Assumptions:

1. The tank is assumed to be filled with the highest concentration for each isotope from all sources into the WGDT. This will ensure maximum concentration of all isotopes. Only the 2 TPBAR failure case is run, as the tritium has only a small impact on the results and using the 2 TPBAR failure source term is conservative. Ten percent (%) of the tritium is released as gas, thus the tritium source terms are:

TPC - Normal Operations:	906.9 Ci (total)* 10%	= 90.69 Ci
1 TPBAR Failure:	12506.9 Ci*10%	= 1250.69 Ci
2 TPBAR Failure:	24106.9 Ci*10%	= 2410.69 Ci

2. Radioactive decay is only taken into account for the time period required to transfer the gasses to the tank except for tritium. The maximum content of the failed decay tank is assumed to be released non-mechanistically to the environment over a two hour time period (Regulatory Guide (RG)-1.24). For tritium, due to its 12.3 year half life, it is considered that no decay occurs.

3. The tank failure is assumed to occur immediately upon completion of the waste gas transfer (RG 1.24).

4. Only one tank is assumed to fail, as all decay tanks are isolated from each other whenever the tanks are in use.

5. The release path of the radioisotopes from the ruptured tank is through the Auxiliary Building.

6. All Control Room dispersion coefficients (χ/Q) are based on three adjacent 22.5 degree sectors centered on the appropriate effluent-intake vents and the methodology of Halitsky et.al. The sectors are assumed to be approximately the same as LOCA Cases 1 (Unit 1 to Vent 1) and 4 (Unit 2 to Vent 2). Since the distances from the effluent vent (Auxiliary Building Vent) to intake vents are different for this problem, the χ/Q values from LOCA calculations cannot be used. The worst case control room emergency air intake vent is used. Halitsky curves assume that the releases come from the Reactor Building. As the distance from the release point increases, the building wake effects decrease. Therefore, assuming that the WGDT release from the Auxiliary Building vents is at the Reactor Building, which is required in order to use the Halitsky curves, and using the distance from the Auxiliary Building vent to the Control Building intake vent will lead to a conservative result because this artificially increases the building wake effects. There are 2 cases of χ/Q values for the control room with a release from the Auxiliary Building exhaust vent. The first is the Auxiliary Building vent to Control Room intake Vent 1. The second is the Auxiliary Building vent to Control Room intake Vent 2.

Vent	0-2 hr	2-8 hr	24 hr	1-4 days	4-30 days
1	6.77E-03	3.58E-03	2.96E-03	2.09E-03	1.57E-03
2	3.59E-03	2.33E-03	1.99E-03	1.42E-03	9.80E-04

LOSS OF COOLANT ACCIDENT

A calculation was performed by TVA to determine offsite doses and the control room operator doses due to a Regulatory Guide 1.4 Loss of Coolant Accident (LOCA) for WBN Unit 1 operation with a TPC. The calculation determined that a LOCA would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, GDC-19 limit.

The calculation used TVA computer codes STP Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which is used to determine 30-day LPZ and 2-hour EAB offsite). The calculation incorporates new χ/Q values as determined by ARCON96. The TPC is included in the analysis. Two TPC cases are determined: one with 100% of the tritium released to the containment atmosphere, and the other with only 3% released. This calculation determined the doses using 3 different methodologies. The gamma, beta and Thyroid International Commission of Radiological Protection (ICRP)-2 doses are all based on TID-14844 methodologies utilizing the ICRP-2 iodine dose conversion factors found in TID-14844 and are the current licensing basis of the plant. The second methodology is the Thyroid (ICRP-30) dose, which is also based on TID-14844, but uses the ICRP-30 iodine dose conversion factors. The ICRP-30 iodine dose conversion factors are less conservative than the ICRP-2 factors. Finally, the third methodology used is the TEDE (Total Effective Dose Equivalent). The TEDE presents an overall weighted dose and is more representative of the impact of all isotopes on the body as a whole. The TEDE dose is not currently part of the design basis of the plant. It is important to note that tritium does not impact the thyroid doses utilizing the TID-14844 methodology, because only iodine is applied to the thyroid dose. However, in fact tritium does contribute to the thyroid dose, as well as other organs of the body. This is why the TEDE is a more representative dose when discussing the impact of tritium.

LOCA Offsite

Design Input:

1. The primary containment free volume available for mixing is: 1.27E6 cubic feet (ft^3).

2. The annulus free volume: (50% of which is assumed 3.75E5 ft^3 available for mixing)

3. The Auxiliary Building volume available for mixing is assumed to be:1.62E5 ft^3 (this is based on a conservative "effective volume" with an assumed mean holdup time of 0.3 hours)

4. The primary containment air return fan flow rate (1 of 2 assumed operating): 40,000 cubic feet per minute (cfm)

5. Total Emergency Gas Treatment System (EGTS) flow rate: 3,600 cfm

6. Auxiliary Building Gas Treatment System (ABGTS) flow rate: 9,000 cfm

7. Primary containment leak rate: 0.25%/day 0-24 hrs, 0.125%/day >24 hrs

8. Annulus In-leakage (from Auxiliary Building): 250 cfm

9. Primary containment leakage to annulus: 75%

10. Primary containment leakage to the Auxiliary Building: 25%

11. EGTS filter efficiencies: 99%

12. ABGTS filter efficiencies: 99%

13. Delay before credit is taken for ABGTS filters: 4 minutes.

14. Radioisotopes available for release: 25% core iodines, 100% noble gases

15. Releases between time=0 to times up to 66 seconds post LOCA from the various systems breaching containment are not taken into account. It is assumed that the consequences of any releases from the systems during the time when the system containment isolation valves are open is insignificant. The purge mass releases bound any potential release through the other systems being isolated, since many of those systems are closed/self contained and do not have a credible leakage path.

16. Direct leakage from the Auxiliary Building to the environment occurs from 30 to 34 minutes post LOCA due to single failure of the ABGTS control system. The leakage is assumed to be 9900 cfm, which is the same as the in-leakage

when there is a driving force (the differential pressure). This value is chosen because, as the differential pressure decreases, the driving force also decreases. The 9900 cfm is maximum leakage during maximum delta pressure (9000 cfm+10%); therefore, it is expected that during essential zero delta pressure, the out-leakage will not exceed 9900 cfm. This calculation establishes 9900 cfm as the maximum permissible leakage.

17. For TPC configurations, two cases are analyzed. Case A assumes 100% of the core tritium release to the containment atmosphere. Case B assumes 3% of the core tritium is released.

18. It is assumed that tritium is not filtered out by the EGTS or ABGTS system.

19. The leakage into the entire EGTS system from the Auxiliary Building is 10.7 cfm or 642 cubic feet per hour (cfh).

20. The leakage into the ABGTS ductwork downstream (and therefore bypassing) the filters and upstream of the fans is 88 cfm or 1672.8 cfh.

21. Ice Condenser Removal Efficiencies for elemental and particulate iodine are per Regulatory Guide 1.4.

22. The iodine is assumed to be 91% elemental, 5% particulate, and 4% organic (Regulatory Guide 1.4)

23. χ/Q's

30 day LPZ :	2 hr EAB
0-2hr - 1.41E-4 2-8hr - 6.68E-5 8-24 hr - 4.59E-5 1-4 days - 2.04E-5	6.07E-4

<u>LOCA Control Room</u>

1. All assumptions from LOCA Offsite hold.

2. All dispersion coefficients (χ/Q) are determined by ARCON96.

3. The worst case control room emergency air intake vent is used for the initial 8 hours and the best case vent for the

remaining duration of the accident. Following a Regulatory Guide 1.4 LOCA the Main Control Room Habitability System (MCRHS) area will be automatically isolated upon the actuation of a safety injection signal or indication of high radiation in the outside air supply stream to the building. Upon isolation both Control Building emergency air cleanup units and both emergency pressurizing air supply fans will operate. Subsequently, one of each is placed in standby mode by the operator. The operator has the capability to compare the concentration of radioisotopes at the emergency air intakes, thus the operators have adequate guidance for choosing the better intake vent.

4. Each of the Control Building air cleanup units has a total recirculation flow rate of 3600 cfm (4000-10%) - (2889 cfm recirculated air and 711 cfm pressurization air). The filter train of each unit has a bank of HEPA filters followed by charcoal filters. This analysis assumes 95% removal efficiency for all species of iodine for the first pass, and 70% removal efficiency is assumed for all species of iodine for the second pass (Regulatory Guide 1.140).

5. Immediately following an accident, it is assumed that the operator on duty at the time will stay in the MCRHS. This is indicated by the use of a conservative 100% occupancy factor for the first 24 hours. After this period, the operators should change shifts in the routine manner. Therefore, it is assumed that one one-way trip between the Control Room and offsite will occur during the 8-24 hour time period. This will be followed by one round trip per day for the duration of the accident. The occupancy factor for 1-4 days is 60%, and for 4-30 days is 40%.

6. The assumptions for the operator travel to and from the site are:
   a) 5 minutes is required for an operator to leave the MCRHS area and arrive at his car.
   b) The distance between the activity release point and the employee parking lot is at least 125 meters. The employee parking lot is assumed to be the worst in relation to the two Shield Buildings.
   c) The distance from the release point to the site boundary (exclusion boundary) is about 1,100 meters.
   d) The distance traveled on access road to the site boundary is conservatively assumed to be about 1,500 meters.
   e) The average speed of the operator's car is assumed to be 25 miles per hour (mph).

7. It is assumed that the makeup flow into the Control Room Habitability Zone is 711 cfm in order to maintain positive pressure.

8. The χ/Q values determined in this calculation are based on centerline values for a plume from the Shield Building vent to the Control Room intake vent. Scenarios have been postulated that have both intake vents and associated filter trains operational (thereby doubling the flow into the control room). The χ/Q off centerline adjustment is proportional to $\exp[-y^2/2(\text{sigma}_y)^2]$ where sigma_y is the horizontal dispersion coefficient [m] and y is the lateral distance from the plume centerline. For the case of WBN, the distances from the exhaust to the intake structures range from 200 to 300 feet (60 - 90 meters). This will lead to a sigma value of about 4 (Pasquill stability factor F). A plume bisecting the intake structures will be at least 60'+ (20 meters) laterally from each intake. The dilution adjustment for horizontal dispersion would therefore be $\exp[-400/(2)(16)] = 4E-6$. Therefore, it can be concluded that a second intake would lead to insignificant radioisotope inflows in the event of two operational intake vents. Therefore, scenarios which would involve off centerline χ/Q values for one or both vents are not included.

9. The unfiltered in-leakage is 51 cfm.

10. The χ/Q's at the parking lot are assumed to be that at the west Control Room intake vent for the respective time periods.

    8-24 hrs - 4.59E-05 seconds per cubic meter (sec/m^3);
    24-96 hrs - 2.04E-05 sec/m^3;
    96-720 hrs - 6.35E-06 sec/m^3

11. To compensate for the fact that the access road is not a straight line from the parking lot to the site boundary, which is what the COROD program assumes, the assumed speed of 25 mph is weighed with the ratio of the distance traveled. The leads to the adjusted speed of 25 mph*[(1100m-125m)/1500m]= 16.25 mph.

12. Since the Unit 1 Shield Building χ/Q values are higher than the Unit 2 Shield Building χ/Qs for all time periods, only

Unit 1 doses are calculated. Per assumption 4, the worst χ/Q values are used in the 0-2 hour and 2-8 hour time periods. The more favorable χ/Q values (for the same unit) are used for time periods after 8 hours. The χ/Q values are:

0-2 hours	-	1.12E-3 sec/m^3;
2-8 hours	-	9.78E-4 sec/m^3;
8-24 hours	-	1.21E-4 sec/m^3;
1-4 days	-	9.36E-5 sec/m^3;
4-30 days	-	7.77E-5 sec/m^3

MAIN STEAM LINE BREAK

This calculation was performed to determine the acceptable permissible steam generator primary-to-secondary leak rate during a steam line break and to show that the offsite and control room operator doses do not exceed the 10 CFR 100 and 10 CFR 50 Appendix A GDC 19 dose limits while operating with a Tritium Production Core. The calculation determined that 1 gallon per minute (gpm) (at standard temperature and pressure) primary-to-secondary leakage in the faulted steam generator would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, GDC-19 limit.

The calculation used TVA computer codes STP Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD, (which determines control room operator dose) and FENCDOSE, (which is used to determine the 30-day LPZ and the 2 hour EAB dose). The calculation incorporates new χ/Q values as determined by ARCON96.

Two methods of determining the resultant dose for the main steam line break were used in accordance with the Standard Review Plan Section 15.6.3 methodology:

1. A pre-accident iodine spike where the iodine level in the reactor coolant spiked upward to the maximum allowable limit of 21 micro curies per gram (µCi/gm) I-131 equivalent (value currently administratively controlled at WBN) just prior to the initiation of the accident.

2. The reactor coolant at the maximum steady state dose equivalent I-131 of 0.265 µCi/gm with an accident initiated iodine spike consisting of a 500 time increase on the rate of iodine release from the fuel.

In both cases, the primary-to-secondary side leak is 150 gallons per day (gpd) per steam generator in the unfaulted loops, and the secondary side activity is at the WBN Technical Specification Limiting Condition for Operation (LCO) 3.7.14 limit of 0.1 µCi/gm dose equivalent Iodine 131 (I-131).

Assumptions for the Postulated MSLB accident:

1. RCS Letdown flow of 124.39 gpm (120 gpm + 4.39 gpm uncertainty) is used.

2. RCS Letdown demineralizer efficiency is assumed to be 1.0 for iodines.

3. ANSI/ANS-18.1-1984 spectrum was used and was scaled up to 0.265 or 21 µCi/gm equivalent iodine.

4. The tritium inventory in the TPC assumes 2 TPBAR failures (98.4 µCi/gm in the reactor coolant).

5. Two cases were used. In the first case, a pre-Accident iodine spike of 21 µCi/gm dose equivalent I-131 in the RCS was used. The second case uses an accident initiated iodine spike which increases the iodine release rate to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium into the reactor coolant from the fuel rods.

7. Primary side to secondary side leakage of 150 gpd standard temperature and pressure per steam generator in the intact loops.

8. It is assumed that the RCS leakage prior to the accident is 11 gpm at standard temperature and pressure.

9. Steam released to the atmosphere
   a) total from the non-defective steam generators (0 - 2 hr), 480,000 lb
   b) total from the non-defective steam generators (2 - 8 hr), 871,000 lb
   c) total from the faulted steam generator (0 - 30 minutes [min]), 150,000 lb steam generator secondary inventory (standard temperature and pressure primary)
   d) total from the faulted steam generator (30 mins - 8 hrs) 1000 lb (standard temperature and pressure) primary-to-secondary leakage

10. Iodine partition coefficients for steaming of steam generator water:
   a) non-defective steam generators (initial inventory) and primary-to-secondary leakage, 0.01.
   b) faulted steam generator (initial inventory and primary-to-secondary leakage), 1.0.

11. Atmospheric dilution factors (χ/Q), are as follows for LPZ seconds per cubic meter (sec/m^3).
   a) 0-2 hr - 1.41E-4
   b) 2-8 hr - 6.68E-5
   c) 8-24 hr - 4.59E-5
   d) 1-4 days - 2.04E-5
   e) 4-30 days - 6.35E-6

12. Atmospheric dilution factor, χ/Q, for 2-hr EAB is 6.07E-4 sec/m^3.

13. Atmospheric dilution factors, χ/Q, are as follows for the control room (sec/m^3).
   a) 0 - 2 hr - 4.03E-3
   b) 2 - 8 hr - 3.35E-3

STEAM GENERATOR TUBE RUPTURE

A calculation was performed by TVA to determine the radiological consequences of a Steam Generator Tube Rupture (SGTR) for WBN Unit 1 operation with a TPC. The calculation determined that a SGTR would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, GDC-19 limit.

The calculation used TVA computer codes STP Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD, (which determines control room operator dose) and FENCDOSE, (which is used to determine the 30-day LPZ and the 2 hour EAB dose).

Assumptions for the Postulated SGTR accident:

1. RCS Letdown maximum flow of 124.39 gpm is used.

2. RCS Letdown demineralizer efficiency is assumed to be 1.0 for iodines.

3. The tritium inventory in the TPC assumes 2 TPBAR failures (98.4 μCi/gm in the reactor coolant).

4. ANSI/ANS-18.1-1984 spectrum was used and was scaled up to 0.265 or 21 μCi/gm equivalent iodine.

5. Two cases were used. In the first case, a pre-Accident iodine spike of 21 μCi/gm I-131 equivalent in the RCS was used. In the second case, an accident initiated iodine spike increases the iodine release rate to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium into the reactor coolant from the fuel rods.

6. Primary side to secondary side leakage of 150 gpd standard temperature and pressure per steam generator in the intact loops.

7. Steam released to the atmosphere
   a) total from the non-defective (secondary side) steam generators (0 - 2 hr), 517,500 lbs.
   b) total from the non-defective (secondary side) steam generators (2 hr - 8 hr), 923,500 lbs.
   c) total from the faulted steam generator (secondary side) (0 - 2 hr), 103,500 lbs.
   d) total from the faulted (secondary side) steam generator, 33,100 lbs (standard temperature and pressure).
   e) total RCS from the faulted steam generator, 176,700 lbs of which 9474.5 lbs flashed.

8. Iodine partition coefficients for steaming of steam generator water:
   a) non-defective steam generators (initial inventory) and primary-to-secondary leakage, 0.01
   b) faulted steam generator (initial inventory and primary-to-secondary leakage), 0.01
   c) faulted steam generator (RCS flashed) 1.0

9. Atmospheric dilution factors, (χ/Q), are as follows for LPZ seconds per cubic meter (sec/m^3):

a)	0-2 hrs	-	1.41E-4
b)	2-8 hrs	-	6.68E-5
c)	8-24 hrs	-	4.59E-5
d)	1-4 days	-	2.04E-5
e)	4-30 days	-	6.35E-6

10. Atmospheric dilution factors, χ/Q, for 2 hr EAB is 6.07E-4 sec/m^3.

11. Atmosphere dilution factors, χ/Q, are as follows for control room (sec/m^3):

    a) 0-2 hrs - 4.03E-3
    b) 2-8 hrs - 3.35E-3

12. Main Control Room related assumptions:

    a) Volume - 257,198 cubic feet
    b) Makeup/pressurization flow - 711 cubic feet per meter (cfm)
    c) Recirculation flow - 3600 cfm
    d) Unfiltered intake - 51 cfm
    e) Filter efficiency - 95 % first pass, 70% second pass, 0% for noble gases
    f) ICRP-30 dose conversion factors
    g) Control Room Isolation time - 20.6 seconds
    h) Occupancy factors:
        1) 0-24 hrs - 100%
        2) 1-4 days - 60%
        3) 4-30 days - 40%

13. Steady-state RCS leak rate of 11 gpm prior to the accident is assumed.

14. The noble gas inventories are maximized by scaling them to the Technical Specification limit of 100/Ebar.

FUEL HANDLING ACCIDENT (FHA)

A calculation was performed by TVA to determine the environmental consequences of a Fuel Handling Accident for WBN Unit utilization of a TPC. The calculation determined that a fuel handling accident would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, GDC-19 limit.

The calculation used TVA computer codes STP Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which is used to determine 30-day LPZ and 2-hour EAB offsite doses). The calculation incorporates new χ/Q values as determined by ARCON96.

Assumptions for the Postulated Fuel Handling Accident:

1. The FHA occurs at 100 hours after shutdown, consistent with the FSAR and the Technical Specifications.

2. All of the fuel rods in one fuel assembly is assumed to be damaged.

3. All activity is assumed to be released to the environment over a two hour time period.

4. All of the gap activity in the damaged rods is released which consists of 10% of the inventory in the rods at the time of the accident, except for the following (per NUREG/CR-5009 for 60 gigawatt days per ton (GWd/t). Note that for lesser burnups the releases are less, therefore use of these 60 GWd/t values for all burnups is conservative):

   Kr-85 = 14%
   Kr-87 = 10%

   Note: The NUREG/CR-5009 value is actually 0.7%. Since STP is limited to 9 classes and the half life of Kr-87 is 76 min, after 100 hours of decay there will be exp(-100*ln(2)/(76/60)) = 1.7E-24 or 1.7E-22% left. Therefore the increase in the gap percentage does not affect the results.

   Kr-88 = 10% Note: The NUREG/CR-5009 value is actually 1%. Since STP is limited to 9 classes and the half life of Kr-88 is 2.84 hr, after 100 hours of decay there will be exp(-100*ln(2)/2.84) = 2.5E-11 or 2.5E-9% left. Therefore, the increase in the gap percentage does not affect the results.

   Kr-89 = 10%
   Xe-133 = 5%
   Xe-135 = 2%
   I-131 = 12%

6. The values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown with a radial peaking factor of 1.65 for the standard core assembly.

7. The iodine gap inventory is composed of inorganic species (99.75%) and organic species(0.25%).

8. The pool decontamination factors for the inorganic iodine is assumed to be 133, and organic iodine is assumed to be 1.

9. The retention of noble gasses in the pool is negligible.

10. For the FHA in containment it is assumed that the Purge Air Exhaust (PAE) System operates for the duration of the accident. It is assumed that the entire inventory is released through the PAE System filters.

11. It is assumed that everything is released to the environment within 2 hours. To assure this, at 2 hours all remaining isotopes in containment (or above the spent fuel pool) are transferred into the environment (using the appropriate filter efficiency as a multiplication factor).

12. The filter efficiencies for the PAE filter are 90% for inorganic iodines and 30% for organic iodines.

13. The filter efficiency for the ABGTS is 99% for all iodines.

14. The effective volume of upper containment is taken as 1/2 the upper containment free volume.
    Containment air volume = 647,000 $ft^3 \div 2$ = 3.235 E+05 ft^3

15. It is assumed that the suction flow for the ABGTS from the spent fuel pit area is the maximum ABGTS flow (9000 cfm +10%).

16. NUREG/CR-5009 implies that Cs-134 and Cs-137 are also in the gap. This calculation assumes these isotopes are not released to the environs.

17. It is assumed that all 24 TPBARs in a fuel assembly with either once or twice burned fuel will break. If all the water were to evaporate, then the amount of tritium release would be: 60μCi/gm * 372,000 gal * 3,785.4 cc/gal * 1 gm/cc * 1E-6 Ci/μCi = 84490 Ci. It is assumed that all the tritium (84490 Ci) in the spent fuel pool is released following the FHA through evaporation of the pool.

18. Purge flow rate = 14,954 cfm.

- Main Control Room related assumptions:
  - a) Volume - 257,198 cu ft
  - b) Makeup/pressurization flow - 711 cfm
  - c) Recirculation flow - 3600 cfm
  - d) Unfiltered intake - 51 cfm
  - e) Filter efficiency - 95% first pass, 70% second pass, 0% for noble gases
  - f) ICRP-30 dose conversion factors
  - g) Control Room Isolation time = 20.6 seconds
  - h) Occupancy factors:
    - i) 0-24 hrs - 100%
    - ii) 1-4 days - 60%
    - iii) 4-30 days - 40%

- χ/Q values for the LPZ and EAB are the same as in the SGTR accident.

- Control Room χ/Q values (sec/m^3):

0-2 hr	-	1.12E-3
2-8 hr	-	9.78E-4
8-24 hr	-	1.21E-4
24-96 hr	-	9.36E-5
96-720 hr	-	7.77E-5

ROD EJECTION ACCIDENTS

A review of this analysis was performed and the analysis remains unchanged by the insertion of TPBARs into the reactor core. The consequences of a postulated rod ejection accident remain bounded by the results of the LOCA accident analysis.

FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT

A calculation was performed by TVA to determine offsite doses and the control room operator doses due to ECCS leakage outside containment following a LOCA for WBN Unit 1 operation with a TPC. The calculation determined that the postulated accident would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, GDC-19 limit.

The calculation used TVA computer codes STP Revision 6, FENCDOSE Revision 4, and COROD Revision 5. The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which is used to determine 30-day LPZ and 2-hour EAB offsite. The calculation incorporates new χ/Q values as determined by ARCON96 and determines thyroid doses based on ICRP-30 dose conversion factors.

Assumptions:

1. It is assumed that only 3760 cc/hr of the ECCS leaks into the Auxiliary Building. The leak begins at the 10 minutes post LOCA, which is the shortest time at which recirculation begins, and lasts the duration of the accident.

2. Noble gasses will normally not remain in solution; however, all of the noble gasses entering the Auxiliary Building from the leak are assumed to become airborne instantaneously and mixed homogeneously. The volume of the Auxiliary Building is arbitrarily set so the holdup is 0.3 hours and accounts for dead end spaces and incomplete mixing in the building.

3. Ten percent of the iodine in the leak will become airborne (partition factor of 10 as suggested by NUREG-0800) and 90% will remain in solution. Since the recirculating water in the ECCS is less than 212 degrees Fahrenheit (based on analyzed post LOCA sump water temperatures), no flashing or phase change will occur at the leak location.

4. One hundred percent of the tritium entering the Auxiliary Building in the ECCS water is assumed to become airborne.

5. Containment Sump volume is 9.63E4 ft^3.

6. Auxiliary Building Sump volume is 1 ft^3 (Arbitrarily small)

7. The step source into the sump consists of 50% of the core iodines per Regulatory Guide 1.89, NUREG-0588, and NUREG-0800.

8. Noble gas daughters flow at and arbitrarily large value of 1E6 ft^3/hr from the containment sump to the containment atmosphere and from the Auxiliary Building Sump to the Auxiliary Building.

9. The leak outside of containment is modeled by a flow from the sump component to the Auxiliary Building at a rate of (3760 cc/hr)(1 ft^3/28317 cc) = 0.13278 ft^3/hr. To model the iodine becoming airborne, the above flow rate is split into

a flow into the Auxiliary Building Sump component and into the Auxiliary Building component. The flows are as follows:

Flow from component 11 (Containment Sump) to 12 (Auxiliary Building Sump) = (0.13278)(0.9) = 0.119504 ft^3/hr

Flow from component 11 (Containment Sump) to 3 (Auxiliary Building) = (0.13278)(0.1) = 0.013278 ft^3/hr

10. For the TPC with 97% tritium in the sump (3% airborne), the leak outside of containment is modeled by a flow from the sump component to the Auxiliary Building at a rate of (3760 cc/hr)(1 ft^3/28317 cc) = 0.13278 ft^3/hr starting at 10 minutes and lasting the duration of the accident.

11. The 2 hour EAB (Site Boundary) TEDE Dose is determined by multiplying the 2 hour LPZ TEDE dose by 4.305.

12. The ARCON96 control room χ/Q's are:

0-2hr	-	4.03E-3
2-8hr	-	3.35E-3
8-24hr	-	2.27E-4
1-4 day	-	1.81E-4
4-30 days	-	1.45E-4