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

JAN 14 2002

10 CFR 50.90

TVA-WBN-TS-01-12

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

Gentlemen:

In the Matter of)
Tennessee Valley Authority)

Docket No.50-390

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE NO. WBN-TS-01-12 - REACTOR COOLANT SYSTEM (RCS) SPECIFIC ACTIVITY

In accordance with the provisions of 10 CFR 50.4 and 50.90, TVA is submitting a request for an amendment to WBN's License NPF-90 to change the TSs for Unit 1. The proposed change lowers the steady state specific activity of the primary coolant from 1.0 microcurries per gram (µCi/gm) Dose Equivalent Iodine-131 (DEI-131) to 0.265 µCi/gm DEI-131, and the 48 hour maximum iodine spike value of 60 µCi/gm DEI-131 to 21 µCi/gm DEI-131. In addition, TVA proposes to change the allowable value for Main Control Room air intake radiation monitor from ≤ 5.77E-04 µCi/cubic centimeters (cc) (20,199 counts per minute [cpm]) to ≤ 9.45E-05 µCi/cc (3,308 cpm).

The above proposed changes are required as a result of TVA's reanalyses of the dose calculations for the Main Steam Line Break and the Steam Generator Tube Rupture accidents. These accident analyses were revised due to comments received from NRC during the review of TVA's license amendment request for a steam generator alternate repair criteria, WBN-TS-99-014. The results of the reanalyses are being addressed within TVA's Corrective Action Program and are being administratively controlled in accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety."

D030

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The WBN Plant Operations Review Committee and the TVA Nuclear Safety Review Board have reviewed this proposed change and determined that operation of WBN Unit 1 in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate TS pages from Unit 1 marked-up to show the proposed change. Enclosure 3 forwards the revised TS pages for Unit 1 which incorporate the proposed change.

As previously discussed, administrative controls are in place for this TS change. Approval of this License Amendment Request is dependent upon NRC's approval of the TS Change for Section 3.4.13, "RCS Operational Leakage," in License Amendment Request, WBN-TS-99-14. WBN-TS-99-14 was submitted for steam generator alternate repair criteria for axial outside diameter stress corrosion cracking dated April 10, 2000, and subsequent letters dated September 18, 2000, August 22, 2001, and November 8, 2001. Thus, TVA does not have a specific need date for NRC's review and approval of this TS change, WBN-TS-01-12.

No regulatory commitments are contained in this letter. If you have any questions about this proposed TS change, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures
cc: See page 3

Subscribed and sworn to before me
on this 14th day of January 2002

E. Jeannette Long
Notary Public

My Commission Expires May 21, 2005

cc (Enclosures):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. L. Mark Padovan, Senior Project Manager
U.S. Nuclear Regulatory Commission
MS 08G9
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. Lawrence E. Nanny, Director
Division of Radiological Health
3rd Floor
L & C Annex
401 Church Street
Nashville, Tennessee 37243

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1
DOCKET NO. 390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-01-12
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF THE PROPOSED CHANGE

TVA proposes to revise the Watts Bar Nuclear Plant (WBN) Unit 1 Technical specification (TS) to lower the steady state and accident specific activity of the primary coolant and to change the allowable value for Main Control Room (MCR) air intake radiation monitor surveillance requirements (SR).

The proposed changes affect the following sections:

1. TS Section 3.4.16 and applicable Bases, "Reactor Coolant System (RCS) Specific Activity," change Action Condition A (3.4.16.A) and SR 3.4.16.2 from 1.0 microcuries per gram (μCi/gm) Dose Equivalent Iodine 131 (DEI-131) to 0.265 μCi/gm DEI-131.

2. TS Section 3.4.16, Figure 3.4.16-1, "Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER," is being deleted and the maximum value of 21 μCi/gm DEI-131 is being added to TS Required Action 3.4.16.A and 3.4.16.C.

3. TS Section 3.3.7, "CREVS Actuation Instrumentation," Table 3.3.7-1 changes the allowable value to the Control Room Radiation and Control Room Air Intakes for SR 3.3.7.1, 3.3.7.2, and 3.3.7.4 from $\leq$ 5.77E-04 μCi/cubic centimeters (cc) (20,199 counts per minute [cpm]) to $\leq$ 9.45E-05 μCi/cc (3,308 cpm).

The specific changes to the TS and Bases are noted in the marked up copies of the applicable TS pages provided in Enclosure 2.

II. REASON FOR THE PROPOSED CHANGE

TVA is proposing to lower the steady state specific activity of the primary coolant, the 48 hour maximum iodine spiking values, and the surveillance requirement allowable values for control room monitors due to revisions to the WBN main steam line break (MSLB) and steam generator tube rupture (SGTR) dose calculations. The change to those calculations resulted in a change to the allowable RCS iodine activity levels. In addition, those levels resulted in a change to the source terms used to establish the safety limit for the MCR radiation monitors.

III. SAFETY ANALYSIS

Background

TVA revised the MSLB and SGTR calculations to update methodologies, to correct deficiencies, and to resolve several NRC concerns that developed during the review of TVA's proposed License Amendment Request WBN-TS-99-014, "Steam Generator Alternate Repair Criteria for Outside Diameter Stress Corrosion Cracking (ODSCC)," dated April 10, 2000 and subsequent responses to NRC's request for additional information (RAI) dated September 18, 2000, August 22, 2001, November 8, 2001. TVA stated in the August 22, 2001 response to NRC's RAI that the changes to the calculations resulted in a change to the RCS specific activity. That change was being evaluated under TVA's Corrective Action Program and administratively controlled in accordance with NRC's Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety." Approval of this License Amendment Request is dependent upon NRC's approval of the TS Change for Section 3.4.13, "RCS Operational Leakage," in the above License Amendment Request, WBN-TS-99-014. The proposed change in the RCS primary to secondary leakage through any one steam generator is an assumption in the MSLB and the SGTR revised calculation.

Several changes were made in methodology and assumptions to the MSLB analysis. These changes were discussed in detail in the August 22, 2001 response to NRC's RAI concerning the MSLB calculation for License Amendment Request WBN-TS-99-014. Applicable methodologies or assumptions were also incorporated into the SGTR analysis. For NRC's convenience a summary of those changes are provided below:

- Two methods of determining the dose for the MSLB were used as recommended in NUREG-0800, "Standard Review Plan," Section 15.1.5, Appendix A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR:"

  1. A pre-accident Iodine spike,
  2. A maximum steady state DEI-131 with an accident initiated spike consisting of a 500 times increase on the release rate of iodine.

- The more accurate ARCON96 methodology is used to determine atmospheric dispersion coefficients (χ/Q *values)* in lieu of the Halitsky methodology.

- International Commission on Radiation Protection Publication 30, (ICRP)-30, dose conversion factors are used in lieu of ICRP-2. (ICRP-2 dose conversion factors are retained in the calculations for determination of DEI-131 concentrations).

- MCR isolation time was corrected from 14 seconds to 20.6 seconds to account for physical processing time and physical time to isolate the intake.

- Recommendation from the Westinghouse's Nuclear Safety Advisory Letter (NSAL)-00-004 related to RCS operational leakage was used.

- MSLB calculation also incorporated steam generator dry out subsequent to the accident.

- Revision 5 to computer code COROD for MCR dose and Revision 4 to FENCDOSE for offsite dose incorporated the outputs from the use of the above methodology changes.

The assumptions and a summary of the results of the MSLB calculation were provided in the August 22, 2001 letter to NRC. However, TVA is providing this information, for NRC's convenience, in Enclosure 1, Attachment 1 of this letter. Enclosure 1, Attachment 2 provides the assumptions and results for the SGTR calculation.

The revised calculations require a reduction for the WBN TS limit for RCS Specific Activity to ensure the MCR dose and the offsite dose are below the acceptable limits.

According to a study performed in 1990 (James P. Adams and Martin B. Sattison, "Frequency and Consequences Associated with a Steam Generator Tube Rupture Event," Nuclear Technology, Volume 90, May 1990, page 361) in which data on maximum iodine coolant spikes and release rates from fuel defects were reported and statistically analyzed, 12 μCi/gm is a 95 percent (%) probability and 95% confidence upper limit value. That is, there is a 95% probability and 95% confidence that plant iodine spikes will be lower than 12 μCi/gm. Consequently, reducing the 48 hour allowable iodine spike to 21 μCi/gm is reasonable. Iodine spiking levels greater than 21 μCi/gm are not expected, therefore, new dose analyses were not performed for elevated iodine levels at power levels less than 80% as depicted on TS Figure 3.4.16-1. Since this change limits a 48 hour maximum iodine spike of 21 μCi/gm for all power levels, Figure 3.4.16-1 is no longer needed and is deleted by this proposed amendment request. Past operating history has indicated that the WBN RCS Iodine activity has remained very low (typically in the range of 10^{-3} µCi/gm). Consequently, reducing the TS allowable steady state iodine activity to 0.265 µCi/gm DEI-131, is reasonable.

The change in RCS activity described above also resulted in a change to source terms used to establish the safety limit for the MCR radiation monitors which isolate the MCR ventilation subsequent to an accident. The change in safety limit resulted in a change in the allowable concentration (and associated count rate) value for TS SR 3.3.7.1, 3.3.7.2, and 3.3.7.4 in Table 3.3.7-1 for the radiation monitors from ≤ 5.77E-04 µCi/cc (20,199 cpm) to ≤ 9.45E-05 µCi/cc (3,308 cpm).

There are no specific precedence for this TS change since this change results from site specific analysis. However, other plants, such as Sequoyah Nuclear Plant which was approved by NRC November 17, 1998, have reduced the RCS specific activity levels when implementing steam generator alternate repair criteria as recommended in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

Safety Analysis

During the postulated SGTR accident (WBN Updated Final Safety analysis Report (UFSAR) Section 15.4.3), the operator is expected to readily determine that a SGTR has occurred, isolate the faulted steam generator from the intact steam generators, isolate feedwater to the ruptured steam generator, cool down the RCS using the intact steam generators, depressurize the RCS to restore reactor coolant inventory, terminate safety injection to stop primary to secondary leakage, control charging flow, letdown, and pressurizer heaters to prevent repressurization of the RCS and reinitiation of leakage into the ruptured steam generator. During the postulated SGTR accident, the limiting single failure was assumed to be the failure of the power operated relief valve (PORV) on the ruptured steam generator. Failure of this valve in the open position will cause an uncontrolled depressurization of the faulted steam generator which will increase primary to secondary leakage and mass release to the atmosphere. It is assumed the ruptured steam generator PORV fails open when the ruptured steam generator is isolated, and the valve was subsequently isolated by locally closing the associated block valve

11.0 minutes after it is assumed to fail open. The operator actions and the thermal and hydraulic analysis previously performed to determine plant response for a postulate SGTR accident remains unaffected by this proposed change to the TS.

During the postulated MSLB accident (WBN UFSAR Section 15.4.2), it is assumed the operator takes action to cool down and depressurize the plant, and place the residual heat removal (RHR) system into service for further RCS heat removal within 8 hours after the accident. Once the RHR system is placed into service and the RCS has been depressurized, there are no more steam releases from the intact steam generators. Therefore, MSLB accident low population zone (LPZ) doses are based on activity releases for the initial 8 hours following the MSLB. The control room dose calculation is extended beyond 8 hours to 30 days because activity will continue to be in the control room atmosphere and additional time is required to either filter (via recirculation) or purge the activity from the control room atmosphere. The operator actions and the thermal and hydraulic analysis previously performed to determine plant response for a postulated MSLB accident remains unaffected by this proposed change to the TS.

Analyses of the radiological consequences for SGTR and MSLB accidents have been performed to determine the allowable RCS specific iodine activity permissible while maintaining the offsite dose and MCR dose within acceptable limits. The evaluation for the SGTR and MSLB accidents considered both pre-accident and accident initiated iodine spikes as recommended by NUREG-0800. The results of the evaluation show that for a RCS pre-accident spike of 21 μCi/gm DEI-131, the SGTR accident produces a limiting MCR thyroid dose of 15.53 rem and 2 hour Exclusion Area Boundary (EAB) thyroid dose of 13.24 rem. For the accident initiated iodine spike with the RCS specific activity at 0.265 μCi/gm DEI-131, the MSLB accident produces a limiting MCR thyroid dose of 14.47 rem and the SGTR accident produces a limiting 2 hour EAB thyroid dose of approximately 6.4 rem. The permissible primary-to-secondary leakage during the postulated MSLB accident with the RCS specific activity of 0.265 μCi/gm DEI-131 is 1.0 gpm (standard temperature and pressure).

Attachments 1 and 2 provides a summary of the gamma, beta, and thyroid dose results for the MSLB and SGTR in addition to the acceptable dose limits for each accident. The radiological dose analysis show the doses for these accidents will remain within the 10 CFR 100 "Reactor Site Criteria," and 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, "Control Room," dose guideline values and are consistent with NUREG-0800 acceptance criteria for these accidents. Consequently, this proposed change to WBN Unit 1 TS is acceptable.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA proposes to revise the Watts Bar Nuclear Plant (WBN) Unit 1 technical specification for RCS specific activity, to lower the allowable steady state and accident specific activity of the primary coolant and to change the technical specification for the Control Room emergency ventilation system actuation instrumentation allowable value for Main Control Room air intake radiation monitor surveillance requirements.

TVA has concluded that operation of WBN Unit 1 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed technical specification change to reduce the steady state and 48 hour reactor coolant system (RCS) allowable iodine concentrations, and to revise the surveillance requirement value for the Main Control Room air intake radiation monitors does not change any operator actions nor does it change plant systems or structures. Therefore, the proposed change to WBN Unit 1 Technical Specification does not result in a significant increase in the probability of an accident.

The calculated radiological consequences at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) are larger than currently discussed in the Final Safety Analysis Report (FSAR) accidents for the main steam line break (MSLB) and steam generator tube rupture (SGTR) (with the exception of thyroid and beta doses being slightly lower for STGR) accidents. The radiological consequences for the SGTR and MSLB accidents increased due to utilizing more conservative methodologies and more conservative assumptions in the calculation. However, the calculated radiological consequences remain within the limits identified in 10 CFR 100, "Reactor Site Criteria," and General Design Criteria (GDC)-19, "Control Room," and are consistent with NUREG-0800, "Standard Review Plan," acceptance criteria.

The surveillance requirement radiation limit for the Main Control Room air intake radiation monitors will be reduced to compensate for the change in source terms which resulted from the use of the methodology changes in the SGTR accident. This change ensures the monitors perform their safety function of control room isolation during accident conditions and does not increase the probability or consequences of an accident previously evaluated.

In summary, the control room dose, the LPZ dose, and the EAB dose for the SGTR and MSLB remain bounded by the acceptance criteria of NUREG-0800 and continue to satisfy an appropriate fraction of the 10 CFR 100 dose limits and the GDC-19 dose limits. The surveillance requirement changes for the Main Control Room radiation monitors ensure the monitors perform their intended design function. Therefore, the proposed change does not result in a significant increase in the consequences of an accident previously analyzed.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed TS change does not alter the configuration of the plant. The changes do not directly affect plant operation. The change will not result in the installation of any new equipment or system or the modification of any existing equipment or systems. No new operation procedures, conditions or modes will be created by this proposed change. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

The methods for calculating the radiological consequences are revised for the MSLB and SGTR analysis to utilize the thyroid dose conversion factors in International Commission on Radiation Protection Publication 30 (ICRP-30) to calculate the dose and ARCON96 methodology to calculate atmospheric dispersion coefficients.

The calculated radiological consequences at the EAB and LPZ are slightly larger than those noted in the FSAR accidents for the MSLB and SGTR (thyroid and beta doses slightly lower for SGTR) accidents. The radiological dose consequences for the SGTR and MSLB accidents increased due to utilizing more conservative methodologies and more conservative assumptions in the calculation. The calculated dose consequences of the evaluated accidents remain less than the dose limits identified in 10 CFR 100 and GDC-19, and are consistent with NUREG-0800 acceptance criteria. The surveillance requirement for the MCR radiation monitors is being reduced for consistency with lower source terms and to ensure the monitors perform their intended design function of isolating the Main Control Room subsequent to an accident. Therefore, it is concluded that the proposed change to lower the RCS Specific Activity and subsequent changes to the Main Control Room radiation monitors are required to ensure the Main Control Room dose and the offsite dose are below the acceptable limits. Therefore these changes do not result in a significant reduction in the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

ENCLOSURE 1
ATTACHMENT 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
WBN-TS-01-12
REACTOR COOLANT SYSTEM (RCS) SPECIFIC ACTIVITY
MAIN STEAM LINE BREAK DOSE CALCULATION SUMMARY

A calculation has been performed by TVA to determine the acceptable permissible steam generator primary-to-secondary leak rate during a steam line break for WBN Unit 1. 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, General Design Criteria (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 Low Population Zone (LPZ) and the 2 hour Exclusion Area Boundary (EAB) dose).

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

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

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

7. Steam released to the atmosphere
    1. total from the non-defective steam generators (0 - 2 hour [hr]), 480,000 pounds (lb)
    2. total from the non-defective steam generators (2 - 8 hr), 871,000 lb
    3. total from the faulted steam generator (0 - 30 minutes [min]), 150,000 lb steam generator secondary inventory (standard temperature and pressure primary)
    4. total from the faulted steam generator (30 mins - 8 hrs) 1000 lb (standard temperature and pressure) primary-to-secondary leakage

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

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

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

11. Atmospheric dilution factors, χ/Q, (ARCON96) are as follows for the control room (sec/m^3).
    1. 0 - 2 hr - 4.03E-3
    2. 2 - 8 hr - 3.35E-3

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
WBN-TS-01-12
REACTOR COOLANT SYSTEM (RCS) SPECIFIC ACTIVITY
MAIN STEAM LINE BREAK DOSE CALCULATION SUMMARY

12. Main Control Room related assumptions
    1. Volume - 257,198 cubic feet
    2. Makeup/pressurization flow - 711 cubic feet per minute (cfm)
    3. Recirculation flow - 3600 cfm
    4. Unfiltered intake - 51 cfm
    5. Filter efficiency - 95 percent (%) first pass, 70% second pass, 0% for noble gases
    6. Isolation time, T = 20.6 sec.
    7. Occupancy factors:
        1. 0-24 hr - 100%
        2. 1-4 days - 60%
        3. 4-30 days - 40%

Main Steam Line Break Dose Calculation Results[1]

1 gpm Primary-to-Secondary Leakage (ARCON96 χ/Q)	Control Room Operator (rem)	SRP Guidance for GDC 19 Limits (rem)	30-Day LPZ (rem)	2-Hour EAB (Site boundary) (rem)	SRP Guidance for 10CFR100 Limits (rem)
Accident Initiated Iodine Spike Case (0.265 µCi/gm steady state)					
Gamma:	0.010	5	0.068	0.073	2.5
Beta:	0.076	30	0.0167	0.019	30
Thyroid (ICRP-30):	14.47	30	2.662	2.515	30
TEDE:[2]	0.504	5	0.266	0.260	5
Pre-Accident Iodine Spike Case (21 µCi/gm max peak)					
Gamma:	0.006	5	0.009	0.028	25
Beta:	0.055	30	0.003	0.009	300
Thyroid (ICRP-30):	12.09	30	1.011	2.527	300
TEDE:[2]	0.410	5	0.070	0.185	25

1. Dose results are based on the Tritium Program core source terms since these source terms bound the conventional core source terms.
2. TEDE dose numbers are provided for information only and are not intended to be the Licensing Basis for WBN.

ENCLOSURE 1
ATTACHMENT 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
WBN-TS-01-012
REACTOR COOLANT SYSTEM (RCS) SPECIFIC ACTIVITY
STEAM GENERATOR TUBE RUPTURE (SGTR) CALCULATION SUMMARY

A calculation has been performed by TVA to determine the radiological consequences of a Steam Generator Tube Rupture (SGTR) for WBN Unit 1. 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, General Design Criteria (GDC) 19 limit.

The calculation used TVA computer codes STP, FENCDOSE, and COROD. 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 doses).

Assumptions for the Postulated SGTR accident

1. RCS Letdown maximum flow of 124.39 gallons per minute (gpm) is used.

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

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

4. 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.

5. Primary side to secondary side leakage of 150 gallons per day (gpd) standard temperature and pressure per steam generator in the intact loops.

6. Steam released to the atmosphere
   a) total from the non-defective (secondary side) steam generators (0 - 2 hour [hr]), 517,500 pounds (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.

7. 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

8. 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

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

10. Atmosphere dilution factors, χ/Q, (ARCON96) are as follows for control room (sec/ m^3):
   a) 0-2 hrs - 4.03E-3
   b) 2-8 hrs - 3.35E-3

11. 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 percent (%) first pass, 70% second pass, 0% for noble gases
   f) International Commission on Radiation Protection Publication 30 (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%

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

ENCLOSURE 1
ATTACHMENT 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
WBN-TS-01-012
REACTOR COOLANT SYSTEM (RCS) SPECIFIC ACTIVITY
STEAM GENERATOR TUBE RUPTURE (SGTR) CALCULATION SUMMARY

Steam Generator Tube Rupture Dose Calculation Results[1]

	Control Room Operator (rem)	SRP Guidance for GDC 19 Limits (rem)	30-Day LPZ (rem)	2-Hour EAB (Site boundary) (rem)	SRP Guidance for 10CFR100 Limits (rem)
Accident Initiated Iodine Spike Case (0.265 μCi/gm steady state)					
Gamma:	0.087	5	0.131	0.518	2.5
Beta:	1.032	30	0.066	0.249	30
Thyroid:	2.241	30	1.554	6.375	30
TEDE[2]	0.180	5	0.282	1.157	5
Pre-Accident Iodine Spike Case (21 μCi/gm maximum peak)					
Gamma:	0.084	5	0.088	0.359	25
Beta:	0.926	30	0.053	0.209	300
Thyroid:	15.53	30	3.150	13.24	300
TEDE[2]	0.597	5	0.278	1.168	25

1. Doses are based on conventional core source terms. Tritium Production Core TEDE and beta doses are reported in the Tritium submittal.
2. TEDE dose numbers are provided for information only and are not intended to be the Licensing Basis for WBN.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-01-12
MARKED PAGES

I. AFFECTED PAGE LIST

Technical Specification

3.3-60
3.4-39
3.4-40
3.4-42

Technical Specification Bases

B 3.4-93
B 3.4-94
B 3.4-95
B 3.4-96
B 3.4-97
Inserts

II. MARKED PAGES

See attached.

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Manual Initiation	2 trains	SR 3.3.7.3	NA
2.	Control Room Radiation Control Room Air Intakes	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.4	≤ 5.77E-04 μC/cc (20199 cpm)
3.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

REPLACE WITH:
≤ 9.45E-05 μCi/cc
(3,308 cpm)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm. **REPLACE WITH:** > 0.265 **REPLACE WITH:** ≤ 21 μCi/gm	-------------NOTE------------- LCO 3.0.4 is not applicable. ------------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Perform SR 3.4.16.2.	4 hours
	AND	
	B.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F. **REPLACE WITH:** > 21 µCi/gm	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē µCi/gm.	7 days
SR 3.4.16.2	-------------------NOTE-------------------- Only required to be performed in MODE 1. --- Verify reactor coolant D~~OSE EQUIVALENT~~ I-131 specific activity ≤ 1.0 µCi/gm. **REPLACE WITH:** ≤ 0.265 µCi/gm	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

DELETE GRAPH

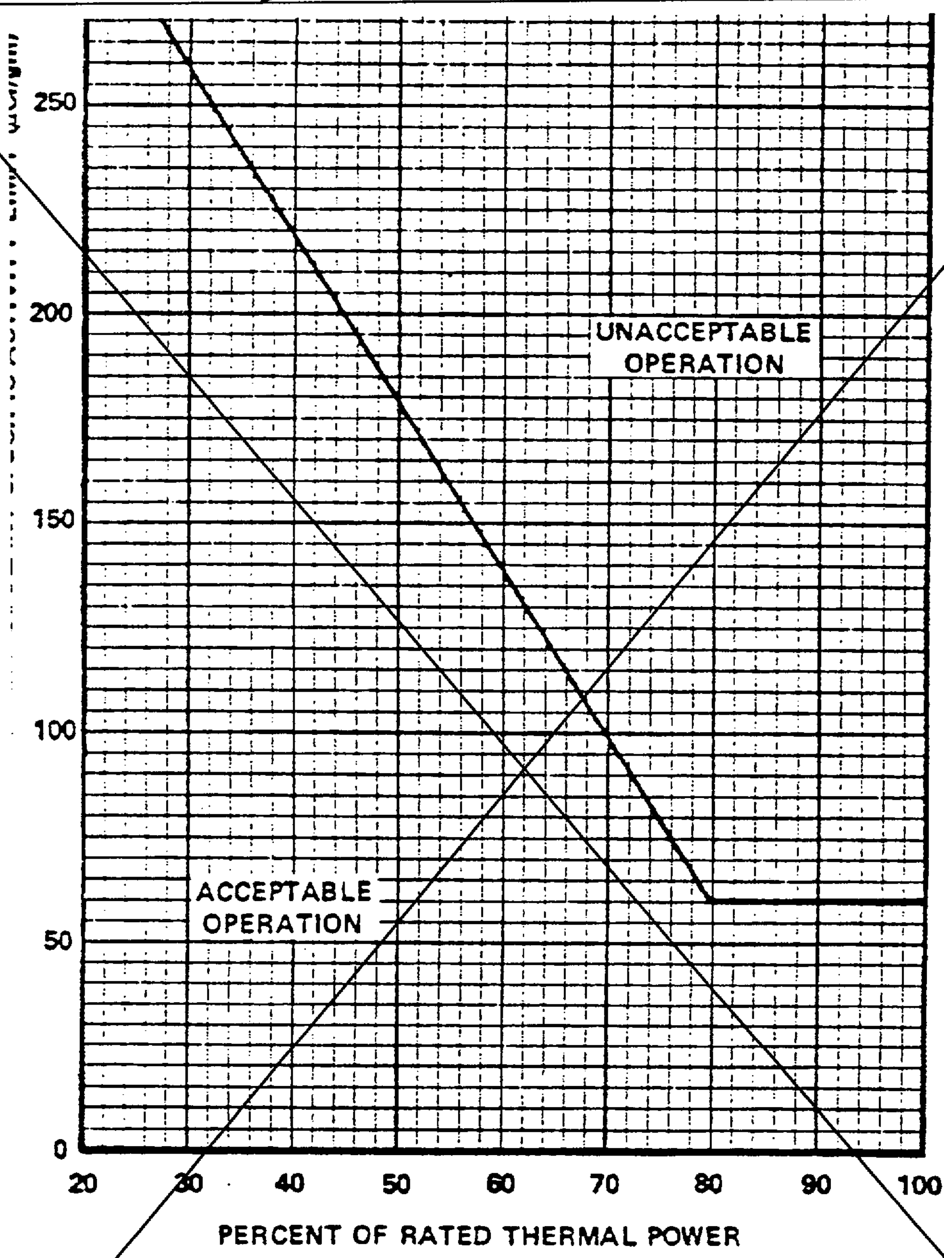

Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

INSERT A

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits **and within the 10 CFR 50, Appendix A, GDC 19 limits** during analyzed transients and accidents.

ADD

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite **and Main Control Room** radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) **or main steam line break (MSLB)** accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small ~~fraction of the 10 CFR 100 dose guideline limits.~~ The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

DELETE. INSERT B

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

DELETE

INSERT C

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary **and Main control Room accident doses** will not exceed the appropriate 10 CFR 100 dose guideline limits **and 10 CFR 50, Appendix A, GDC 19 limits** following a SGTR **or MSLB** accident. The SGTR **and MSLB** safety analyses (Ref. 2) assume the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of **150 gallons per day (GPD)**. The safety analyses assume the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.14, "Secondary Specific Activity."

REVISE TO READ

(continued)

REPLACE WITH: these analyses

BASES

INSERT

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR **and MSLB accidents establish** the acceptance limits for RCS ~~specific activity. Reference~~ to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

REPLACE WITH: 0.265

DELETE AND INSERT D

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.

REPLACE WITH: 21

DELETE

ADD:

The analysis also assumes a loss of offsite power at the same time as the SGTR **and MSLB** event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. **The MSLB results in a reactor trip due to low steam pressure.**

ADD

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

DELETE AND INSERT E

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131.

DELETE

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR

(continued)

BASES **DELETE**

APPLICABLE SAFETY ANALYSES (continued)

accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the ~~NRC~~ Policy Statement.

REPLACE WITH: 0.265

LCO

The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by Ē (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

DELETE AND INSERT F

The SGTR **and MSLB** accident analysis (Ref. 2) shows that the ~~2 hour site boundary~~ dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR **or MSLB** lead to site boundary doses that exceed the ~~10 CFR~~ 100 dose guideline limits.

ADD

INSERT G

ADD: and Main Control Room accident dose

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity ~~are~~ necessary to contain the potential consequences of an SGTR to within the acceptable **Main Control Room and** site boundary dose values.

ADD

REPLACE WITH: accident

For operation in MODE 3 with RCS average temperature < 500°°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

REPLACE WITH: limit of 21 μCi/gm is

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples ~~at intervals of 4 hours must be~~ taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(s) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1 and B.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1

REPLACE WITH:
greater than
21 μCi/gm

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following rapid power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

(continued)

TECHNICAL SPECIFICATION BASES 3.4.16 INSERTS
WBN-TS-01-12

INSERT A

The maximum dose to the whole body and the thyroid that an individual occupying the Main Control Room can receive for the accident duration is specified in 10 CFR 50, Appendix A, GDC 19.

INSERT B

... and ensure the Main Control Room accident dose is within 10 CFR 50, Appendix A, GDC 19 dose guideline limits.

INSERT C

The evaluations showed the potential offsite and Main Control Room dose levels for a SGTR and MSLB accident were within the appropriate 10 CFR 100 and GDC 19 dose guideline limits.

INSERT D

... an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state iodine concentration of 0.265 µCi/gm DOSE EQUIVALENT I-131.

INSERT E

The safety analysis shows the radiological consequences of SGTR and MSLB accidents are within the appropriate 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do no exceed 21 µCi/gm DOSE EQUIVALENT I-131, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 21 µCi/gm DOSE EQUIVALENT I-131.

INSERT F

and accident dose to personnel in the Main Control Room during the Design Basis Accident (DBA) will be within the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the DBA will be within the allowed whole body dose.

TECHNICAL SPECIFICATION BASES 3.4.16 INSERTS
WBN-TS-01-12
(continued)

INSERT G

… or Main Control Room accident dose that exceed the 10 CFR 50, Appendix A, GDC 19 dose limits.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN)
UNIT 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-01-12
REVISED PAGES

I. AFFECTED PAGE LIST

Technical Specification

3.3-60
3.4-39
3.4-40

Technical Specification Bases

B 3.4-93
B 3.4-94
B 3.4-95
B 3.4-96
B 3.4-97

II. REVISED PAGES

See attached.

Table 3.3.7-1 (page 1 of 1)
CREVS Actuation Instrumentation

	FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Manual Initiation	2 trains	SR 3.3.7.3	NA
2.	Control Room Radiation Control Room Air Intakes	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.4	≤ 9.45E-05 μCi/cc (3,308 cpm)
3.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 0.265 μCi/gm.	------------NOTE------------ LCO 3.0.4 is not applicable. --------------------------------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 21 μCi/gm	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Perform SR 3.4.16.2.	4 hours
	<u>AND</u>	
	B.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 > 21 μCi/gm.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ 100/Ē μCi/gm.	7 days
SR 3.4.16.2	-------------------NOTE-------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.265 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The maximum dose to the whole body and the thyroid that an individual occupying the Main Control Room can receive for the accident duration is specified in 10 CFR 50, Appendix A, GDC 19. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits and within the 10 CFR 50, Appendix A, GDC 19 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite and Main Control Room radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) or main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits, and ensure the Main Control Room accident dose is within the appropriate 10 CFR 50, Appendix A, GDC 19 dose guideline limits.

The evaluations showed the potential offsite and Main Control Room dose levels for a SGTR and MSLB accident were within the appropriate 10 CFR 100 and GDC 19 guideline limits.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary and Main Control Room accident doses will not exceed the appropriate 10 CFR 100 dose guideline limits and 10 CFR 50, Appendix A, GDC 19 dose guideline limits following a SGTR or MSLB accident. The SGTR and MSLB safety analysis (Ref. 2) assume the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gallons per day (GPD). The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.14, "Secondary Specific Activity."

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the SGTR and MSLB accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analyses are for two cases of reactor coolant specific activity. One case assumes specific activity at 0.265 μCi/gm DOSE EQUIVALENT I-131 with an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state iodine concentration of 0.265 μCi/gm DOSE EQUIVALENT I-131. The second case assumes the initial reactor coolant iodine activity at 21 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant equals the LCO limit of $100/\bar{E}$ μCi/gm for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR and MSLB event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal. The MSLB results in a reactor trip due to low steam pressure.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of a SGTR and MSLB accident are within the appropriate 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 21 μCi/gm DOSE EQUIVALENT I-131, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 21 μCi/gm DOSE EQUIVALENT I-131.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

LCO

The specific iodine activity is limited to 0.265 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the Design Basis Accident (DBA) will be within the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary and accident dose to personnel in the Main Control Room during the DBA will be within the allowed whole body dose.

The SGTR and MSLB accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels and Main Control Room accident dose are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SGTR or MSLB, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits, or Main control Room accident dose that exceed the 10 CFR 50, Appendix A, GDC 19 dose limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an accident to within the acceptable Main Control Room and site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limit of 21 μCi/gm is not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(s) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1 and B.2

With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is greater than 21 μCi/gm, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following rapid power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

(continued)