November 17, 1998

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA2194 AND MA2195)(TS 98-02)

Dear Mr. Scalice:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-77 and Amendment No. 227 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. These amendments are in response to your application dated June 26, 1998. The amendments revise the SQN Technical Specifications by lowering the specific activity limit of the primary coolant from 1.0 microcurie/gram dose equivalent iodine-131 (^{131}I) to 0.35 microcuries/gram dose equivalent ^{131}I, as well as the instantaneous ^{131}I values in Figure 3.4-1. These changes are provided for in NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." This change allows a proportional increase in main steam line break induced primary-to-secondary leakage when implementing the alternate steam generator tube repair criteria, which the NRC has already approved for Units 1 and 2.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. Please direct any questions you or your staff should have to me at 301-415-2010.

Sincerely,

Original signed by:

Ronald W. Hernan, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000327
P PDR

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 237 to License No. DPR-77
2. Amendment No. 227 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

Distribution (w/enclosure):

[redacted]	W. Beckner
PUBLIC	G. Hill (4)
SQN r/f	T. Harris (TLH3 w/ SE)
L. Plisco, RII	J. Zwolinski (A)
ACRS	OGC
R. Hernan	B. Clayton
C. Miller	F. Hebdon
J. Hayes	L. Brown

DF01

Document Name: G:\SQN\2194.AMD

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* See previous concurrence

OFFICE	PDII-3/PM		PDII-3/LA		OGC		PDII-3/D	C
NAME	RHernan		BClayton		MYoung*		FHebdon	
DATE	11/17/98		11/17/98		11/3/98		11 ~~10~~/ 17 /98	

OFFICIAL RECORD COPY

November 17, 1998

Mr. J. A. Scalice
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Executive Vice President
Tennessee Valley Authority
6A Lookout Place
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Chattanooga, TN 37402-2801

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Project Directorate II-3
Division of Reactor Projects - I/II
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Docket Nos. 50-327 and 50-328

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OFFICE	PDII-3/PM		PDII-3/LA		OGC		PDII-3/D	C
NAME	RHernan		BClayton		MYoung*		FHebdon	
DATE	11/17/98		11/17/98		11/3/98		11/17/98	

OFFICIAL RECORD COPY

Mr. J. A. Scalice
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 17, 1998

Mr. J. A. Scalice
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENTS FOR THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. MA2194 AND MA2195)(TS 98-02)

Dear Mr. Scalice:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-77 and Amendment No. 227 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. These amendments are in response to your application dated June 26, 1998. The amendments revise the SQN Technical Specifications by lowering the specific activity limit of the primary coolant from 1.0 microcurie/gram dose equivalent iodine-131 (^{131}I) to 0.35 microcuries/gram dose equivalent ^{131}I, as well as the instantaneous ^{131}I values in Figure 3.4-1. These changes are provided for in NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." This change allows a proportional increase in main steam line break induced primary-to-secondary leakage when implementing the alternate steam generator tube repair criteria, which the NRC has already approved for Units 1 and 2.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. Please direct any questions you or your staff should have to me at 301-415-2010.

Sincerely,

Ronald W. Hernan

Ronald W. Hernan, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 237 to License No. DPR-77
2. Amendment No. 227 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. DPR-77

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 26, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 237 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 17, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 237

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4 4-19	3/4 4-19
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
B 3/4 4-4a	B 3/4 4-4a
B 3/4 4-5	B 3/4 4-5

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.35 microcuries/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*

a. With the specific activity of the primary coolant greater than 0.35 microcuries/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. R121

b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. R121

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant greater than 0.35 microcuries/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

R121

*With T_{avg} greater than or equal 500°F.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 0.35 μCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

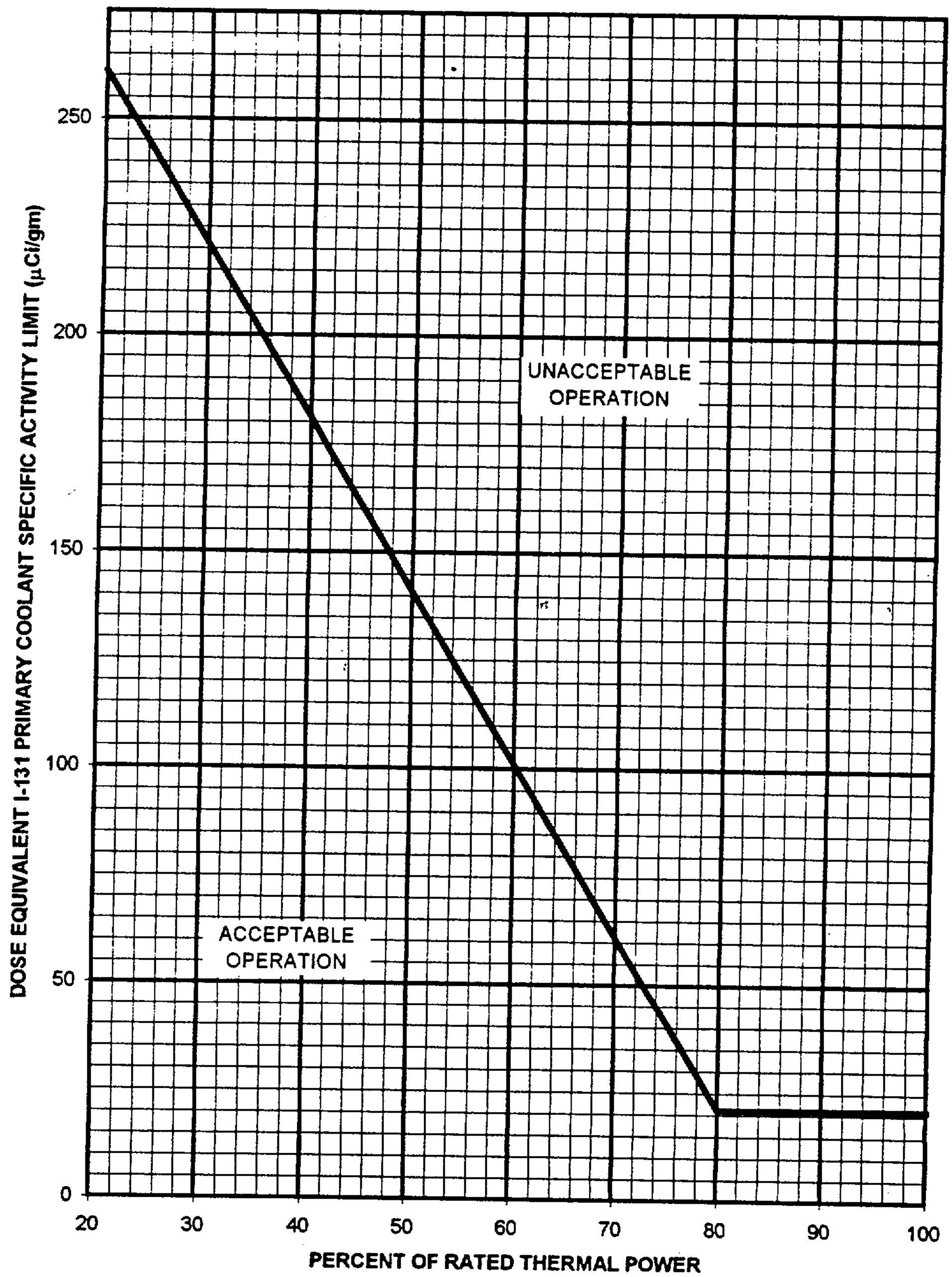

FIGURE 3.4-1
DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.35 μCi/gram Dose Equivalent I-131

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurances of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 600 gallons per day for all steam generators and 150 gallons per day for any one steam generator will minimize the potential for a significant leakage event during steam line break. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 8.21 gpm at atmospheric conditions and 70°F in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 8.21 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 8.21 gpm. R226

The 150-gallons per day limit incorporated into SR 4.4.6 is more restrictive than the standard operating leakage limit and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that, should a significant leak be experienced, it will be detected, and the plant shut down in a timely manner.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

R226

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Sequoyah Nuclear Plant site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than or equal to 0.35 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.35 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 should be limited to no more than 800 hours per year since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. R121

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 227
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 26, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 227 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 17, 1998

ATTACHMENT TO LICENSE AMENDMENT NO.227

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4 4-24	3/4 4-24
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.35 microcurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries per gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*:

R107

a. With the specific activity of the primary coolant greater than 0.35 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

b. With the specific activity of the primary coolant greater than 100/$\bar{E}$ microcurie per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. R107

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant greater than 0.35 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/E microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

R107

*With T_{avg} greater than or equal to 500°F.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for Ē Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 0.35 μCi/gram DOSE EQUIVALENT I-131 or 100/Ē μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

#Until the specific activity of the primary coolant system is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subcritical for 48 hours or longer.

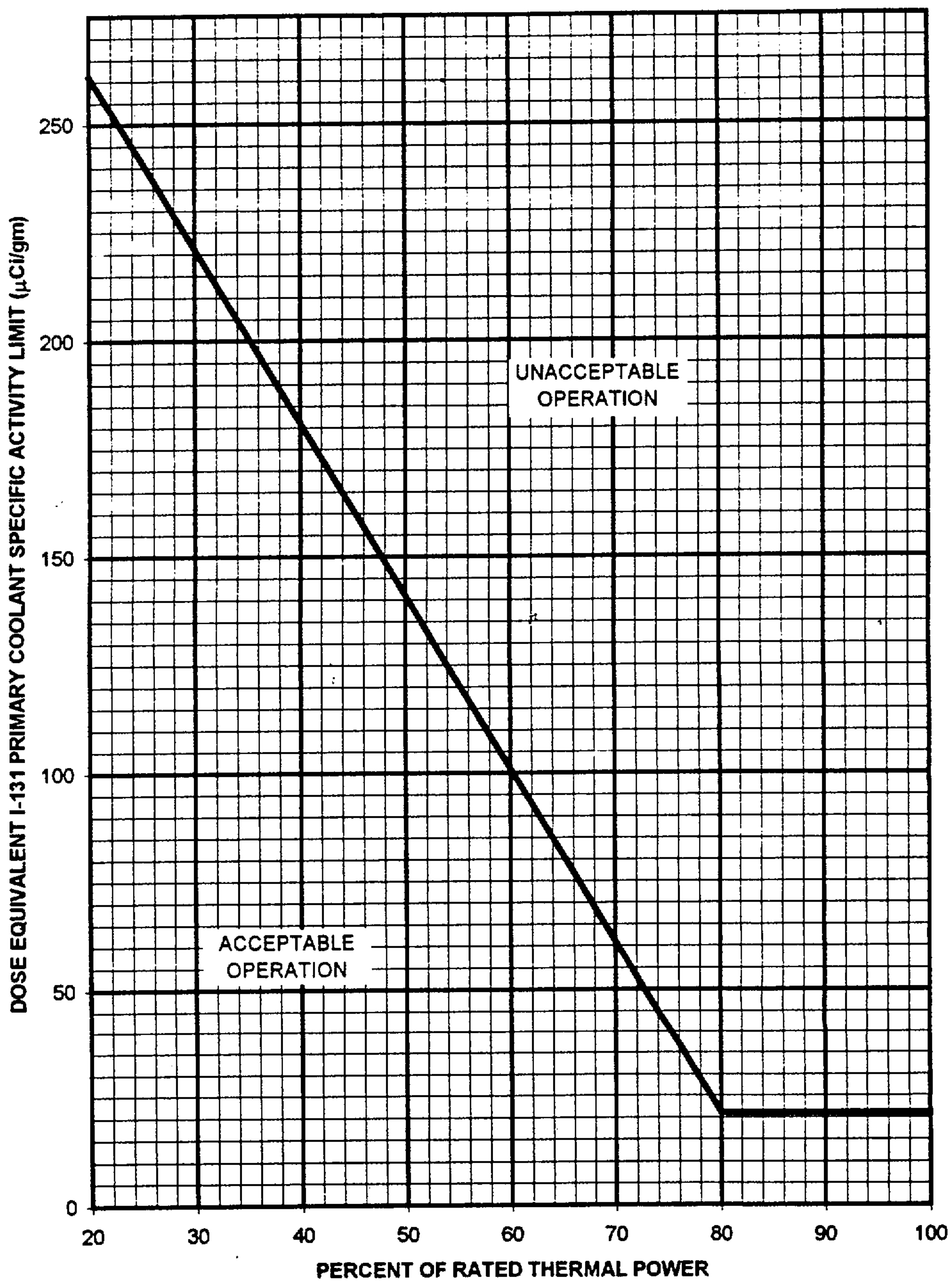

FIGURE 3.4-1

DOSE EQUIVALENT I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 0.35 µCi/gram Dose Equivalent I-131

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurances of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 600 gallons per day for all steam generators and 150 gallons per day for any one steam generator will minimize the potential for a significant leakage event during steam line break. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 8.21 gpm at atmospheric conditions and 70 °F in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 8.21 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 8.21 gpm. R213

The 150-gallons per day limit incorporated into SR 4.4.6 is more restrictive than the standard operating leakage limit and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that, should a significant leak be experienced, it will be detected, and the plant shut down in a timely manner.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

R213

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Sequoyah site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.35 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.35 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 should be limited to no more than 800 hours per year since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. R107

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

In a letter to the U.S. Nuclear Regulatory Commission (NRC) dated June 26, 1998, the Tennessee Valley Authority (TVA) requested amendments to Operating Licenses DPR-77 and DPR-79 for Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. The amendments would modify Technical Specification (TS) Figure 3.4-1, reducing the maximum allowable instantaneous value of dose equivalent iodine-131 (^{131}I) in primary coolant as a function of reactor power level and modify TS 3.4.8.a, reducing the 48-hour value for dose equivalent ^{131}I in primary coolant. In conjunction with these proposed changes, TVA also planned to increase the maximum allowable primary-to-secondary leakage for the faulted steam generator (SG) assumed in their analysis of the consequences of a main steamline break (MSLB) accident. The proposed increase was from 3.7 gpm to 8.21 gpm at 70°F.

With the proposed change to TS Figure 3.4-1, the unacceptable range of operation, the maximum allowable specific activity level at which the plant would be required to initiate shutdown actions at rated thermal power levels of 80% or above, would now be at primary coolant activity levels greater than 20 microcuries per gram (µCi/g) of dose equivalent ^{131}I. Previously, this value had been 60 µCi/g. A similar reduction in primary coolant activity levels of dose equivalent ^{131}I in the power range 20-80% was incorporated into the figure.

TVA proposed to lower the 48-hour value of dose equivalent ^{131}I in primary coolant in TS 3.4.8.a to 0.35 µCi/g from the previous value of 1.0 µCi/g. TVA considered such a reduction in primary coolant activity level appropriate with the concomitant increase in primary to secondary leakage and consistent with the guidance provided in NRC Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

2.0 BACKGROUND

TVA previously requested changes to the SQN Units 1 and 2 TSs (via Change Requests 95-15 and 95-23) to add an alternate SG tube plugging criteria for tubes with outside diameter stress corrosion cracking (ODSCC) indications at nondented tube support plate intersections in accordance with GL 5-05. The NRC approved these changes in TS Amendments 214 (Unit 1) and 211 (Unit 2), dated October 11, 1995, and April 3, 1996, respectively. At the time that these amendment requests were made, SQN did not request a decrease in the specific activity of the primary coolant dose equivalent ^{131}I in order to expedite the review process. The NRC staff concluded that such a reduction in primary coolant activity level was not required since the NRC calculated doses for the control room, the Exclusion Area Boundary (EAB), and the Low Population Zone (LPZ), still met the staff's acceptance criteria for an MSLB accident even with the increase in primary to secondary leakage resulting from the implementation of the alternate repair criteria.

The alternate SG tube plugging criteria require each indication left in service be evaluated and assigned a leakage quantity that would be postulated to occur in the event of an MSLB accident. The summation of the leakage quantity must be less than a specified value so that the resultant offsite dose would be a fraction of the 10 CFR Part 100 offsite allowable dose limit. Since the offsite dose is directly dependent on the reactor coolant specific activity of dose equivalent ^{131}I, decreasing the maximum allowable reactor coolant specific activity will allow a larger quantity of tubes with axial ODSCC to remain in service by allowing a proportional increase in primary to secondary leakage during a postulated MSLB accident.

Currently, TVA projects the end-of-cycle MSLB leak rate from tubes left in service to be less than 4.0 gallons per minute (gpm) total or 3.7 gpm for the SG in the faulted loop based on the reactor coolant specific activity limited to 1.0 μCi/g dose equivalent ^{131}I. This projection includes a probability of detection adjustment, allowances for nondestructive examination uncertainties, and ODSCC growth rates. If the maximum allowable 48-hour value dose equivalent ^{131}I in primary coolant activity is reduced to 0.35 μCi/g, the primary-to-secondary leakage rate associated with an MSLB accident could be increased to 11.9 gpm total or 11.6 gpm for the SG in the faulted loop at operating conditions (2250 psia and 590°F). This leak rate equates to 8.51 gpm total and 8.21 gpm for the faulted loop at atmospheric conditions and 70°F.

Presently, indications less than the alternate repair limit (i.e., indications allowed to remain in service by the alternate plugging criteria) are required to be plugged or repaired in order to prevent the allowable leakage limit from being exceeded.

3.0 EVALUATION

3.1 Assessment of Radiological Consequences

3.1.1 Background

TVA performed an evaluation to determine the maximum permissible SG primary-to-secondary leak rate during an MSLB. The evaluation considered both pre-existing and accident-initiated iodine spike cases as required by GL 95-05.

TVA selected 30 rem as the thyroid dose acceptance criteria for an MSLB, with an assumed accident-initiated iodine spike, based on the guidance of the Standard Review Plan (NUREG-0800) Section 15.1.5, Appendix A. They selected 50 percent of the 10 CFR Part 100 thyroid dose guideline, or 150 rem, as the thyroid dose acceptance criteria for the case of a pre-existing iodine spike. TVA also calculated the whole-body doses to both offsite and control room personnel, as well as the skin and thyroid doses to control room personnel. The EAB doses were calculated for the initial 2-hour period following the MSLB. 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 reactor coolant system (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 faulted and intact SGs. Thus, LPZ and control room doses are based on activity releases for the initial 8 hours following the MSLB. TVA extended the control room dose calculation beyond 8 hours (to 30 days) because activity will remain in the control room atmosphere beyond the period of time in which it is brought initially into the control room. This activity remains in the control room until it is removed either via filtration and purging due to outleakage from the control room. TVA determined that the quantity of activity was no longer significant after 24 hours. After that there was no increase in the thyroid dose.

TVA indicated that the methodology and assumptions performed in support of this amendment request are the same as the evaluation documented in Section 4.0 of Westinghouse Commercial Atomic Power Topical Report 13990 (WCAP-13990), entitled "Sequoyah Units 1 and 2 SG Tube Plugging Criteria for Indications at Tube Support Plates," May 1994, with the following differences (WCAP-13990 was transmitted to NRC with SQN TS Changes 95-15 and 95-23):

1. Initial primary coolant iodine activity - 0.35 µCi/g dose equivalent ^{131}I.

2. Initial secondary coolant iodine activity - 0.1 µCi/g dose equivalent ^{131}I (at NRC request during a previous submittal).

3. The iodine partition coefficient for the primary-to-secondary leakage in the intact SGs was assumed to be 0.01 to reflect that the leakage is below the mixture level.

The results of the TVA evaluation led them to conclude that the thyroid dose at the EAB for the accident-initiated spike case yields the limiting leak rate. TVA drew this conclusion based on a 30-rem thyroid dose for the accident-initiated spike case and initial primary and secondary coolant iodine activity levels of 0.35 µCi/g and 0.1 µCi/g dose equivalent ^{131}I, respectively. Based upon these conditions, TVA calculated a limiting leak rate of 11.9 gpm total or 11.6 gpm to the faulted SG at operating conditions of 2250 psia and 590°F. At atmospheric conditions and 70°F, this leak rate equates to 8.51 gpm total and 8.21 gpm for the faulted SG and 150 gallons per day (approximately 0.1 gpm/SG) for the three intact SGs.

3.1.2 Analysis

The staff has reviewed the licensee's calculations and performed confirmatory dose calculations for an MSLB accident using the methodology associated with Standard Review Plan (SRP) 15.1.5, Appendix A. Two assessments were performed. One was based upon a pre-existing iodine spike activity level of 20 µCi/g of dose equivalent ^{131}I in primary coolant and the other was based upon an accident-initiated iodine spike activity level of 0.35 µCi/g of dose equivalent ^{131}I in primary coolant. For the accident-initiated spike, the staff assumed that the accident-initiated spike case resulted in an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 0.35 µci/g of dose equivalent ^{131}I. For the accident-initiated spike and the pre-existing spike cases, the staff calculated doses for individuals located at the EAB, LPZ and the control room. The parameters that were used in the staff's assessment are shown in Table 1. The resulting doses calculated by the staff are shown in Table 2.

In WCAP-13990, the atmospheric dispersion factor, x/Q, for the 0-8 hour period for the LPZ and control room was divided into two increments, 0-2 hours and 2-8 hours. During conference calls with TVA on September 17, 21, and 25, 1998, these x/Q values were discussed. TVA explained that the SQN interim plugging criteria amendment x/Q values are the same values which were presented in Table 15A-2 of the SQN Final Safety Analysis Report (FSAR). These values were calculated from a year's worth of site meteorological data from the period April 1971 through March 1972 and remain unchanged since the plant was originally licensed. TVA did not make separate calculations of the control room x/Q values specifically for the postulated MSLB accident for SQN. Instead, TVA explained that the values used in this MSLB amendment request analysis (i.e., the factor of two increase above the SQN loss-of-coolant accident (LOCA) x/Q values) reflect engineering judgment based upon the SQN building geometry and a comparison with the ratio of the MSLB and LOCA control room x/Q values calculated for the Watts Bar plant. The licensee noted the strong similarity of the Watts Bar and SQN plant building geometries. In addition, the staff noted that the wind speed estimates used in the licensee's assessment were somewhat different than expected, based on data presented in Table 2.3.2-1 of the SQN FSAR.

Since the licensee's control room x/Q values for an MSLB accident release point were based on the application of engineering judgments to the LOCA control room x/Q values, for the confirmatory calculations, the staff decided to utilize the staff's operating license safety evaluation x/Q values for the LPZ and to calculate a x/Q value for the control room using the Murphy/Campe methodology of SRP 6.4. The staff calculated a value for the control room and applied it for the 0-2 hour period. To estimate a 2-8 hour value, the staff then took the 0-2 hour value and divided it by a factor of 4 in accordance with the guidance of SRP 6.4 for control room designs with dual inlets and the capability to manually select the inlet with the lower inlet activity level. The control room design qualified for this factor of 4 by meeting the Seismic Category I, tornado missile, redundant radiation monitor requirements, and the single failure criteria presented in SRP 6.4.

With the incorporation of the above noted x/Q values, the staff calculated the LPZ and control room operator doses to be within the guidelines of SRP 15.1.5, Appendix A, and SRP 6.4. Therefore, the staff concluded that the licensee's analyses are acceptable and that a leak rate of 8.21 gpm is an acceptable limit for the maximum primary-to-secondary leakage initiated in the faulted SG for the MSLB accident when coupled with the proposed change in primary coolant activity levels of dose equivalent ^{131}I.

GL 95-05 states that lowering the primary coolant dose equivalent ^{131}I activity is an acceptable means for accepting higher projected primary-to-secondary SG leakage rates during a postulated MSLB accident. Therefore, based on the above stated dose values being a small fraction of 10 CFR Part 100 dose guideline values and consistent with SRP (NUREG-0800) acceptance criteria. Therefore, the proposed changes to TS Figure 3.4-1 to change the dose equivalent ^{131}I coolant specific activity limit versus percent of rated power, and to reduce the primary coolant specific activity limit from 1.0 to 0.35 μCi/g in TS 3.4.8, are acceptable. Associated changes were also made to the TS BASES consistent with the corresponding analysis of the MSLB.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 17235, dated April 8, 1998). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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TABLE I

INPUT PARAMETERS FOR SQN EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary coolant concentration of 20 µCi/g of dose equivalent ^{131}I.

<u>Pre-existing Spike Value (µCi/g)</u>

I-131	=	15.4
I-132	=	5.6
I-133	=	24.7
I-134	=	3.5
I-135	=	13.6

2. Volume of primary coolant and secondary coolant:

Primary Coolant Volume (ft^3)	12,600
Primary Coolant Temperature (°F)	590
Mass of Primary Coolant (lb)	554,000
Secondary Coolant Steam Volume (ft^3)	3,546
Secondary Coolant Liquid Volume (ft^3)	2,322
Secondary Coolant Steam Temperature (°F)	526.2
Secondary Coolant Feedwater Temperature (°F)	434.6

3. Technical Specifications limits for DE I-131 in the primary and secondary coolant:

Primary Coolant DE I-131 concentration (µCi/g)	0.35
Secondary Coolant DE I-131 concentration (µCi/g)	0.10

4. Technical Specifications value for the primary to secondary leak rate.

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	600

5. Maximum primary-to-secondary leak rate to the faulted and intact SGs:

Faulted SG(gpm)	8.21
Intact SGs (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.01
Primary to Secondary Leakage	1.0

7. Steam Released to the environment

Faulted SG (lbs, 0-2 hours)	87,000
Intact SGs (lbs, 0-2 hours)	479,000
(lbs, 2-8 hours)	1,030,000

8. Letdown Flow Rate (gpm) 75

9 . Release Rate for 0.35 μCi/g DE I-131:

		Ci/hour
I-131	=	1,750
I-132	=	4,300
I-133	=	4,415
I-134	=	6,316
I-135	=	4,515

10. Atmospheric Dispersion Factors (s/m^3)

EAB (0-2 hours)	1.64×10^{-3}
LPZ (0-8 hours)	1.96×10^{-4}
Control Room (0-2 hours)	5.2×10^{-3}
Control Room (2-8 hours)	1.3×10^{-3}

11. Control Room Parameters

Filter Efficiency (%)	95
Volume (ft^3)	260,000
Makeup flow (cfm)	1,000
Recirculation Flow (cfm)	2,600
Unfiltered Inleakage (cfm)	51
Occupancy Factors	
0-1 day	1.0

Table 2

THYROID DOSES FROM SQN MAIN STEAM LINE BREAK ACCIDENT (REM)

LOCATION	DOSE	
	Pre-existing Spike	Accident-initiated Spike**
EAB	47.5*	26.9
LPZ	7.0*	15
Control Room**	14.6	21

* Acceptance Criterion = 300 rem thyroid
** Acceptance Criterion = 30 rem thyroid