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PSEG
Nuclear LLC

U. S. Nuclear Regulatory Commission
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**IMPLEMENTATION OF ALTERNATIVE SOURCE TERM (AST)
REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS AND
UPDATED FINAL SAFETY ANALYSIS REPORT
SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311**

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to Appendix A of the Technical Specifications for the Salem Nuclear Generating Station, Units 1 and 2. In accordance with 10 CFR 50.91 (b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG proposes, in accordance with the provisions of 10 CFR 50.67, "Accident Source Term" to revise the Salem Units 1 and 2 source term used for design basis radiological analyses. This revision follows guidance set forth by the Nuclear Regulatory Commission (NRC) in Regulatory Guide (RG) 1.183. PSEG is additionally requesting changes to the Technical Specifications and UFSAR to revise certain requirements based on the radiological dose analysis margins obtained by the application of AST. A discussion of the proposed Technical Specifications changes is provided in Attachment 1. A summary of the calculations performed to evaluate the TEDE doses for the Design Basis Accidents is also contained in Attachment 1. The marked-up and proposed Technical Specifications pages are provided in Attachment 2. The marked-up and proposed changes to certain portions of the UFSAR are provided in Attachment 3. PSEG calculations completed to support the revised dose analysis for this submittal are available for your review upon request.

As stated in the 180-day response to Generic Letter 2003-01 (LR-N03-0471 dated December 9, 2003), PSEG is submitting this request for amendment to revise the dose analysis for Salem Units 1 and 2 to bring the plant in alignment with the dose analysis in regards to Control Room Envelope (CRE) unfiltered inleakage. For Salem Units 1 and 2, tracer gas testing was performed from May 31 to June 4, 2003, to measure the inleakage to the Salem CRE. The results of the Tracer Gas Testing are included in this submittal.

PSEG has evaluated the proposed Technical Specifications changes and has determined that they do not involve a significant hazards consideration as defined in 10CFR 50.92. The basis for this determination is provided in Attachment 1. PSEG has also determined that operation with the proposed changes will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion as set forth in 10CFR 51.22 (c) (9). Pursuant to 10CFR 51.22 (b), no environmental impact statement or environmental assessment is needed in conjunction with the approval of the proposed changes. The basis for our determination is provided in Attachment 1. A summary of the resulting doses is included with this submittal page 59, Attachment 1.

PSEG requests NRC Staff review of this submittal and issuance of a license amendment by April 30, 2005. Due to the number of program and procedure changes necessary for implementation, PSEG requests ninety days implementation from the issuance date of the amendment.

Should you have any questions regarding this letter, please contact Mr. John Nagle at 856-339-3171.

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on 4/26/04

Sincerely,

D. F. Garchow
Vice President
Engineering/Technical Support

Attachments (5)

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Salem Nuclear Generating Station Units 1 and 2
Facility Operating Licenses DPR-70 and DPR-75

Attachment 1

Evaluation of Application
Adopting Alternative Source Term
Regulatory Guide 1.183

Table of Contents

1.0 BACKGROUND

The current Salem source term was developed using Technical Information Document (TID)-14844, Calculation of Distance Factors for Power and test Reactor Sites, (Reference 1). "During the past 30 years substantial additional information on fission product releases has been developed based on significant severe accident research." This research "now allows more realistic estimates of the "source term" release into containment, in terms of timing, nuclide types, quantities, and chemical form, given a severe core-melt accident." As such, 10 CFR 50.67 allows plants whose initial operating license was issued before January 10, 1997, to seek to revise their current accident source term used in their design basis radiological analysis.

2.0 DESCRIPTION

PSEG is voluntarily applying for NRC approval to replace the current source term with the alternative source term described in Attachment 2. The alternate source term proposed in this request adheres to NRC guidance provided by Regulatory Guide (RG) 1.183 (Reference 3) except as otherwise stated. Under previous licensing actions, PSEG previously submitted the selective application of Alternate Source Term licensing changes as they apply to the Fuel Handling Accident. (PSEG submittal associated with Amendment 251 for Unit 1 and Amendment 232 for Unit 2)

This request contains an evaluation of the consequences of design basis accidents (DBAs) previously analyzed in the Salem safety analysis report. In accordance with provisions set forth in 10 CFR 50.67, "Accident Source Term," PSEG will demonstrate with reasonable assurance that:

1. A person located at the boundary of the Salem exclusion area during any 2-hour period following the onset of the postulated fission product release, would receive a radiation dose not exceeding 25 Roentgen-Equivalent-Man (rem) total effective dose equivalent (TEDE).
2. A person located at the outer boundary of the Salem low population zone, who is exposed to a radioactive cloud emitted by the postulated fission product release for the entire time of its passage, would receive a radiation dose not exceeding 25 rem TEDE.
3. Adequate radiation protection is provided to permit access to and occupancy of the Salem and Hope Creek control rooms for the entire duration of the postulated accident, with no person in the control room receiving radiation exposure that exceeds 5 rem TEDE.

Assumptions and Methodologies

Safety evaluations required by 10 CFR 50.34 form the basis for determinations made under 10 CFR 50.92 and § 50.59. Under 10 CFR 50.67, applicants for AST are required to re-analyze these safety evaluations. Salem design-basis analyses presented herein use the assumptions and models selected by the NRC staff and defined in RG 1.183, except where otherwise stated, to provide appropriate and prudent safety margins. However, PSEG Nuclear has developed a conservative model for the transport of the amount of radioiodine that is postulated to be available for release to the environment due to leakage from ESF systems that recirculate sump water outside of the primary containment. The model is an alternative to the deterministic approach indicated in the assumptions for ESF system leakage provided in Appendix A of RG 1.183.

Except as otherwise stated, PSEG takes credit for Engineered Safety Features and other appropriately qualified, safety-related, accident mitigation features. In some cases, PSEG has opted to not take credit for a qualified accident mitigation feature in order to provide an additional measure of conservatism. Selected numeric input values are conservative to assure conservative postulated dose. Except as otherwise required by regulatory guidance, analyses use Technical Specification values where indicated.

Implementation of AST is a significant change to the Salem design basis. PSEG has re-analyzed the design-basis accidents included with this proposal, and the assumptions and methods used are compatible with the requested AST and TEDE criteria. Implementation of AST included in future requests will require re-analysis to employ AST and TEDE criteria in the affected design basis transients.

Meteorological data collected in accordance with the site-specific meteorological measurements program described in the Salem UFSAR were used in generating accident (χ/Q) values. PSEG applied this data using atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room, licensed by the NRC, to perform the radiological analyses identified by RG 1.183.

Atmospheric Dispersion Factors

Control room atmospheric dispersion factors (χ/Qs) for plant vent releases were previously calculated using the ARCON96 computer code (Ref. 23). The dispersion factors were addressed in PSEG's submittal associated with Amendments 251 for Unit 1 and 232 for Unit 2. Atmospheric dispersion factors for the non-LOCA releases through the Main Steam Safety Relief Valves (MSSVs) and pressure relief panels are established in Reference 22 using the ARCON96 computer code and guidance in Regulatory Guide 1.194. The newly established χ/Qs were used to analyze the dose consequences of non-LOCA design basis accidents.

Accident Source Term

Reactor-core fission-product inventory available for release to containment as postulated in this request assumes maximum full power operation, as licensed. PSEG calculates core power as the current licensed rated thermal power of 3459 MWt times 1.05, yielding 3632 MWt, which includes the 2% instrumentation uncertainty and a small margin for power uprate. The assumed period of irradiation is sufficient to allow dose-significant radionuclide activity to reach equilibrium or to reach maximum values.The PSEG calculations (Reference 6 and 9 through 12) provide details of the LOCA and Non-LOCA design basis accident analyses performed according to guidelines set forth in RG 1.183.

Dose Calculations

PSEG completed dose calculations using the proposed AST and TEDE acceptance criteria. These calculations assume that a person located at, or beyond, the exclusion area boundary (EAB) will receive a dose that is the sum of committed-effective-dose-equivalent (CEDE) from inhalation and the deep-dose-equivalent (DDE) from external exposure. They consider radionuclides, including progeny from decay of parent radionuclides that are significant to dose consequence and released radioactivity.

Analyses consider the radionuclides compositions listed in Table 5, § 3.4, cf RG 1.183, and assume that design-basis-accidents release fission-products to containment in particulate form, except for elemental iodine, organic iodine, and noble gases. Radioiodine fractions released to containment in a postulated accident are assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodine, including both gap releases and fuel pellets releases.

3.0 PROPOSED CHANGES

PSEG is voluntarily applying for NRC approval to replace the current source term with the alternative source term described. The alternate source term proposed in this request adheres to NRC guidance provided by Regulatory Guide (RG) 1.183 (Reference 3) except as otherwise stated. Under previous licensing actions, PSEG submitted the selective application of Alternate Source Term licensing changes as they apply to the Fuel Handling Accident.

This amendment request would permit changes to the Technical Specifications and the Licensing Basis as described in the Salem UFSAR. These changes are:

- Implementation of revised AST in accordance with RG 1.183 for Salem Units 1 and 2,
- Crediting recirculation sprays following the LBLOCA for long term containment iodine removal.
- Adjusting the current Control Room assumed in-leakage by replacing it with values based upon the Tracer Gas Test completed in June 2003.
- Modify the current restrictive operational limits by allowing increased ESF leakage outside containment to 1 gpm from the current 1 gph,
- Relocating, from the Technical Specifications to the UFSAR, the Auxiliary Building Filtration System surveillances. Even though the LOCA calculations performed do not credit the ABVS filtration system, PSEG proposes to relocate the charcoal/HEPA filtration surveillances to the UFSAR in order to maintain compliance with the occupational exposure guidelines of 10CFR 20, and
- The removal of the TS requirements, including the existing 24-hour LCO, for the ABVS filtration system. This change is justified under the 10 CFR 50.36 criteria used in determining whether a limiting condition for operation (LCO) is required to be included in the Technical Specifications.

- Modifying the Technical Specifications requirements for the Auxiliary Building Ventilation System. The function of the system is described in the proposed Bases. The Applicability has been expanded to "At All Times" to provide consistency with the accident analyses assumptions. The slightly negative pressure requirement ensures that out leakage from the Auxiliary Building following a DBA LOCA is minimized. The present value of –0.10 inches water gauge is not changed from the current Technical Specifications Bases Page B 3/4 7-5c.

4.0 RADIOLOGICAL CONSEQUENCES

Introduction

The Salem Units 1 and 2 licensing basis for the radiological consequences analyses for Chapter 15 of the UFSAR is currently based on methodologies and assumptions that are derived from TID-14844 (Reference 1) and other early guidance.

Regulatory Guide (RG) 1.183 (Reference 3) provides guidance on application of alternative source terms (AST) in revising the accident source terms used in design basis radiological consequences analyses, as allowed by 10CFR50.67. The alternative source term methodology as established in RG 1.183 is being used to calculate the offsite and control room radiological consequences for Salem Units 1 and 2. The following UFSAR Chapter 15 radiological consequences analyses are revised: LBLOCA, Steam Generator Tube Rupture (SGTR), Locked Rotor, Rod Ejection, Main Steamline Break (MSLB), Waste Gas Decay Tank (WGDT) Rupture, and Volume Control Tank (VCT) Rupture. Each accident and the specific input assumptions are described in detail in PSEG calculations of this report. The accidents are analyzed using the RADTRAD 3.02 computer code (Reference 21) and accident-specific nuclide inventory files (NIFs). A summary of the resulting doses is included on page 59 of Attachment 1.

Common Analysis Inputs and Assumptions

The assumptions and inputs described in this section are common to analyses discussed in this report. The accident specific inputs and assumptions are discussed below.

The TEDE offsite dose acceptance criteria specified in Table 6 of Regulatory Guide 1.183 is implemented in lieu of the whole body and thyroid dose guidelines provided in 10 CFR 100.11. Also, the 5-rem TEDE control room dose acceptance criterion specified in 10 CFR 50.67 is implemented in lieu of the 5-rem whole body and equivalent organ dose guidelines provided in 10 CFR 50 Appendix A GDC 19.

The total effective dose equivalent (TEDE) doses are determined at the exclusion area boundary (EAB) for the limiting two-hour interval and at the low population zone (LPZ) and to control room personnel (CR) for the duration of the event. The dose conversion factors (DCFs) used in determining the committed effective dose equivalent (CEDE) or inhalation doses are from Reference 19 and the deep dose equivalent (DDE) from external exposure are from Reference 20. The TEDE dose is equivalent to the CEDE dose plus DDE for the duration of exposure.

Parameters modeled in the control room personnel dose calculations include CR air intake χ/Qs, normal operation flowrates, emergency operation intake and recirculation flowrates, unfiltered inleakage, control room envelope volume, filter efficiencies, and control room operator breathing rates and occupancies. In the analyses presented in this report, the control room is modeled as a discrete volume.

The Salem current licensed thermal power level of 3,459 MW_t is multiplied by 1.05 to generate an AST core inventory at a thermal power level of 3,632 MW_t (= 3,459 x 1.05), thereby providing a 2% margin for power measuring instrument uncertainty and additional margin for a future power uprate.

Regulatory Guide 1.183 provides guidance for evaluating the radiological consequences of DBAs. The following DBAs were analyzed:

- Large Break Loss-of-Coolant Accident (LBLOCA)
- Fuel Handling Accident (previously addressed in PSEG's submittal associated with Amendment 251 for Unit 1 and Amendment 232 for Unit 2)
- Steam Generator Tube Rupture (SGTR),
- Reactor Coolant Pump Locked Rotor,
- Rod Ejection,
- Main Steam-Line Break (MSLB),
- Waste Gas Decay Tank (WGDT) Rupture, and
- Volume Control Tank (VCT) Rupture.

Each of these accidents and their specific design inputs and assumptions are described in detail in the various calculations (References 6, and 9 through 12). A summary report of these calculations is provided below.

In addition to the above-described calculations, PSEG has completed additional assessments to determine the total effect to the safe operation of the facility and acceptable doses to plant staff and members of the public. The assessments and the results are summarized below:

Emergency Planning Effectiveness Review

An effectiveness review was conducted in accordance with the requirements of 10 CFR 50.54 (q) to assess if the proposed changes represent a decrease in the effectiveness of the PSEG Nuclear Emergency Plan. This evaluation concluded that the effectiveness of the Emergency Plan is not affected by these changes.

Radiological Impact to EQ program

The post-LOCA 120-day gamma dose for the safety related electrical components exposed to the post-LOCA sump water in the recirculation piping in the auxiliary building is expected to increase due to the increase cesium (30% vs. 1%) in the AST. However, the existing conservatism in the Salem TID source term for the post-LOCA sump water compensates for the increase in the 120-day integrated gamma dose due to the increased cesium in the AST. The 120-day post-LOCA gamma integrated doses in the Salem EQ program for the safety related equipment exposed to the recirculation piping remain bounding for the increased cesium in the AST. Therefore, the qualification status of the equipment in the current Salem EQ Program will not be affected. (Reference 14)

Post-LOCA Access to Vital Areas

Regulatory Position 4.4 of Regulatory Guide 1.183 identifies that with respect to the acceptance criteria for the various NUREG-0737 items, for facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii). Therefore, PSEG Nuclear has assessed the impact of AST implementation on post-LOCA access to vital areas at Salem Generating Station (Reference 8). Post-LOCA doses to perform the post-LOCA emergency operating procedure (EOP) functions have been analyzed. The TEDE doses are generally higher in the Auxiliary Building due to higher assumed ESF leakage resulting in a higher inhaled dose contribution and the higher AST-based Cs137 levels have affected whole body dose contributions especially those associated with obtaining and analyzing reactor coolant and containment sump samples.

The requirements to have and maintain post-accident sampling systems were eliminated at Salem Units 1 and 2 by Amendment Nos. 254 and 235, respectively. In support of the development contingency plans for obtaining and analyzing samples of reactor coolant, containment sump, and containment atmosphere, PSEG Nuclear evaluated its capability to obtain and analyze sample in a reasonable amount of time. With implementation of AST, in some limiting cases of obtaining post-LOCA chemistry samples, the higher doses have affected the amount of time required for obtaining and analyzing samples. The assessment results are consistent with the elimination of the post-accident sampling systems from the Salem licensing basis. The resulting doses remain in compliance with NUREG-0737, Section II.B.2, and would not prevent operators from accessing areas of the plant to perform necessary functions without exceeding 5 rem (TEDE is used in the AST analysis).

Salem Technical Support Center Habitability

An evaluation was completed to determine the post-LOCA radiological habitability of the Salem Technical Support Center (TSC) using an alternative source term and TEDE dose criteria.

Supplement 1 to NUREG 0737, Section 8.2, "TSC", states that the TSC will be provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Therefore, a 5 rem TEDE 30-day dose guideline is applied to the TSC. The TSC dose is based on the control room breathing rate and occupancy factors specified in RG 1.183, Section 4.2.6. (Reference 7).

The post-LOCA dose at the TSC is 3.11 rem TEDE, which is within the 5-rem TEDE allowable limit.

Hope Creek Station Control Room Habitability

PSEG evaluated the Hope Creek control room habitability due to a LOCA occurring at the Salem Station. Unit 2, which has been determined to be the most limiting for radiological impact on the adjacent Hope Creek plant control room habitability due to being in closer proximity to the Hope Creek station. (Reference 13)

The post-LOCA results indicate that the Hope Creek CR is habitable during and following a LOCA occurring at Salem Unit 2. Post-LOCA Hope Creek CR dose due to a LOCA occurring at the Salem 2 is 0.79 rem TEDE, which is within the 5-rem TEDE allowable limit.

4.1 LARGE BREAK LOSS OF COOLANT ACCIDENT RADIOLOGICAL ANALYSIS

An abrupt failure of the main reactor coolant pipe is assumed to occur and it is assumed that the emergency core cooling features fail to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity from the core is released to the containment and from there released to the environment by means of containment leakage and leakage from the emergency core cooling system.

The reanalysis of the LBLOCA offsite and control room doses for Salem uses the following RG 1.183 source term characteristics in place of those identified in TID-14844 and Regulatory Guide 1.4:

- Iodine chemical species
- Fission product release timing
- Fission product release phases through early in-vessel
- Fission product release fractions
- Fission product groups

PURPOSE

The purpose of this calculation is to determine the post-Loss Of Coolant Accident (LOCA) doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) using the Alternative Source Term (AST), guidance in the Regulatory Guide (RG) 1.183, RADTRAD3.02 computer code, and Total Effective Dose Equivalent (TEDE) dose criteria for the following post-LOCA release paths:

- Containment Leakage
- Engineered Safeguards Feature (ESF) Leakage
- Containment Pressure-Vacuum Relief Line Release
- Back-leakage to the Refueling Water Storage Tank (RWST)

ANALYTICAL APPROACH

Core Inventory

The Salem 1 & 2 core inventory used in the LOCA radiological consequence analysis consisted solely of noble gas and halogen isotopes. The AST analysis requires radionuclide inventories for the additional radionuclide groups shown in Table 5 of Regulatory Guide 1.183. Therefore, a post-LOCA core inventory required for the AST analysis is developed.

Containment Leakage

Activity Transport in Primary Containment

Per RG 1.183, Appendix A, Section 3.3, the mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour. The mixing rate between the sprayed and unsprayed regions is assumed to be two turnovers of the unsprayed regions per hour, which is 21,833 cfm. This mixing rate is conservatively less than the post-LOCA airflow of 43,251 cfm based the operation of two of five CFCUs.

Containment Spray Coverage

Westinghouse Report, WCAP-7952, "Iodine Removal by Spray in the Salem Containment," dated, August 1972 indicates that the spray coverage for Salem was originally 90%. In 1996 a conservative spray coverage of 60% was derived by performing a volumetric depiction of spray coverage in the containment. The spray pattern was taken from the nozzle manufacturer's data at ambient atmospheric conditions. The coverage was then compressed based on a compression multiplier at conservative containment temperature and pressure conditions after the LOCA.

With the new AST methodology the timing of the release of fission products occurs later after the LOCA compared to the TID-14844 methodology. With AST the iodine release is not complete until 1.8 hours after the accident. With this timing change from TID-14844 to AST, the decreasing containment temperature and pressure increases the spray effectiveness areas. Additionally, the use of a conservative volumetric depiction of spray coverage does not recognize the extensive turbulence in the sprayed and unsprayed air interaction process as was done in the earlier WCAP, which considered coverage to be 100% in the spray drop zone regions where there were no equipment interferences. Some space in the top of the containment dome (above the spray headers) was considered unsprayed in the 1996 calculation, but was considered to be sprayed in the Westinghouse report. The sequence of events with AST methodology makes it reasonable to assume that the fission products released through the depressurization of the RCS during a LOCA will be scrubbed through the sprayed region before it migrates to the top of the containment dome. After the rapid initial containment pressure transient has peaked, the only flows into the dome volume are gases replacing steam condensing in the dome or gases transferred there due to turbulence between the dome and lower volumes. Regardless, when the significant fission products begin to enter the lower region of containment from the LOCA the containment pressure has already peaked and the sprays are already operating. It is reasonably conservative to assume the fission products do not reach the dome region without passing through spray.

Therefore for this AST analysis the containment spray coverage was reconstituted using greater spray coverage in the open sprayed sub-volumes below the spray headers and above the operating floor and in the sub-volumes above the reach of the upward directed sprays at the top of the containment. The calculated spray coverage increased to 80.2% and for conservatism 75% was used in these analyses.

Containment spray removal of iodine and particulates (aerosols) is assumed to be initiated at 90 seconds after the start of the LOCA event. In the initial containment spray injection phase, the spray water is drawn from the refueling water storage tank (RWST). The injection phase terminates at 48 minutes after the start of the LOCA, at which time the containment spray recirculation phase begins and the spray water is drawn from the containment sump. A containment spray interruption of less than 5 minutes is expected during the transition from the injection to the recirculation phase. This containment spray interruption is conservatively assumed to be 10 minutes in this analysis (from 48 to 58 minutes). During the interruption, the containment spray is not credited in removing the aerosol and elemental iodine activities from the containment atmosphere. The containment spray injection phase elemental iodine removal coefficient is calculated to be 29 hr^{-1} using Salem plant-specific design condition parameters.

However, Standard Review Plan (SRP) 6.5.2 limits the use of the elemental iodine removal coefficient (λ_E) to 20 hr^{-1}. Therefore, a λ_E of 20 hr^{-1} is used during the injection phase. The spray aerosol removal coefficient (λ_P) in the injection phase is calculated to be 4.44 hr^{-1} using the plant-specific design input information. During the injection phase, one containment spray pump (CSP) is assumed to operate (postulating a single failure of the other pump) at a minimum flow rate of 2,600 gpm. The flow rate is reduced to 1,900 gpm during the recirculation phase. Therefore, for the recirculation phase λ_E and λ_P are re-calculated based on the reduced flow rate.

SRP 6.5.2, Section 4.c sets forth a maximum decontamination factor (DF) of 200 for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. Regulatory Guide 1.183 specifies that the maximum activity to be used in determining the containment spray DF is defined as the iodine activity in the columns labeled "Total" in Table 2 of RG 1.183 multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosols are treated as particulates in SRP methodology).

The DF for the containment atmosphere achieved by the containment spray system can be determined using the following equation:

$$DF = 1 + \frac{V_s H}{V_c}$$

Where: V_s = volume of liquid in containment sump

V_c = containment net free volume less V_s

H = partition coefficient

This equation is used to calculate the minimum DF of 546 for the Salem Fuel Upgrade/Margin Recovery (FU/MR) project. The DF of 100 (which is smaller than both 200 and 546) is conservatively used in the containment leakage analysis.

The SRP 6.5.2 also states that the particulate iodine removal rate should be reduced by a factor of 10 when a particulate DF of 50 is reached. The maximum iodine activity in the containment atmosphere occurs at the end of early-in-vessel release phase, which is 1.8 hrs after a LOCA. Therefore, the corresponding λ_E and λ_P cutoff times are calculated based on an assumption that the maximum iodine activity occurs at 1.8 hrs after onset of a LOCA. Although containment spray can be operated for a long time after the LOCA, containment spray operation is assumed to be terminated at 4.0 hrs after a LOCA. The corresponding assumptions and design inputs are shown in Reference 6. The resulting containment leakage doses are summed with the doses from other post-LOCA sources.

Long-Term Iodine Partition

RG 1.183, Appendix A, Section 1, requires evaluation of the re-evolution of iodine for a sump pH value of less than 7. During the injection and recirculation phases, the sump water pH will remain at greater than 7 including consideration of the effect of acids and bases created during the LOCA event and radiolysis products of electrical products.. Consequently, the re-evolution of dissolved iodine from the sump is not considered.

Engineered Safeguards Feature (ESF) System Leakage

ESF systems that recirculate containment sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands; pump shaft seals, flanged connections, and other similar components in the Residual Heat Removal (RHR), Safety Injection (SI), and Chemical & Volume Control (CVC) systems. The radiological consequences from the postulated leakage are analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Since the containment spray (CS) system operation is extended during the recirculation phase, it is also accounted for in the total ESF leakage. The recirculation phase system components subject to leakage during the recirculation phase are located in the Auxiliary Building. However, Auxiliary Building filtration is not credited in this AST dose analysis.

Post-LOCA Iodine Source Term in Sump Water

Regulatory Guide 1.183, Appendix A, Section 5.1, requires that with the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to instantaneously and homogeneously mix in the primary containment sump water. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. The sump water activity in this analysis is established in the following section based on a suitably conservative iodine transport of containment airborne activity to the sump, which is permitted in the RG 1.183, Section 5.1. The requirements of Section 5.1 of RG 1.183, Appendix A for a suitably conservative iodine transport model is adopted in this AST dose analysis and justified in the following section:

Forty percent of iodine released from the fuel to the containment is assumed to be transported to the containment sump. If the sump pH is controlled at values 7 or greater, the radioiodine released from the fuel to containment should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The containment sump pH is maintained at greater than 7 during and following a LOCA at the Salem plants. With the exception of elemental and organic iodine and noble gases, *fission products should be assumed to be in particulate form.* Based on these regulatory requirements, the chemical form of sump water iodine could be 4.85% elemental iodine, which is volatile and subject to becoming airborne from the ESF leakage in proportion to a flashing rate, and 95% cesium iodide (CsI), which remains waterborne provided the pH of the containment sump water is > 7. The remaining 0.15% organic iodide is in gaseous form is released to the containment atmosphere, and has its dose contribution accounted for in the containment leakage.

ESF Leakage Release Path

The ESF leakage during the post-LOCA injection and recirculation phases includes leakage through valve packing glands; pump shaft seals, flanged connections, and other similar components associated with the containment spray pumps, charging pumps, safety injection pumps, and residual heat removal (RHR) pumps. These release points are located in the Auxiliary Building. An ESF leakage rate of 1 gpm is assumed in the analysis. The mixing of ESF leakage in the Auxiliary Building air spaces and dilution with other exhaust air flows in the plant vent is not credited in this analysis.

ESF Leakage Dose Analysis

The leakage from systems outside containment that could contain highly radioactive fluids after a LOCA, e.g. ECCS, is engineered safeguards feature (ESF) leakage. The value used in the LOCA dose analysis is doubled to be consistent with the guidance provided in Regulatory Guide 1.183. The nominal ESF leakage value used in this AST analysis is conservatively increased from the present value of ~1 gph to 1 gpm (from 3790 cc/hr to 1 gpm). This increase in allowable ESF leakage provides a more reasonable operating margin. The normal operation of the positive displacement charging pump (PDP) enhances reliability of the safety related accident mitigating charging pumps by not relying on these pumps during normal operation. This conservatively higher value of ESF leakage significantly contributes to the calculated control room dose yet dose consequences are still maintained within the regulatory limits. The offsite doses are a small fraction of the limits. The higher allowable ESF leakage limits will not change the Salem 1 & 2 operational strategy to maintain ESF leakage as low as reasonably achievable.

ESF system leakage is conservatively assumed to begin at 20 minutes, representing the earliest possible recirculation start time with the ESF operating at maximum capacity. The design basis ESF leakage rate of 3790 cc/hr (= 1 gal/hr) is increased to 1 gpm. Per the guidance in Regulatory Guide 1.183, the ESF leakage of 1 gpm is doubled and an iodine flashing fraction of 10 percent is applied resulting in an equivalent ESF leakage rate to the environment of 0.2 gpm (= 0.02673 cfm).

It is determined that the 4.85% of elemental iodine released from the core to containment atmosphere and transported to the containment sump becomes the iodine source for the ESF leakage. The elemental iodine of 4.85% is increased by a factor of 3 to add a large conservatism in the resulting ESF leakage doses. The resulting elemental iodine represents a 0.05824 fraction of the core iodine inventory. Therefore, the RADTRAD model is developed for the ESF leakage analysis using the RADTRAD release fraction and timing (RFT) file with an iodine release fraction of 0.05824 and applicable design inputs and assumptions. A sump water volume of 43,930 ft^3 is conservatively used, which is less than the sump water volume of 45,510 ft^3 used in the sump pH analysis. The resulting doses are summed with the doses from other post-LOCA sources in Reference 6.

Containment Pressure-Vacuum Relief Line Release

The containment pressure-vacuum relief line release occurs following the large break LOCA, and before containment isolation. The entire reactor coolant system (RCS) inventory is assumed to be instantaneously released and homogeneously mixed in the containment atmosphere. The containment pressurization due to the RCS mass and energy release combined with the presence of the containment pressure-vacuum relief line result in the potential for a limited release of airborne activity to the environment. One hundred percent of the radionuclide inventory in the RCS liquid is assumed to be released into the containment at the initiation of the LOCA. A release of gap activity into the containment is not considered since the containment pressure-vacuum relief line release duration of 5 seconds terminates this release path prior to the onset of the gap release phase at 30 seconds (Reference 3, Table 4).

The RCS iodine activity is based on the technical specification for RCS equilibrium activity. Iodine spikes are not considered. Per Technical Specification (TS) LCO for Specific Activity, the primary coolant equilibrium iodine concentration permitted by the technical specifications is 1 μCi/gm Dose Equivalent (DE) I-131. Technical Specification Section 1.10 defines DE I-131 as that concentration of I-131 (in μCi/g)

(DE) I-131. Technical Specification Section 1.10 defines DE I-131 as that concentration of I-131 (in μCi/g) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present using the thyroid dose conversion factors (DCFs) specified in Table III of TID-14844; the TS Section 1.10 definition of DE I-131 will be revised to be based on the DCFs specified in Federal Guidance Report 11 (Reference 19). Therefore, this analysis calculates DE I-131 in terms of the thyroid dose conversion factors specified in FGR 11.

The plant-specific RCS iodine concentrations corresponding to 1% fuel defects are converted to 1 μCi/g DE I-131. The total isotopic iodine activity in the RCS is conservatively calculated using the RCS mass. A 1 percent failed fuel primary coolant iodine concentration of 3.58 μCi/gm DE I-131, which is conservatively greater than the TS limit of 1.0 μCi/gm DE I-131, is used to establish iodine scaling factor. Since 1 percent failed fuel introduces more iodine activity into the primary coolant than is allowed by the Tech Specs, it can be conservatively assumed that 1 percent failed fuel introduces more non-iodine activity into the RCS than is allowed by the Tech Spec limit of 100/E-bar. Therefore, the RCS noble gas concentrations corresponding to 1% fuel defects are conservatively used to determine the total RCS noble gas activity released in the containment based on the total coolant mass in the RCS. The isotopic activity is divided by the core thermal power level of 3,632 MW_t in to obtain the isotopic Ci/MW_t, which is used to develop the RADTRAD NIF SPURGE_def.txt. The CR is assumed to be in a normal mode operation during the release. The resulting doses are summed with the doses from other post-LOCA sources.

Post-LOCA Back-Leakage to the Refueling Water Storage Tank (RWST)

The Salem 1 & 2 RWSTs are located west of the containment buildings. The shortest distance of 76.74 meter between the centerlines of the Unit 2 RWST and the Unit 2 CR air intake is more than double the distance of 30.25 m between the centerlines of the plant vent and the CR air intake. The χ/Qs for the RWST release would be proportionately lower than those for the containment and ESF leakage releases via the plant vent. The worst case back-leakage to the RWST is estimated to be 100 cc/hr (5.886E-05 cfm), which is negligibly small. The negligibly small amount of back-leakage is further diluted into a large RWST air space volume of 35,806 ft^3 before it is vented to the atmosphere and carried to the CR air intake by smaller χ/Qs. The post-LOCA back-leakage to the RWST will result in insignificantly small dose consequences; therefore, its dose contribution is not analyzed.

ASSUMPTIONS

Regulatory Guide 1.183, Appendix A, provides guidance on modeling assumptions that are acceptable to the NRC staff for the evaluation of the radiological consequences of a LOCA. The following sections address the applicability of these modeling assumptions to this Salem Units 1 and 2 LOCA analyses. These assumptions are incorporated as design inputs in Section 5.0 of Reference 6 and are incorporated in these analyses.

Source Term Assumptions

Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Guide 1.183 as follows:

Core Inventory

The assumed inventory of fission products in the reactor core and available for release to the containment is based on the maximum power level of 3,632 MWt corresponding to current fuel enrichment and fuel burnup, which is 1.05 times the Salem current licensed thermal power of 3,459 MW_t.

Release Fractions and Timing

The core inventory release fractions, by radionuclide groups, for the gap release and early-in-vessel release for a Design Basis Accident (DBA) LOCA are listed in Table 2 of Reference 3. These fractions are applied to the equilibrium core inventory using the release timing specified in Table 4 of Reference 3.

Radionuclide Composition

The elements in each radionuclide group to be considered in the LOCA design basis analyses are shown in Table 5 of Reference 3.

Chemical Form

The containment sump water pH is greater than 7 during and following a LOCA. Consequently, the chemical forms of radioiodine released to the containment can be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form

Activity Transport in Primary Containment

The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment).
Reduction in airborne radioactivity in the containment by natural deposition within the containment is credited using the RADTRAD3.02 Powers model for aerosol removal coefficient with a 10-percentile probability.

Reduction in airborne radioactivity in the containment by the containment spray system is credited. The Salem plant-specific containment spray coverage analysis indicates that the containment spray covers

75% of the containment volume. A mixing rate of 2 turnovers of the unsprayed region volumes per hour, which is equivalent to 21,833 cfm, is conservatively used in the containment leakage analysis.

The SRP 6.5.2 sets forth a maximum decontamination factor (DF) of 200 for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Table 2 of Reference 3 multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine. The minimum DF of 546 was calculated for the fuel upgrade/margin recovery project. However, a DF of 100 is conservatively used in the analysis for the containment leakage analysis.

The SRP 6.5.2 states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The cutoff times for the elemental and particulate removal coefficients are calculated based on an assumption that the maximum iodine activity occurs at 1.8 hours after onset of a LOCA.

The primary containment is assumed to leak at the allowable Technical Specification peak pressure leak rate of 0.1% by weight for the first 24 hours. This leak rate is reduced to 0.05% by weight after 24 hours.

The containment pressure-vacuum relief release following the large break LOCA prior to containment isolation is analyzed using the 100% RCS inventory based on technical specification RCS equilibrium activity and resulting doses are summed with the postulated doses from the other release paths in Reference 6.

Assumptions on ESF System Leakage

ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems. The radiological consequences from the postulated leakage are analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for a PWR, and are incorporated in the Design Inputs:

Regulatory Guide 1.183 (Appendix A, Section 5.1) states that with the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to instantaneously and homogeneously mix in the primary containment sump water at the time of release from the core. Regulatory Guide 1.183 (Appendix A, Section 5.1) also states that in lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.

The ESF leakage is taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. The ESF leakage of 1 gpm is doubled. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems, which is 20 minutes, and end at the latest time the releases from these systems are terminated, which is 720 hours.

Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank. For Salem Units 1 and 2, the dose contribution from the RWST leakage is insignificantly small, therefore, it is not analyzed.

With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.

Since the temperature of the ESF leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne is assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, is determined using a constant enthalpy, h, process, to be no more than 10% based on the maximum time-dependent temperature of the sump water circulating outside the containment. The FF of 10% is used for the entire duration of the ESF leakage.

The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, is not credited. The Auxiliary Building Ventilation System (ABVS) filtration is not credited. The ESF leakage is not assumed to be released to the environment with mixing with the Auxiliary Building volume.

Offsite Dose Consequences

Regulatory Guide 1.183 (Ref. 3, Section 4.1) provides guidance to be used in determining the total effective dose equivalent (TEDE) for persons located at the exclusion area boundary (EAB) and at the outer boundary of the low population zone (LPZ). The following sections address the applicability of this guidance to the Salem Units 1 and 2 LOCA and non-LOCA analyses. These assumptions are incorporated as design inputs in Section 5.0 of applicable analyses. The assumptions for the offsite dose consequences are common to all DBA analyses, therefore, they are not repeated in the remaining DBA analyses.

Consistent with RG 1.183, this dose calculation determines the TEDE. The TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE considers all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.

Consistent with RG 1.183, the exposure-to-CEDE factors for inhalation of radioactive material are derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers". This calculation models the CEDE dose conversion factors (DCFs) in the column headed "effective" yield doses in Table 2.1 of Federal Guidance Report 11 (Ref. 19), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion".

Consistent with RG 1.183, for the first 8 hours, the breathing rate of persons offsite is assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate is assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate is assumed to be 2.3×10^{-4} cubic meters per second.

Consistent with RG 1.183, the DDE is calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE is used in lieu of DDE in determining the contribution of external dose to the TEDE. This calculation models the EDE dose conversion factors in the column headed "effective" in Table III.1 of Federal Guidance Report 12 (Ref. 20), "External Exposure to Radionuclides in Air, Water, and Soil".

Consistent with RG 1.183, the TEDE is determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is determined and used in determining compliance with the dose criteria in 10 CFR 50.67. For the LOCA the postulated EAB doses should not exceed the criteria established in RG 1.183 Table 6. This assumption is incorporated as a design input.

EAB Dose Acceptance Criterion: 25 rem TEDE

The RADTRAD Code (Ref. 21) used in this analysis determines the maximum two-hour TEDE by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The time increments appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release.

Consistent with RG 1.183, the TEDE is determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and is used in determining compliance with the dose criteria in 10 CFR 50.67. For the LOCA the postulated LPZ doses should not exceed the criteria established in RG 1.183 Table 6. This assumption is incorporated as a design input.

LPZ Dose Acceptance Criterion: *25 rem TEDE*

Consistent with RG 1.183 no correction is made for depletion of the effluent plume by deposition on the ground.

Control Room Dose Consequences

The following sections for the CR dose consequences are common to all DBA analyses; therefore, they are not repeated in the remaining DBA analyses.

CR Airborne Doses from Filtered and Unfiltered Inleakage

Since the radioactive material in the areas and structures adjacent to the CR envelope is assumed to be the same as that in the post-LOCA radioactive plume released from the facility, they collectively represent the CR airborne doses from the filtered intake and unfiltered inleakage from the various post-LOCA sources. The CR airborne doses from these various post-LOCA release paths and the resulting doses are summarized in Reference 6.

Radiation Shine from External Radioactive Plume

The post-LOCA radioactive plume released from the plant vent contains the radioactive sources from the containment leakage, ESF leakage, and the containment pressure-vacuum relief line release. The CR envelope, which has two feet of concrete bulk shielding, is submerged within the post-LOCA radioactive plume and the CR operator is exposed to gamma radiation through the walls and roof. The RADTRAD3.02 code calculates the site boundary whole-body gamma dose based on semi-infinite cloud immersion. Therefore, the χ/Qs for the LPZ receptor modeled in RADTRAD files S150CS75CL01.psf and S75ESF1GPM are modified by replacing them with the newly developed χ/Qs for the CR center location. The semi-infinite gamma dose is calculated at the center of CR at the roof elevation. The ARCON96 code is used to calculate the new set of χ/Qs for the CR center location.

The containment pressure-vacuum relief dose contribution is extremely small; therefore, it is excluded from this analysis. The total whole body gamma dose is calculated to be 4.33 rem. This is a semi-infinite dose above the center of the CR on the roof. This gamma dose is attenuated by the 2-foot thick concrete roof above the CR and 10 feet of air. The energy dependent gamma dose attenuation factors are calculated. The gamma attenuation factors are substantially low for the low energy bins. The average gamma dose attenuation factor of 0.0136 is conservatively used, which is the average for the five mean energies in the gamma energy range of 0.85 Mev through 2.75 Mev. The resulting gamma dose from the external cloud shine dose to a CR operator would be 0.059 rem, which is added to the dose contributions from other post-LOCA sources.

Radiation Shine From Radioactive Material In The Reactor Containment

It is only the post-LOCA containment leakage activity confined in the containment dome air space above the operating floor that can conceivably contribute to the direct shine dose to the CR operator (lower locations within the containment building are heavily self-shielded by the reactor cavity and steam generator enclosure structures). The containment cylindrical concrete wall is 4'-6" thick below the spring line, and 3'-6" thick above the spring line. The CR outside concrete wall is 2'-0" thick. The concrete shielding of at least 5'-6" between the post-LOCA radioactive source in the containment dome and the CR operating floor provides ample shielding to totally shield the post-LOCA containment shine dose to the CR operator.

Radiation Shine from CR Filter

The CREACS charcoal filter trains are located west of the main CR. The width (2'-6") and height (4'-3") of the charcoal bed are measured. The post-LOCA aerosol buildup on the HEPA filter and the iodine buildup on the charcoal filter are calculated as follows:

Post-LOCA Iodine & Aerosol Activity on CR Charcoal/HEPA Filter- Containment Leakage

The RADTRAD3.02 code calculates the cumulative elemental and organic iodine atoms and the aerosol mass deposited on the CR recirculation charcoal/HEPA filters. The CR intake filter iodine and aerosol activities are calculated for the containment leakage. The relationship between the aerosol mass and activity is established based on the information obtained from RADTRAD run S150CS75CL01.o0. The aerosol mass deposited on the CR HEPA recirculation filter is calculated by the RADTRAD code for the duration of the accident. Knowing the CR intake and recirculation filtration flow rates, the relationship can be established to calculate the aerosol mass deposited on the HEPA filter. The total aerosol mass deposited on the CR HEPA filter due to the containment leakage is used with the aerosol mass/activity relationship established to calculate the aerosol isotopic activities deposited on the CR HEPA filter. The iodine atoms are converted into the isotopic iodine activities using the atom/activity relation established.

Post-LOCA Iodine & Aerosol Activity on CR Charcoal/HEPA Filter – ESF Leakage

Similarly, the iodine and aerosol deposited on the CR charcoal/HEPA filter are calculated for the post-LOCA ESF leakage. The post-LOCA ESF leakage consists of a non-aerosol iodine release (97% of elemental iodine + 3% of organic iodine), therefore, there is no aerosol mass deposited on the CR HEPA filter (S75ESF1GPM.o0, CR Compartment Nuclide Inventory @ 720 hrs). The iodine atoms are converted into the isotopic iodine activities using the atom/activity relationship. The iodine isotopic activities deposited on the CR charcoal/HEPA filter due to the containment and ESF leakages are shown in Reference 6.

MicroShield Analysis of CR Charcoal/HEPA Filter Shine

The total charcoal/HEPA filter aerosol and iodine isotopic activities are input into the MicroShield Computer RunSCREAS.MS5 with the source geometry, dimension, and detector location to compute the direct dose rate from the CR filter. Due to the limitations of the MicroShield code, which calculates the dose rate at the dose point location within the projected area, the dose point location at the center of the charcoal filter projected area is conservatively modeled because the actual dose point involves the larger direct distance in the air and slant distance in the concrete shielding. The 720-hrs direct dose from the CR filter shine is calculated using the CR occupancy factors and added to doses from other post-LOCA sources.

As stated in the 180-day response to Generic Letter 2003-01, PSEG is submitting this request for amendment to revise the dose analysis for Salem Units 1 and 2 to bring the as-built plant in alignment with the dose analysis in regards to Control Room Envelope (CRE) unfiltered inleakage. For Salem Units 1 and 2, tracer gas testing was performed from May 31 to June 4, 2003, to measure the inleakage to the Salem CRE. The Salem Control Room Emergency Air Conditioning System (CREACS) can be aligned to respond to a design basis radiological accidents with either a single train (either Unit 1 or Unit 2 train) operating by itself or with both trains operating to pressurize the control room. The recirculation mode operation of CREACS is for toxic chemical events and smoke events and is not used during a DBA radiological event. Five tests were performed. Three tests were performed in single train alignment of CREACS in its pressurization mode (two with the Unit 1 train operating by itself and one with the Unit 2 train operating by itself), one test was performed with both CREACS trains operating in its pressurization mode, and one test was performed in the recirculation mode. The results of the tracer gas testing identified that two of the four test results for pressurization were below 60 cfm unfiltered inleakage (both tests were Unit 1 single train configuration) and two tests (Unit 2 single train and both trains configuration) measured unfiltered inleakage in the range of 90 to 100 cfm. Thus, a conservative value of 150cfm is used for this analysis.

Regulatory Guide 1.183 provides guidance to be used in determining the total effective dose equivalent (TEDE) for persons located in the control room (CR). The following sections address the applicability of this guidance to the Salem Units 1 and 2 LOCA analyses. These assumptions are incorporated as design inputs in Section 5.0 of Reference 6. The requirements in the following sections for the CR dose consequences are common to all DBA analyses; therefore, they are repeated in the remaining DBA analyses.

Consistent with RG 1.183, the CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel:

- Contamination of the control room atmosphere by the filtered CR ventilation inflow through the CR air intake and by unfiltered inleakage of the radioactive material contained in the post-accident radioactive plume released from the facility,
- Contamination of the control room atmosphere by filtered CR ventilation inflow via the CR air intake and by unfiltered inleakage of airborne radioactive material from areas and structures adjacent to the control room envelope,
- Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud shine dose),
- Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose), and
- Radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in CR intake and recirculation filters [i.e., CR filter shine dose].

Consistent with RG 1.183, the radioactive material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values. These parameters do not result in non-conservative results for the control room.

Consistent with RG 1.183, the models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, are structured to provide suitably conservative estimates of the exposure to control room personnel.

Consistent with RG 1.183, credit for engineered safety features (ESF) that mitigate airborne radioactive material within the control room is assumed. Such features include control room pressurization, and intake and recirculation filtration. CR isolation is actuated by ESF signals and radiation monitors (RMs). Several aspects of CREACS operation can delay the CR isolation. A conservative delay of 1 minute is assumed for the CR isolation to be fully operational.

Consistent with RG 1.183, credit is not taken for the use of personal protective equipment (e.g., protective beta radiation resistant clothing, eye protection, or self-contained breathing apparatus [SCBA]) or prophylactic drugs (i.e., potassium iodide [KI] pills).

Consistent with RG 1.183, the CR dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second. These assumptions are incorporated as design inputs.

Consistent with RG 1.183, the control room doses are calculated using the offsite dose analysis dose conversion factors identified in RG 1.183, Section 4.1. The DDE from photons is corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The RADTRAD Code used in this analysis uses the following expression to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room:

$$DDE_{finite} = (DDE_{\infty} \times V^{0.338}) / 1173$$

Consistent with RG 1.183, for the LOCA the postulated CR doses should not exceed the 5-rem TEDE criterion established in 10 CFR 50.67. This assumption is incorporated as a design input.

CR Dose Acceptance Criterion: 5 rem TEDE

DESIGN INPUTS:

General Considerations

Applicability of Prior Licensing Basis

The implementation of an AST is a significant change to the design basis of the facility and assumptions and design inputs used in the analyses. The characteristics of the AST and the revised TEDE dose calculation methodology may be incompatible with many of the analysis assumptions and methods currently used in the facility's design basis analyses. The Salem specific design inputs and assumptions used in the current analyses were assessed for their validity to represent the as-built condition of the plant and evaluated for their compatibility to meet the AST and TEDE methodology. The analysis in this calculation ensures that analysis assumptions, design inputs, and methods are compatible with the requirements of the AST and the TEDE criteria.

Credit for Engineered Safety Features

Credit is taken only for those accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The flowrate assumed from the sprayed-to-unsprayed regions is less than the flow due to operation of 2 out of 5 containment fan cooler units. A delay of 1 minute in the CR pressurization, and operation of one train of CREACS are conservatively assumed to maximize the resulting doses. The CR intake radiation monitor capability to align with the less contaminated air intake is credited in the analysis.

Assignment of Numeric Input Values

The numeric values that are chosen as inputs to analyses required by 10 CFR 50.67 are compatible to AST and TEDE dose criteria and selected with the objective of maximizing the postulated dose. Use of 10% higher flow rate for the CREACS intake and 10% lower recirculation flow rate, 1 minute delay in the CREACS initiation time, operation of one train of CREACS, and use of ground release χ/Qs demonstrate the inherent conservatisms in the plant design and post-accident response. Most of the design input parameter values used in the analysis are those specified in the Technical Specifications.

Meteorology Considerations

Atmospheric dispersion factors (χ/Qs) for the onsite release points such as the plant vent for containment and ESF leakage release paths are re-established using the NRC sponsored computer code ARCON96. The control room χ/Qs are reconstituted using the SGS plant specific meteorology and appropriate regulatory guidance. The site boundary χ/Qs were accepted by the staff in the previous licensing proceedings.

The requirements in the above design input sections are common for all DBA analyses; therefore they are not repeated in the remaining DBA analyses.

Accident-Specific Design Inputs/Assumptions

The design as-built inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are contained in Section 5.0 of Reference 6. The design inputs are compatible with the requirements of the AST and TEDE dose criteria and the assumptions are consistent with those identified in Appendix A of RG 1.183.

RESULTS SUMMARY

a) Post-LOCA EAB, LPZ, and CR doses at the Salem Nuclear Generating Station without the interruption of containment spray are summarized as follows:

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (rem) Receptor Location		
	Control Room	**EAB**	**LPZ**
Containment Leakage	6.87E-01	2.27E+00 (occurs @ 0.5 hr)	4.32E-01
ESF Leakage	2.79E+00	9.87E-01 (occurs @ 0.4 hr)	7.45E-01
Containment Relief Line Release	3.99E-04	3.34E-05 (occurs @ 0.0 hr)	5.67E-06
Back-leakage to RWST	Negligible	Negligible	Negligible
Containment Shine	Negligible	Negligible	Negligible
External Cloud	5.80E-02	N/A	N/A
CR Filter Shine	2.64E-01	N/A	N/A
Total	**3.80E+00**	**3.26E+00**	**1.18E+00**
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**
	RADTRAD Computer Run No.		
Containment Leakage	S150CS75CL00	S150CS75CL00	S150CS75CL00
ESF Leakage	S75ESF1GPM	S75ESF1GPM	S75ESF1GPM
Cont. Relief Line Release	SPURGE00	SPURGE00	SPURGE00

b) Post-LOCA EAB, LPZ, and CR doses at the Salem Nuclear Generating Station with the 10 minutes interruption of containment spray are summarized as follows:

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (rem) Receptor Location		
	Control Room	**EAB**	**LPZ**
Containment Leakage	7.17E-01	2.48E+00 (occurs @ 0.5 hr)	4.67E-01
ESF Leakage	2.79E+00	9.87E-01 (occurs @ 0.4 hr)	7.45E-01
Containment Relief Line Release	3.99E-04	3.34E-05 (occurs @ 0.0 hr)	5.67E-06
Back-leakage to RWST	Negligible	Negligible	Negligible
Containment Shine	Negligible	Negligible	Negligible
External Cloud	5.90E-02	N/A	N/A
CR Filter Shine	2.64E-01	N/A	N/A
Total	**3.83E+00**	**3.47E+00**	**1.21E+00**
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**
	RADTRAD Computer Run No.		
Containment Leakage	S150CS75CL01	S150CS75CL01	S150CS75CL01
ESF Leakage	S75ESF1GPM	S75ESF1GPM	S75ESF1GPM
Cont. Relief Line Release	SPURGE00	SPURGE00	SPURGE00

CONCLUSIONS

The post-LOCA results indicate that the EAB, LPZ, and CR doses due to a LOCA are within their allowable limits.

4.2 STEAM GENERATOR TUBE RUPTURE RADIOLOGICAL ANALYSIS

PURPOSE:

The purpose of this evaluation is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to a Steam Generator Tube Rupture (SGTR) using the Alternative Source Term (AST) methodology and total effective dose equivalent (TEDE) dose criteria. The dose consequences are representative of an accident occurring in either unit. The SGTR analysis is performed using the guidance in Regulatory Guide 1.183 and its Appendix F.

ANALYTICAL APPROACH:

This analysis uses Version 3.02 of the RADTRAD computer code to calculate the potential radiological consequences of the SGTR accident. The RADTRAD code is documented in NUREG/CR-6604.

The calculation assumes that the CR air intake monitors preferentially select the less contaminated air intake when only one CREACS train is available. There are two CREACS trains that each provides emergency filtration and air conditioning services to the combined CR for the Salem 1 & 2 plants. Each CREACS train is safety related and required to be operable by Technical Specifications. Each CREACS train takes outside air supplied through two independent ducts equipped with safety related fans and radiation monitors, which make the CREACS air supply system single failure proof. The redundant air intake monitors preferentially select the less contaminated air intake during an accident condition when the radiation level at normal intake exceeds the setpoint value. Based on the Salem plant-specific CREACS design and performance, the post-accident CREACS response is credited in the analysis with the CR air intake monitor's ability to preferentially select the less contaminated air intake. This CR response is identical for all non-LOCA DBA analyses; therefore, it is not repeated in the remaining non-LOCA DBA analyses.

This calculation addresses the reactor coolant activity concentrations corresponding to (1) a preaccident iodine spike and (2) a concurrent iodine spike. There is no potential fuel failure during this event.

Preaccident Iodine Spike Release

In the preaccident iodine spike release scenario, a reactor transient has occurred prior to the postulated SGTR accident and has raised the primary coolant (PC) iodine concentration to the maximum value permitted by technical specifications. The PC iodine concentration of 60 μCi/g Dose Equivalent (DE) of I-131 is obtained from Figure 3.4-1 of Technical Specifications. The iodine dose conversion factors are developed to establish the iodine scaling factor based on the iodine concentration of 1 μCi/g DE I-131, which is used to convert the iodine concentration of 1% fuel defects to 1 μCi/g DE I-131. The total isotopic iodine activity in the RCS is conservatively calculated using the PC mass determined at "cooled" liquid conditions. Since the Salem 1 & 2 PC normal noble gas isotopic concentrations are not readily available, the 100/E-Bar noble gas concentrations are not calculated. The PC noble gas concentrations corresponding to 1% fuel defects are conservatively used to determine the total RCS noble gas activity, which are used to develop the RADTRAD nuclide inventory file (NIF) SGTRPRE_def.txt for the preaccident iodine spike case. The post-SGTR activity transport and CR response models are used to determine the post-SGTR radiological consequences using the Salem plant-specific as-built design inputs and assumptions. The CR response model utilizes the Salem CR air intake monitor to preferentially select the less contaminated air intake after 1 minute. A delay time of 1 minute for CR pressurization mode to be fully operational is conservatively assumed in this analysis. The EAB, LPZ, and CR doses are summarized in the Results Summary for the preaccident iodine spike case. The following paragraphs describe the activity transport paths from the one faulted and three intact steam generators (SGs). These activity transport paths are applicable to both the preaccident and concurrent iodine spike cases.

Faulted SG Release

The PC is postulated to be released into the faulted SG having a single SG tube ruptured. The resultant equilibrium break flow is assumed to continue for 30 minutes after the initiation of the event, at which time it is assumed that the operator isolates the faulted S/G and depressurizes the RCS pressure ending the transfer of reactor coolant to the secondary side of the faulted S/G and the release to the atmosphere. An assessment was done that determined that with a more sophisticated LOFTTR2 analysis the break flow termination could be extended to 55 minutes without exceeding the integrated break flow assumed in the original analysis. To maximize the offsite doses it is assumed that offsite power is lost so that the main steam condensers are not available. Iodine activity is released from the faulted SG to the environment in proportion to the steaming rate. The iodine activity from the P-T-S leakage is assumed to be directly released in the SG volume and partitioning of iodine is not credited. PC noble gas activity entering the faulted SG is assumed to be released to the environment without holdup or decontamination in the secondary system. For the secondary liquid iodine release from the faulted SG, an iodine partition coefficient of 10 is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183 to maximize the resulting dose and the SG steaming rates are reduced accordingly for the secondary liquid iodine release.

Intact SGs Release

Due to the SGTR transient the reactor shuts down and the RCS loses a substantial amount of coolant inventory with the potential of over flooding the faulted SG. In order to remove decay heat the plant begins to release the secondary coolant via the intact SGs' atmospheric relief valves. The steam release from the three intact SGs continues for 32 hours until the Residual Heat Removal (RHR) system is aligned to dissipate heat. During the 32 hours before the RHR system is initialized, the PC is assumed to leak into the three intact SGs at the maximum technical specification leak rate of 1 gpm. The PC noble gases are released directly to the environment from the RCS without holdup while the iodine activity is directly released from the intact SGs in proportion to the steaming rate. For the secondary liquid iodine release from the three intact SGs, an iodine partition coefficient of 10 is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183 to maximize the resulting dose.

Concurrent Iodine Spike:

In the concurrent iodine spike release scenario, the primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). The isotopic iodine appearance rates are calculated in Section 6.2 and Table 7 of Reference 10 using the specific activity production and removal rates based on the most conservative letdown system parameters. The total isotopic iodine activity is calculated using the spike duration of 8 hours. The total isotopic iodine and noble gas activities are used to develop the RADTRAD NIF SGTRCON_def.txt for the concurrent iodine spike case. The post-SGTR activity transport and CR response models are used to determine the post-SGTR radiological consequences using the Salem plant-specific as-built design inputs and assumptions. The CR response model utilizes the Salem CR air intake monitor to preferentially select the less contaminated air intake after 1 minute. The activity in the fuel is postulated such that the 99% of all fuel activity is released into the RCS within 8 hours at a calculated rate of 9.60 cfm. The activity transport from the RCS to the environment and the CR response are the same as that for the preaccident iodine spike case.

Intact Steam Generator Iodine Activity Release:

The iodine activity in the SGs is calculated based on a secondary side liquid activity concentration of 0.1 µCi/g DE I-131 using a larger SG volume. The iodine partition coefficient of 10 is conservatively used in lieu of the value of 100 recommended in Regulatory Guide 1.183 to maximize the resulting dose. The iodine activity is assumed to be released to the environment at the SG steaming rates.

ASSUMPTIONS

Regulatory Guide 1.183, Appendix F, provides guidance on modeling assumptions that are acceptable to the NRC staff for the evaluation of the radiological consequences of a SGTR accident. The following sections address the applicability of these modeling assumptions to a Salem Units 1 and 2 SGTR accident analyses. These assumptions are incorporated as design inputs in Section 5.0 of Reference 10.

The radioactive material released and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values.

Source Terms

There is no fuel damage postulated for the SGTR accident. Per RG 1.183 since no or minimal fuel damage is postulated for the limiting event, the released activity is the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking, a preaccident case and a concurrent case, are assumed.

Preaccident Iodine Spike Source Term

Consistent with RG 1.183 the first iodine spiking case assumes that a reactor transient has occurred prior to the postulated SGTR and has raised the primary coolant iodine concentration to the maximum value permitted by the technical specifications.

Per Technical Specification (TS) LCO for Specific Activity, the maximum primary coolant iodine concentration permitted by the technical specifications at full power is shown in TS Figure 3.4-1 to be 60 µCi/gm Dose Equivalent (DE) I-131. Technical Specification 1.10 defines DE I-131 as that concentration of I-131 (in µCi/g) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present using the thyroid dose conversion factors specified in Table III of TID-14844; this analysis calculates DE-131 in terms of the thyroid dose conversion factors specified in Federal Guidance Report 11 (Ref. 19).

Concurrent Iodine Spike Source Term

Consistent with RG 1.183 the second iodine spiking case assumes that the primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in Technical Specifications (i.e., concurrent iodine spike case). The assumed iodine spike duration is 8 hours.

Per Technical Specification LCO for Specific Activity, the equilibrium primary coolant iodine concentration permitted by the technical specifications is 1.0 µCi/gm DE I-131.

Noble Gas Source Term

In both preaccident and concurrent iodine spike cases the primary coolant noble gas (Xenon and Krypton) concentrations are assumed to correspond to the maximum coolant activity allowed by the Technical Specifications.

Per Technical Specification LCO Specific Activity the equilibrium primary coolant non-iodine concentration permitted by the technical specifications is 100/E-BAR. The parameter E-BAR is defined as the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

Since 1 percent failed fuel introduces more iodine activity into the primary coolant than is allowed by the Tech Specs, it can be conservatively assumed that 1 percent failed fuel introduces more non-iodine activity into the RCS than is allowed by the Tech Spec limit of 100/E-bar. Therefore, the PC noble gas concentrations corresponding to 1% fuel defects are conservatively used to determine the total RCS noble gas activity.

Activity Release from the Fuel into the Primary Coolant

Consistent with RG 1.183 the activity released from the fuel (for both the preaccident and concurrent iodine spike cases) is assumed to be released instantaneously and homogeneously through the primary coolant.

Iodine Chemical Form

Consistent with RG 1.183 the chemical form of iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released during normal operations, including iodine spiking.

Transport

Primary-to-Secondary (P-T-S) Leak Rates

Consistent with RG 1.183 (Ref. 3, Appendix F, Section 5.1) the primary-to-secondary leak rate is apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized. Technical Specification LCO for RCS Operational Leakage specifies a maximum limit of one gpm total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator. The total P-T-S leak rate of 1 gpm into the three intact SGs is postulated to maximize the dose because the P-T-S leakage continues for a longer duration (32 hours) in the intact SGs in comparison to the faulted SG (30 minutes).

Primary-to-Secondary Leak Primary Coolant Density

Consistent with RG 1.183 the primary coolant density, used in converting the volumetric primary-to-secondary leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr), is assumed to be 1.0 gm/cc (62.4 lbm/ft^3). Conversely, the temperature dependent specific volumes are used for converting the mass flow rates into volumetric flow rates to maximize the releases and consequently the doses.

Primary-to-Secondary Leak Duration

Consistent with RG 1.183 the primary-to-secondary leakage is assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 212°F. In this analysis primary coolant is conservatively assumed to leak into the intact SGs for 32 hours until the RHR system is put in operation.

Release of Fission Products

Consistent with RG 1.183 the release of fission products from the secondary system should be evaluated with the assumption of coincident loss of offsite power (LOOP). The offsite power is assumed to be lost so that the main steam condensers are not available for removal of the decay heat.

Noble Gas Releases to the Environment

All noble gases radionuclides released from the primary system are assumed to be released to environment without reduction or mitigation.

Iodine & Particulate Transport Model

Consistent with RG 1.183 the transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates. The post-SGTR thermal hydraulic condition in the faulted SG is such that a large amount of PC released through the ruptured tube excessively increases the coolant mass inventory, which eliminates the possibility of SG dryout condition and subsequently flashing of PC in the faulted SG. The coolant mass in the intact SGs provides adequate tube submergence to eliminate the flashing of PC in the primary-to-secondary leakage. Therefore, the transport model described in Regulatory Positions 5.5 & 5.6 of Appendix E is not applicable to the post-SGTR thermal-hydraulic conditions exist in the faulted and intact SGs. Because no fuel failure is assumed and the concentration of particulates in the PC is negligibly small compared to PC iodine and noble gas, therefore particulates are not considered in the analysis.

<u>Control Room Dose Consequences</u>

Regulatory Guide 1.183 provides guidance to be used in determining the total effective dose equivalent (TEDE) for persons located in the control room (CR). The following sections address the applicability of this guidance to the Salem Units 1 and 2 SGTR accident analyses. These assumptions are incorporated as design inputs.

Control Room Operator Dose Contributors

Consistent with RG 1.183, the CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel:

- Contamination of the control room atmosphere by the filtered CR ventilation inflow through the CR air intake and by unfiltered inleakage of the radioactive material contained in the post-accident radioactive plume released from the facility,
- Contamination of the control room atmosphere by filtered CR ventilation inflow via the CR air intake and by unfiltered inleakage of airborne radioactive material from areas and structures adjacent to the control room envelope,
- * Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud shine dose),
- Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose). This form of radiation shine is not applicable to a SGTR accident in which the activity is not released into the containment air space, and
- * Radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in CR intake and recirculation filters [i.e., CR filter shine dose]).

  * Note: The external airborne cloud shine dose and the CR filter shine dose due to a SGTR accident are insignificant compared to those due to a LOCA). Therefore, these direct dose contributions are considered to be insignificant and are not evaluated for a SGTR accident.

DESIGN INPUTS

The as-built design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in the Section 5.0 of Reference 10. The design inputs are compatible with the AST and TEDE dose criteria, and the assumptions are consistent with those identified in Appendix F of RG 1.183.

RESULTS SUMMARY

a) The results of the SGTR accident with the preaccident iodine spike are summarized in the following table with the CR monitors preferentially selecting the less contaminated CR air intake:

	SGTR Accident - Preaccident Iodine Case TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
P-T-S Iodine Release SGTRPID00	5.05E-01	2.17E+00 (occurs at t = 0)	3.25E-01
SC Liquid Iodine Release SGTRSLID00	3.83E-03	4.97E-03 (occurs at t = 0)	1.33E-03
Noble Gas Release SGTRPNG00	9.28E-03	3.37E-02 (occurs at t = 0)	4.88E-03
Total	5.18E-01	2.21E+00	3.31E-01
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**

b) The results of the SGTR accident with the concurrent iodine spike are summarized in the following table with the CR monitors preferentially selecting the less contaminated CR air intake:

	SGTR Accident - Concurrent Iodine Case TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Iodine Release	5.72E-01	1.53E+00	3.23E-01
SGTRCID01		(occurs at t = 0)	
SC Liquid Iodine Release SGTRSLID00	3.83E-03	4.97E-03 (occurs at t = 0)	1.33E-03
Noble Gas Release	9.28E-03	3.37E-02	4.88E-03
SGTRPNG00		(occurs at t = 0)	
Total	5.85E-01	1.57E+00	3.29E-01
Allowable TEDE Limit	**5.00E+00**	**2.50E+00**	**2.50E+00**

CONCLUSIONS

The SGTR accident results presented indicate that the EAB, LPZ, and CR doses due to a SGTR accident are within their allowable limits.

4.3 MAIN STEAM LINE BREAK RADIOLOGICAL ANALYSIS.

PURPOSE:

The purpose of this calculation is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to a Main Steam Line Break (MSLB) accident using the Alternative Source Term (AST) methodology and total effective dose equivalent (TEDE) dose criteria. The dose consequences are representative of an accident occurring in either unit.

The MSLB accident is analyzed using plant specific design and licensing bases inputs, which are compatible to the TEDE dose criteria. The MSLB analysis is performed using the guidance in Regulatory Guide 1.183 and its Appendix E.

ANALYTICAL APPROACH:

This calculation addresses the reactor coolant activity concentrations corresponding to (1) a preaccident iodine spike and (2) a concurrent iodine spike. There is no potential fuel failure during this event.

Preaccident Iodine Spike Release

In the preaccident iodine spike release scenario, a reactor transient has occurred prior to the postulated MSLB accident and has raised the primary coolant (PC) iodine concentration to the maximum value permitted by technical specifications. The PC iodine concentration of 60 μCi/g Dose Equivalent (DE) of I-131 is obtained from Figure 3.4-1 of the Technical Specifications. The iodine dose conversion factors are developed and used to establish the iodine scaling factor based on the iodine concentration of 1 μCi/g DE I-131, which is used to convert the iodine concentration of 1% fuel defects to 1 μCi/g DE I-131 in Table 3. The total isotopic iodine activity in the RCS is conservatively calculated using the PC mass determined at "cooled" liquid conditions. Since the Salem Units 1 & 2 PC normal noble gas isotopic concentrations are not readily available, the 100/E-Bar noble gas concentrations are not calculated. The PC noble gas concentrations corresponding to 1% fuel defects are conservatively used in Table 5 to determine the total RCS noble gas activity. The total isotopic activities released to the RCS are used to develop the RADTRAD nuclide inventory file (NIF) SMSLBPRE_def.txt for the preaccident iodine spike case.

The post-MSLB preaccident iodine spike activity transport and CR response models are shown in Figures 1, 3 & 4 of Reference 11, which are used to determine the post-MSLB radiological consequences using the Salem Units 1 & 2 plant-specific as-built design inputs and assumptions listed in Sections 4 & 5 of Reference 11. The CR response model utilizes the Salem Units 1 & 2 CR air intake monitor to preferentially select the less contaminated air intake after 1 minute (Ref. 9.10). A delay time of 1 minute for the CR pressurization mode to be fully operational is conservatively assumed. The EAB, LPZ, and CR doses are summarized in Results Summary Section for the preaccident iodine spike case.

The following paragraphs describe the activity transport paths from the one faulted and three intact steam generators (SGs). These activity transport paths are applicable to both the preaccident and concurrent iodine spike cases.

Faulted SG Releases

Following a MSLB the faulted SG is assumed to steam dry. PC is assumed to leak into the faulted SG at the technical specification maximum leak rate of 0.35 gpm through any one SG [0.35 gpm/day = (500 gal/day) / (1440 min/day)]. To maximize the offsite doses it is assumed that offsite power is lost so that the main steam condensers are not available. PC noble gas activity entering the faulted SG via Primary-to-Secondary (P-T-S) leakage is assumed to be released directly to the environment for 0-32 hrs through the pressure sensitive penetration area pressure relief panels (PAPRPs) using the applicable χ/Q values without reduction or mitigation. Since the faulted SG is assumed to be dried-out, the PC iodine activity entering the faulted SG via P-T-S leakage is also assumed to be released directly to the environment for 0-32 hrs through the PAPRPs. P-T-S activity dilution in the faulted SG is not credited. The release from the main steam line break outside containment is postulated as a ground level release to the environment.

Intact SGs Releases

To maximize the calculated doses it is assumed that offsite power is lost so that the main steam condensers are not available. Consequently, following the MSLB transient the plant must cool down by releasing secondary coolant via the intact SGs atmospheric relief valves. The steam release from the three intact SGs continues for 32 hours until the Residual Heat Removal (RHR) system is aligned to dissipate heat. During the 32 hours before the RHR system is initialized, the PC is assumed to leak into the three intact SGs at 0.65 gpm (i.e., the maximum technical specification total leakrate (1 gpm) minus the 0.35 gpm leakrate into the faulted SG). The PC noble gas activity entering the intact SGs via P-T-S leakage is assumed to be released directly to the environment without reduction or mitigation. The PC iodine activity entering the intact SGs via P-T-S leakage is assumed to be diluted within the smaller Unit 1 intact SGs liquid mass. An iodine partition coefficient of 1.0 (i.e., no P-T-S leakage iodine retention in the intact SGs) is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183 to maximize the resulting dose. The P-T-S leakage iodine activity is assumed to be released to the environment at the SG steaming rates. The MSSV Set 1 being closer to the CR air intake, the χ/Q values corresponds to the MSSV Set 1 release are conservatively used for the intact SG releases.

Concurrent Iodine Spike:

In the concurrent iodine spike release scenario, the primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 µCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). The isotopic iodine appearance rates are calculated using the specific activity production and removal rates based on the most conservative letdown system parameters. The total isotopic iodine activity is calculated using the spike duration of 8 hours. The total isotopic iodine and noble gas activities are and used to develop the RADTRAD NIF SMSLBCON_def.txt for the concurrent iodine spike case. The RADTRAD model postulates that the 99% of all fuel activity is released into the RCS within 8 hours at a calculated rate of 9.60 cfm. The activity transport from the RCS to the environment and the CR outside air intake radiation monitor response are the same as described for the preaccident iodine spike case. The post-MSLB concurrent iodine spike activity transport and CR response models are used to determine the post-MSLB radiological consequences using the Salem Units 1 & 2 plant-specific as-built design inputs and assumptions.

Initial Steam Generator Iodine Inventory Activity Release:

The initial iodine activity in the SGs is calculated based on the maximum secondary side liquid activity concentration of 0.1 µCi/g DE I-131 as allowed by the Technical Specifications. When evaluating secondary liquid iodine releases, the larger Unit 2 SG volume is used to maximize the total iodine activity in the secondary liquid, which in turn yields higher secondary liquid release dose consequences. The faulted and intact SGs volumes and liquid masses are calculated, and the corresponding secondary liquid iodine activities are calculated for the intact and faulted SG, respectively. The results are included in Reference 11.

Following a MSLB the faulted SG is assumed to steam dry in two hours. For the initial secondary liquid iodine inventory release from the faulted SG, an iodine partition coefficient of 1.0 (i.e., no secondary liquid iodine retention) is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183. The faulted SG secondary liquid iodine activity is assumed to be released to the environment through the PAPRPs at the faulted SG steaming rate of 24.67 cfm for 0-2 hours. The initial secondary liquid iodine activity release from the faulted is calculated and included in Reference 11.

For the initial secondary liquid iodine inventory release from the three intact SGs, an iodine partition coefficient of 1.0 (i.e., no secondary liquid iodine retention) is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183. The intact SGs secondary liquid iodine activity is assumed to be released to the environment at the intact SGs steaming rates. The initial secondary liquid iodine activity release from the intact SGs is modeled using the MSSV X/Q values. The intact SGs liquid iodine activities are calculated and included in Reference 11.

ASSUMPTIONS

Regulatory Guide 1.183, Appendix E, provides guidance on modeling assumptions that are acceptable to the NRC staff for the evaluation of the radiological consequences of a MSLB accident. The following sections address the applicability of these modeling assumptions to an Salem Units 1 & 2 MSLB accident analysis. These assumptions are incorporated as design inputs Section 5.0 of Reference 11.

The radioactivity material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values.

Source Terms

No fuel damage is postulated for the MSLB accident. Per RG 1.183 since no or minimal fuel damage is postulated for the limiting event, the released activity is the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking, a preaccident case and a concurrent case, are assumed.

Preaccident Iodine Spike Source Term

Consistent with RG 1.183 the first iodine spiking case assumes that a reactor transient has occurred prior to the postulated MSLB and has raised the primary coolant iodine concentration to the maximum value permitted by the technical specifications.

Per Technical Specification (TS) LCO for Specific Activity, the maximum primary coolant iodine concentration permitted by the technical specifications at full power is shown in TS Figure 3.4-1 to be 60 μCi/gm Dose Equivalent (DE) I-131. Technical Specification Section 1.10 defines DE I-131 as that concentration of I-131 (in μCi/g) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present using the thyroid dose conversion factors (DCFs) specified in Table III of TID-14844; the TS Section 1.10 definition of DE I-131 will be revised to be based on the DCFs specified in Federal Guidance Report 11 (Ref. 19).

Concurrent Iodine Spike Source Term

Consistent with RG 1.183 the second iodine spiking case assumes that the primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in technical specifications (i.e., concurrent iodine spike case). The assumed iodine spike duration is 8 hours.

Per Technical Specification LCO for reactor coolant specific activity, the equilibrium primary coolant iodine concentration permitted by the Technical Specifications is 1.0 μCi/gm DE I-131.

Noble Gas Source Term

In both preaccident and concurrent iodine spike cases the primary coolant noble gas (Xenon and Krypton) concentrations are assumed to correspond to the maximum coolant activity allowed by the technical specifications.

Per Technical Specification LCO Specific Activity, the equilibrium primary coolant non-iodine concentration permitted by the technical specifications is 100/E-BAR. The parameter E-BAR is defined as the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

Since 1 percent failed fuel introduces more iodine activity into the primary coolant than is allowed by the Tech Specs, it can be conservatively assumed that 1 percent failed fuel introduces more non-iodine activity into the RCS than is allowed by the Tech Spec limit of 100/E-bar. Therefore, the PC noble gas concentrations corresponding to 1% fuel defects are conservatively used to determine the total RCS noble gas activity modeled for both the preaccident and concurrent iodine spike cases.

Activity Release from the Fuel into the Primary Coolant

Consistent with RG 1.183 the activity released from the fuel (for both the preaccident and concurrent iodine spike cases) is assumed to be released instantaneously and homogeneously through the primary coolant.

Iodine Chemical Form

Consistent with RG 1.183, the chemical form of iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released during normal operations, including iodine spiking.

<u>Transport</u>

Primary-to-Secondary Leak Rates

Consistent with RG 1.183) for facilities such as Salem Units 1 & 2 that have traditional primary-to-secondary leak rate specifications (both per generator and total of all generators), the leakage is apportioned between faulted and intact steam generators in such a manner that the calculated dose is maximized. Per Technical Specification LCO for RCS operational leakage, the primary-to-secondary leak rate into the faulted steam generator (i.e., the generator supplying the ruptured main steam line) is 0.35

gpm [= (500 gal/day) / (1440 min/day)]. The total primary-to-secondary leak rate into the three intact steam generators is 0.65 gpm (corresponding to the maximum technical specification leak rate for all steam generators of 1 gpm per TS LCO] minus the leak rate into the faulted steam generator of 0.35 gpm).

Primary-to-Secondary Leak Primary Coolant Density

Consistent with RG 1.183 the primary coolant density used in converting the volumetric primary-to-secondary leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) is assumed to be 1.0 gm/cc (62.4 lbm/ft^3). Conversely, the temperature dependent specific volumes are used for converting the mass flow rates into volumetric flow rates to maximize the releases and consequently the doses.

Primary-to-Secondary Leak Duration

Per RG 1.183, the primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected (i.e., intact) steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated. Shutdown cooling will be placed into operation at 32 hours.

In this analysis primary coolant is assumed to leak into the faulted and intact steam generators and be available for release to the environment for 32 hours after the initiation of the event, at which time shutdown cooling will be placed into operation.

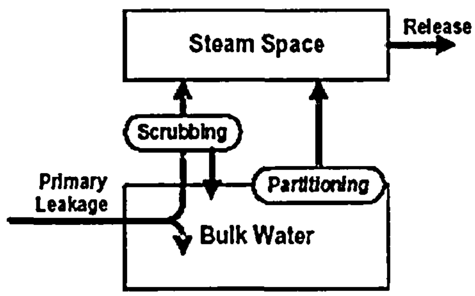
Noble Gas Releases to the Environment

Consistent with RG 1.183, all noble gas radionuclides released from the primary system (via the primary-to-secondary leaks) are assumed to be released to the environment without reduction or mitigation.

Transport Model

RG 1.183 presents the following generic steam generator transport model for use in evaluating an MSLB accident:

Steam Generator Transport Model



Release via the Faulted Steam Generator

Per RG 1.183 Appendix E, Section 5.5.1, a portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment without mitigation. Per RG 1.183 Appendix E, Section 5.6, operating experience and analyses has shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip. The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered.

In this analysis the faulted steam generator is assumed to steam dry. Consistent with RG 1.183 Appendix E, Section 5.5.1, during the period of faulted steam generator dryout, all of the 0.35 gpm primary-to-secondary leakage into the faulted steam generator is assumed to flash to vapor and all of the iodine and noble gas activity P-T-S leakage into the faulted steam generator is assumed to be released directly to the environment with no credit taken for iodine partitioning or scrubbing in the faulted steam generator.

Release via the Intact Steam Generators:

Per RG 1.183 Appendix E, Section 5.5.1 with regard to the unaffected (i.e., intact) steam generators used for plant cooldown, the 0.65 gpm primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence. All of the noble gas activity in the primary-to-secondary leakage into the three intact steam generators is assumed to be released directly to the environment with no credit taken for iodine partitioning or scrubbing in the faulted generator. All of the iodine activity in the P-T-S leakage into the three intact steam generators is assumed to mix with the secondary water without flashing during this period of total tube submergence. Consistent with RG 1.183 Appendix E, Section 5.5.4, the radioactivity in the bulk water of the three intact steam generators (consisting of the initial secondary side water iodine activity and the iodine activity introduced via primary-to-secondary leakage) is assumed to become vapor at a rate that is a function of the steaming rate and an assumed iodine partition coefficient of 1 to maximize the resulting dose.

Control Room Dose Consequences

Regulatory Guide 1.183 provides guidance to be used in determining the total effective dose equivalent (TEDE) for persons located in the control room (CR). The following sections address the applicability of this guidance to the Salem Units 1 & 2 MSLB accident analysis. These assumptions are incorporated as design inputs.

Control Room Operator Dose Contributors

Consistent with RG 1.183, the CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel:

- Contamination of the control room atmosphere by the filtered CR ventilation inflow through the CR air intake and by unfiltered inleakage of the radioactive material contained in the post-accident radioactive plume released from the facility,
- Contamination of the control room atmosphere by filtered CR ventilation inflow via the CR air intake and by unfiltered inleakage of airborne radioactive material from areas and structures adjacent to the control room envelope,
- * Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud shine dose),
- Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose). This form of radiation shine is not applicable to a MSLB accident in which the activity is not released into the containment air space, and
- * Radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in CR intake and recirculation filters [i.e., CR filter shine dose]).

* Note: The external airborne cloud shine dose and the CR filter shine dose due to a MSLB accident are insignificant compared to those due to a LOCA (see the core release fractions for LOCA and non-LOCA design basis accidents. Therefore, these direct dose contributions are considered to be insignificant and are not evaluated for a MSLB accident.

DESIGN INPUTS

The as-built design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in Section 5.0 of Reference 11. The design inputs are compatible with the AST and TEDE dose criteria, and the assumptions are consistent with those identified in Appendix E of RG 1.183.

RESULTS SUMMARY

a) The results of the MSLB accident with the preaccident iodine spike are summarized in the following table with the CR monitors preferentially selecting the less contaminated CR air intake:

	MSLB Accident - Preaccident Iodine Case TEDE Dose (rem)		
	Receptor Location		
	Control Room	**EAB**	**LPZ**
P-T-S Iodine Release FSG SMSLFPID00	6.62E-02	2.14E-02 (occurs at t = 0)	1.01E-02
P-T-S NG Release FSG SMSLFNG00	2.15E-04	8.77E-05 (occurs at t = 0)	4.11E-05
P-T-S Iodine Release ISG SMSLINPID00	3.62E-02	3.44E-02 (occurs at t = 8.0 hrs)	1.40E-02
P-T-S NG Release ISG SMSLBNG00	1.52E-04	1.62E-04 (occurs at t = 0)	7.57E-05
Liquid Iodine Release FSG SMSFSLID00	4.39E-03	7.10E-03 (occurs at t = 0)	1.02E-03
Liquid Iodine Release ISG SMSINSLID00	1.81E-02	3.15E-02 (occurs at t = 0)	4.79E-03
Total	**1.25E-01**	**9.46E-02**	**3.00E-02**
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**

FSG = Faulted SG   ISG = Intact SG

b) The results of the MSLB accident with the concurrent iodine spike are summarized in the following table with the CR monitors preferentially selecting the less contaminated CR air intake:

	MSLB Accident - Concurrent Iodine Case TEDE Dose (rem)		
	Receptor Location		
	Control Room	**EAB**	**LPZ**
P-T-S Iodine Release FSG SMSLFCID01	5.32E-01	1.85E-01 (occurs at t = 5.0 hrs)	7.75E-02
P-T-S NG Release FSG SMSLFNG00	2.15E-04	8.77E-05 (occurs at t = 0)	4.11E-05
P-T-S Iodine Release ISG SMSLINCID01	2.92E-01	3.12E-01 (occurs at t = 8.4 hrs)	1.05E-01
P-T-S NG Release ISG SMSLBNG00	1.52E-04	1.62E-04 (occurs at t = 0)	7.57E-05
Liquid Iodine Release FSG SMSFSLID00	4.39E-03	7.10E-03 (occurs at t = 0)	1.02E-03
Liquid Iodine Release ISG SMSINSLID00	1.81E-02	3.15E-02 (occurs at t = 0)	4.79E-03
Total	**8.47E-01**	**5.36E-01**	**1.88E-01**
Allowable TEDE Limit	**5.00E+00**	**2.50E+00**	**2.50E+00**

FSG = Faulted SG ISG = Intact SG

CONCLUSIONS:

The results of MSLB accident analyses indicate that the EAB, LPZ, and CR doses due to a MSLB accident are within their allowable limits.

4.4 LOCKED ROTOR RADIOLOGICAL ANALYSIS

PURPOSE:

The purpose of this calculation is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to a Reactor Coolant Pump (RCP) Locked Rotor Accident (LRA) using the Alternative Source Term (AST) methodology and total effective dose equivalent (TEDE) dose criteria. The dose consequences are representative of an accident occurring in either unit. The LRA is analyzed using plant specific design and licensing bases inputs, which are compatible to the TEDE dose criteria. The LRA analysis is performed using the guidance in Regulatory Guide 1.183 and its Appendix G.

ANALYTICAL APPROACH

The principal radiation source term for the LRA is the fuel gap activity in the failed fuel caused by the transient of the accident. It is assumed that 5% of the fuel cladding in the core has failed, with the failed fuel rod gap activity released immediately to the primary coolant (PC). The fuel gap isotopic activity release fractions are in accordance with RG 1.183. Previous LRA analysis had assumed a pre-accident iodine spike activity release in addition to the failed fuel activity release. In this analysis the pre-accident iodine spike activity release is not modeled. The exclusion of the pre-accident iodine spike is consistent with the guidance presented in RG 1.183, which states that the activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. Per the comparison of 5% failed fuel and pre-accident iodine spike activities presented, an activity release based on the projected 5% fuel damage would yield higher dose consequences than would a release of activity at the maximum Technical Specification limits. Therefore, the pre-accident iodine spike activity release is not modeled with the failed fuel activity release.

The iodine and noble gas core inventory at a core power level of 3600 MWt is obtained. The design basis core inventory is established based on scaling the 3600 MWt core inventory to an inventory based on 105% of the rated thermal power level of 3,459 MWt. The design basis core iodine and noble gas activity releases from the 5% failed fuel are calculated using the iodine and noble gas gap release fractions addressed in RG 1.183, Table 3. The activity released from the failed fuel is assumed to be instantaneously and homogeneously mixed through the primary coolant. The initial iodine and noble gas activity in the PC is calculated in Table 3 using the 1% fuel defects RCS activity concentrations and the Reactor Coolant System (RCS) coolant mass. This initial PC activity is added to the 5% failed fuel activity, which is used to develop RADTRAD nuclide inventory file (NIF) SLRA1_def.txt. The RCS activity is assumed to be released into the four SGs via a primary-to-secondary (P-T-S) leak rate of 1 gpm. Noble gases present in the P-T-S leakage are released directly to the environment without retention in the SGs. It is assumed that the SG tubes are not submerged in the SGs liquid, and consequently iodine introduced into the SGs via the P-T-S leakage is assumed to be directly released to the environment in proportion to the steam release rate, with no credit taken for iodine partitioning in the SG liquid. Thirty-two hours after the accident, the RHR system is assumed to be in operation and releases from the SGs have been terminated. The steaming rates of SGs are calculated based on the loss of load mass releases, which are applicable to the locked rotor accident.

The initial secondary coolant iodine activity used in this calculation is based on the Technical Specification limit of 0.1 μCi/g Dose Equivalent of I-131. The secondary liquid iodine activity is calculated and used in RADTRAD NIF file SLRASL_Iodine_def.txt to calculate dose consequences due to the secondary liquid iodine activity release. Secondary liquid iodine is assumed to be directly released to the environment in proportion to the steam release rate, with credit taken for iodine partitioning in the SG liquid. For the secondary liquid iodine release from the SGs, an iodine partition coefficient of 10 is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183. Thirty-two hours after the accident, the RHR system is assumed to be in operation, and no steam is released after this time.

ASSUMPTIONS

Regulatory Guide 1.183, Appendix G, provides guidance on modeling assumptions that are acceptable to the NRC staff for the evaluation of the radiological consequences of a LRA. The following sections address the applicability of these modeling assumptions to an Salem Units 1 and 2 Locked Rotor accident analyses. These assumptions are incorporated as design inputs in Section 5.0 of Reference 12.

The radioactivity material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values.

Source Term

There is 5% fuel damage postulated for the LRA per RG 1.183, Appendix G, Section 2. Since fuel damage is postulated for the limiting event, a radiological analysis is required.

Activity Release from the Fuel into the Primary Coolant

Consistent with RG 1.183 the activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

Iodine Chemical Form

Consistent with RG 1.183 the chemical form of iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine releases from the primary-to-secondary (P-T-S) leakage and the secondary liquid.

Transport

Primary-to-Secondary Leak Rates

Consistent with RG 1.183 the primary-to-secondary leak rate in the four steam generators is assumed to be the leak rate limiting condition for operation specified in the technical specifications.

Primary-to-Secondary Leak Primary Coolant Density

Consistent with RG 1.183, the primary coolant density used in converting the volumetric primary-to-secondary leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) is assumed to be 1.0 gm/cc (62.4 lbm/ft^3). It is assumed that the SG tubes are not submerged in the SGs liquid, and consequently iodine introduced into the four SGs via the P-T-S leakage is assumed to be directly released to environment with no credit taken for iodine partitioning in the SG liquid. For the secondary liquid iodine release from the SGs, an iodine partition coefficient of 10 is conservatively assumed in lieu of the value of 100 recommended in Regulatory Guide 1.183.

Primary-to-Secondary Leak Duration

Consistent with RG 1.183 the primary-to-secondary leakage is assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 212°F. In this analysis primary coolant is conservatively assumed to leak into the SGs for 32 hours until the RHR system is put in operation.

Release of Fission Products

Consistent with RG 1.183, the release of fission products from the secondary system is evaluated with the assumption of a coincident loss of offsite power (LOOP). The offsite power is assumed to be lost so that the main steam condensers are not available for removal of the decay heat.

Noble Gas Releases to the Environment

Consistent with RG 1.183 all noble gas radionuclides released from the primary system are assumed to be released to environment without reduction or mitigation.

Control Room Dose Consequences

Regulatory Guide 1.183 provides guidance to be used in determining the total effective dose equivalent (TEDE) for persons located in the control room (CR). The following sections address the applicability of this guidance to the Salem Units 1 and 2 LRA analyses. These assumptions are incorporated as design inputs.

Control Room Operator Dose Contributors

Consistent with RG 1.183, the CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel:

- Contamination of the control room atmosphere by the filtered CR ventilation inflow through the CR air intake and by unfiltered inleakage of the radioactive material contained in the post-accident radioactive plume released from the facility,
- Contamination of the control room atmosphere by filtered CR ventilation inflow via the CR air intake and by unfiltered inleakage of airborne radioactive material from areas and structures adjacent to the control room envelope,
- * Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud shine dose),
- Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose). This form of radiation shine is not applicable to a LRA in which the activity is not released into the containment air space, and
- * Radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in CR intake and recirculation filters [i.e., CR filter shine dose]).

  * Note: The external airborne cloud shine dose and the CR filter shine dose due to a LRA are insignificant compared to those due to a LOCA. Therefore, these direct dose contributions are considered to be insignificant and are not evaluated for a LRA.

DESIGN INPUTS

The as-built design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in Section 5.0 of Reference 12. The design inputs are compatible with the AST and TEDE dose criteria, and the assumptions are consistent with those identified in Appendix G of RG 1.183.

RESULTS SUMMARY

The results of the Locked Rotor accident are summarized in the following table with the CR monitors preferentially selecting the less contaminated air intake:

	RCP Locked Rotor Accident TEDE Dose (rem)		
	Receptor Location		
	Control Room	EAB	LPZ
P-T-S Iodine Release SLRAID00	1.29E+00	1.23E+00 (occurs @ 8.3 hrs)	5.01E-01
SC Liquid Iodine Release SLRASLID00	3.54E-03	6.24E-03 (occurs @ 0.0 hr)	1.32E-03
Noble Gas Release SLRANG00	9.52E-03	2.24E-02 (occurs @ 0.0 hr)	5.54E-03
Total	1.30E+00	1.26E+00	5.08E-01
Allowable TEDE Limit	**5.00E+00**	**2.50E+00**	**2.50E+00**

CONCLUSIONS

The Reactor Coolant Pump Locked Rotor accident results indicate that the EAB, LPZ, and CR doses due to a LRA are within their allowable limits.

4.5 ROD EJECTION RADIOLOGICAL ANALYSIS

PURPOSE

The purpose of this calculation is to determine the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to a Rod Ejection Accident (REA) using the Alternative Source Term (AST) methodology and total effective dose equivalent (TEDE) dose criteria. The dose consequences are representative of an accident occurring in either unit.

The REA is analyzed using plant specific design and licensing bases inputs, which are compatible to the TEDE dose criteria. The REA analysis is performed using the guidance in Regulatory Guide 1.183 and its Appendix H.

ANALYTICAL APPROACH

This calculation addresses two release cases. In the first, 100% of the activity released from the fuel is assumed to be released instantaneously and homogeneously throughout the containment atmosphere and available for release to the environment. In the second, 100% of the activity released from the fuel is assumed to be completely dissolved in the primary coolant (PC) and available for release to the environment through the secondary system.

Containment Leakage Release

The following activity is assumed to be instantaneously and homogeneously distributed in the containment following a control rod ejection accident:

- 10% of the core iodine and 10% of the core noble gases in the fuel gap of 10% failed fuel,
- 25% of the core iodine and 100% of the core noble gases in the 0.25% melted fuel, and
- 100% of the iodine and noble gases initially present (i.e., pre-REA) in the reactor coolant system (RCS)

During the first 24 hours the containment is assumed to leak at its maximum technical specification leak rate of 0.10 volume percent per day and at 50% of this leak rate for the remaining duration of the accident. Although allowed by Regulatory Guide 1.183, no credit is taken for a reduction in the amount of radioactive material available for leakage from the containment due to natural deposition and containment spray.

The post-REA activity released into the containment from the fuel and RCS is calculated. The iodine and noble gas core inventory at a core power level of 3600 MWt is obtained.. The design basis core inventory is established based on scaling the 3600 MWt core inventory to an inventory based on 105% of the rated thermal power level of 3,459 MWt . The pre-REA RCS activity available for release to the containment is calculated using the primary coolant activity concentrations based on 1% fuel defects and RCS mass. The post-REA iodine and noble gas activity released from the 10% failed fuel and 0.25% melted fuel are calculated in Table 3 with appropriate release fractions applied to the Table 1 design basis core inventory. The activities released from the fuel and RCS into the containment are input into RADTRAD nuclide inventory file (NIF) SREACNT_def.txt.

The post-REA containment leakage release activity transport and CR response models are used to determine the post-REA radiological consequences using the Salem plant-specific as-built design inputs and assumptions. The post-REA containment leakage release CR response model takes credit for CR operator action to manually initiate the CREACS operation 30 minutes after a REA because the activity concentration from this release path does not exceed the CR monitor setpoint. It is conservatively assumed that the CR HVAC draws outside air from the more contaminated CR air intake.

Secondary System Releases

To maximize the calculated secondary system release doses it is assumed that offsite power is lost so that the main steam condensers are not available. Due to the REA, the reactor is shutdown and the plant begins discharging secondary coolant via the steam generator (SG) relief valves. Steam release from the

SGs continues for 110 seconds with the total steam mass release of 5.12E+05 lbs. During the 110 seconds release period reactor coolant is assumed to leak into the SGs at the maximum technical specification rate of 1 gpm. In the secondary system release case, 100% of the activity released from the fuel is assumed to be completely dissolved in the primary coolant (PC) and available for release to the SGs via primary-to-secondary (P-T-S) leakage. The reactor coolant noble gases that enter the SGs via P-T-S leakage are released directly to the environment without reduction and mitigation. It is conservatively assumed that the SG tubes are uncovered (i.e., not submerged in the SGs liquid), and consequently iodine introduced into the SGs via the P-T-S leakage is assumed to be directly released to the environment in proportion to the steam release rate, with no credit taken for iodine partitioning in the SG liquid. Although the affected unit CR intake preferentially aligns with the less contaminated air intake in 1 minute after the REA, the 0-2 hour Unit 1 CR air intake χ/Q for MSSV Set 1 release is conservatively used for the P-T-S leakage iodine release for entire duration of 110 seconds without crediting the CREACS ability to align with less contaminated air intake.

The post-REA iodine and noble gas activities released from the 10% failed fuel and 0.25% melted fuel, which are completely dissolved in the PC and available for release to SGs, are calculated in Table 4 of Reference 9 with appropriate release fractions, and the RADTRAD NIF SREARCS_def.txt is developed using the total activity released to PC. The post-REA secondary system release activity transport and CR response models are used to determine the post-REA radiological consequences resulting from the P-T-S leakage. The post-REA secondary system release CR response model utilizes the Salem CR air intake monitors to preferentially select the less contaminated air intake because the activity concentration from this release path instantly exceeds the CR monitor setpoint. The assumed delay time of 1 minute for the CR pressurization mode to be fully operational is conservatively assumed.

The EAB, LPZ, and CR doses from the containment leakage and secondary system releases are combined. The actual doses from a REA would be a composite of the two pathways.

ASSUMPTIONS

Regulatory Guide 1.183, Appendix H, provides guidance on modeling assumptions that are acceptable to the NRC staff for the evaluation of the radiological consequences of a PWR rod ejection accident. The following sections address the applicability of these modeling assumptions to a Salem Units 1 and 2 REA analyses. These assumptions are incorporated as design inputs in Section 5.0 of Reference 9.

The radioactivity material releases and radiation levels used in the control room dose analysis are determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values.

Source Term

Fuel Damage Source Term

Fuel damage is postulated for the REA. Ten percent (10%) of the core is assumed to experience clad damage and one-fourth of one percent (0.25%) of the core is assumed to experience fuel melt as a result of the REA.

Consistent with RG 1.183, the release from the breached fuel to the containment and available for containment release, or to the primary coolant and available for release to the secondary system, is based on the estimate of the number of fuel rods breached (i.e., 10%) and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap.

Consistent with RG 1.183, the release attributed to fuel melting is based on the fraction of the fuel (i.e., 0.25%) that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment.

Consistent with RG 1.183, for the secondary system release pathway, 100% of the noble gases and 50% of the iodines in the fraction of the fuel (i.e., 0.25%) that reaches or exceed the initiation temperature for fuel melting are released to the reactor coolant.

The above source term requirements are incorporated as design inputs.

No Fuel Damage Source Term

Consistent with RG 1.183, since fuel damage is postulated for the REA a radiological analysis for the REA is required.

Release Cases to be Considered

Consistent with RG 1.183, two release cases are considered in this REA analysis. In the first, 100% of the activity released from the fuel is assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel is assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

Containment Airborne Iodine Form

Consistent with RG 1.183, the chemical form of radioiodine released to the containment atmosphere is assumed to be 95% cesium iodide (CsI) (i.e., particulate), 4.85% elemental iodine, and 0.15% organic iodide.

Steam Generator Release Iodine Form

Consistent with RG 1.183, iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.

Transport

Transport From Containment

Reduction in Radioactive Material

RG 1.183 allows for a reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features to be taken into account. Conservatively, in this analysis no reduction in the post-REA airborne activity is credited.

Containment Leakage

Consistent with RG 1.183 the containment is assumed to leak at the leak rate of 0.1 V%/day incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.

Transport From Secondary System

Primary-to-Secondary Leak Rates

Consistent with RG 1.183 a leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications is assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated. Releases from the steam generators terminate in 110 seconds, once the full secondary side mass in all four steam generators has discharged through the main steam safety valves.

Primary-to-Secondary Leak Primary Coolant Density

Consistent with RG 1.183 the primary coolant density used in converting the volumetric primary-to-secondary leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) are assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

Noble Gas Transport Model (Releases to the Environment)

Consistent with RG 1.183 all noble gas radionuclides released from the primary system (via the primary-to-secondary leaks) are assumed to be released to the environment without reduction or mitigation.

Iodine and Particulate Transport Model (Releases to the Environment)

RG 1.183 states that the transport model should be utilized for iodine and particulate. In this analysis, all iodine leaked into the steam generators is diluted within the steam generator liquid mass, and released to the environment at the steam generator steaming rate assuming an iodine partition coefficient of one. Since the post-REA release is less than 2 minutes in duration, and since iodine partitioning is not credited in the analysis, use of the recommended transport model is not appropriate for the steam mass release from the intact SGs.

Control Room Dose Consequences

Regulatory Guide 1.183 provides guidance to be used in determining the total effective dose equivalent (TEDE) for persons located in the control room (CR). The following sections address the applicability of this guidance to the Salem Units 1 and 2 REA analyses. These assumptions are incorporated as design inputs.

Control Room Operator Dose Contributors

Consistent with RG 1.183, the CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel:

- Contamination of the control room atmosphere by the filtered CR ventilation inflow through the CR air intake and by unfiltered inleakage of the radioactive material contained in the post-accident radioactive plume released from the facility,
- *Contamination of the control room atmosphere by filtered CR ventilation inflow via the CR air intake and by unfiltered inleakage of airborne radioactive material from areas and structures adjacent to the control room envelope,*
- * Radiation shine from the external radioactive plume released from the facility (i.e., external airborne cloud shine dose),

- * Radiation shine from radioactive material in the reactor containment (i.e., containment shine dose), and
- * Radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in CR intake and recirculation filters [i.e., CR filter shine dose].

* Note: The external airborne cloud shine dose, containment shine dose, and the CR filter shine dose due to a REA are insignificant compared to those due to a LOCA. Therefore, these direct dose contributions are considered to be insignificant and are not evaluated for a REA.

DESIGN INPUTS

The as-built design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in Section 5.0 of Reference 9.0. The design inputs are compatible with the AST and TEDE dose criteria and assumptions are consistent with those identified in Appendix E of RG 1.183.

RESULTS SUMMARY:

The results of the Rod Ejection accident are summarized in the following table with the CR monitors preferentially selecting the less contaminated air intake:

	Rod Ejection Accident TEDE Dose (rem)		
	Receptor Location		
	Control Room	**EAB**	**LPZ**
Containment Leakage SREACL01	1.03E+00	2.15E-01 (occurs @ 0.0 hr)	1.26E-01
P-T-S Iodine Release	3.26E-01	3.50E-02	5.00E-03
SREAID00		(occurs @ 0.0 hrs)	
Noble Gas Release	7.41E-03	3.14E-03	4.50E-04
SREANG00		(occurs @ 0.0 hr)	
Total	1.37E+00	2.53E-01	1.31E-01
Allowable TEDE Limit	**5.00E+00**	**6.30E+00**	**6.30E+00**

CONCLUSIONS

The Rod Ejection Accident results indicate that the EAB, LPZ, and CR doses due to a REA are within their allowable limits.

4.6 WASTE GAS DECAY TANK AND VOLUME CONTROL TANK RUPTURE RADIOLOGICAL ANALYSIS

This engineering evaluation addresses the radiological consequences of both Waste Gas Decay Tank (WGDT) and Volume Control Tank (VCT) rupture accidents using Alternative Source Term (AST) methodology including TEDE dose criteria.

WGDT AND VCT RUPTURE SCENARIOS

In accordance with the current WGDT and VCT Rupture dose analysis of record (AOR), the noble gas activity inventories in the WGDT and VCT are released to the environment via the plant vent as an instantaneous puff). No credit is taken for activity hold-up or decontamination. The released activity is atmospherically dispersed to the CR and offsite dose receptors.

The dose conversion factors used for this conversion are the effective TEDE for air submersion as listed in FGR-12 (Ref. 20). The dose rate due to the release of the WGDT activity inventory is greater than the dose rate due to the release of the VCT activity inventory. It is therefore concluded that the control room and offsite dose consequences of a puff release of the WGDT activity inventory bound the dose consequences of a puff release of the VCT activity inventory.

ANALYSIS METHODOLOGY

This engineering evaluation uses a conservative and simple radiological dose consequence analysis to evaluate the WGDT rupture accident. The assumptions, design inputs, and methodology described in this section are modeled in a RADTRAD computer code run, which calculates the WGDT rupture control room and offsite dose consequences.

This evaluation uses Version 3.02 of the RADTRAD computer code to calculate the potential radiological consequences of the WGDT rupture accident. The RADTRAD code is documented in NUREG/CR-6604.

The noble gas activity inventory in the WGDT is conservatively released to the environment via the plant vent as an instantaneous puff. The puff release is modeled in the RADTRAD computer code run by modeling a small WGDT volume of 10 ft^3 and a large release rate of 200,000 ft^3/minute.

This calculation is an ARCON96 code analysis that evaluates atmospheric dispersion between the Unit 1 plant vent and the control room (CR) HVAC air intake. Per the ARCON96 calculation, the χ/Q value defining a release from the plant vent to the CR HVAC air intake during the first moments of the WGDT Rupture release (when the entire inventory is released to the environment) is 1.78e-3 sec/m^3.

This engineering evaluation conservatively models only the CR normal mode of operation. No credit is taken for a switchover to the CR emergency mode of operation. No credit is taken for filtration of the contaminated outside air that enters the control room via the HVAC system or via inleakage. No credit is taken for cleanup of the contaminated CR air by the emergency recirculation unit filters.

The remainder of the WGDT Rupture radiological consequence model is consistent with the control room and offsite modeling employed in Steam Generator Tube Rupture (SGTR) AST dose Calculation.

RESULTS AND SUMMARY

The WGDT rupture accident control room and offsite doses are shown in the following table:

WGDT Rupture Accident Doses
(per RADTRAD output file SWGDT00.o0)

Dose Receptor	Dose (rem TEDE)
Control Room (Event Duration Dose)	2.07E-02
Exclusion Area Boundary (2-hour Dose)	4.10E-02 (from 0 to 2 hours)
Low Population Zone (Event Duration Dose)	5.86E-03

CONCLUSIONS

This engineering evaluation provides a radiological dose consequence analysis for the scenario of a Salem WGDT rupture accident. Using conservative and simple methods, this engineering evaluation calculates the WGDT rupture control room and offsite doses. These doses are trivially small compared to the 5 rem TEDE control room and 25 rem TEDE offsite dose criteria specified in 10 CFR 50.67

5.0 OVERALL CONCLUSIONS

Full implementation of the Alternative Source Term Methodology, as defined in Regulatory Guide 1.183, into the design basis accident analysis has been made to support the proposed changes to the Salem licensing basis.

Analyses of the Loss-of-Coolant Accident (LOCA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Reactor Coolant Pump Shaft Seizure (Locked Rotor), Control Rod Ejection, Letdown Line Break, Waste Gas Decay Tank and Volume Control Tank ruptures have been analyzed using RG 1.183 methodology. The analysis used assumptions consistent with the proposed changes in the Salem Units 1 and 2 licensing basis and the calculated doses do not exceed the defined acceptance criteria delineated in RG 1.83 and 10CFR 50.67.

A summary of Alternative Source Term Analysis results is shown below.

Salem Generating Station, Units 1 and 2
Summary of Alternative Source Term (AST) Analysis Results

Accident	EAB Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	Margin (%)	LPZ Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	Margin (%)	Control Room Dose (rem TEDE)	Acceptance Criteria (rem TEDE)	Margin (%)
LOCA									
Offsite and Salem Impact	3.26	25	87%	1.18	25	95%	3.83	5	23%
Hope Creek Impact	N/A	N/A	N/A	N/A	N/A	N/A	0.79	5	84%
SGTR									
Pre-accident Iodine Spike	2.21	25	91%	0.33	25	99%	0.52	5	90%
Coincident Iodine Spike	1.57	2.5	37%	0.33	2.5	87%	0.59	5	88%
MSLB									
Pre-accident Iodine Spike	0.09	25	100%	0.03	25	100%	0.13	5	98%
Coincident Iodine Spike	0.54	2.5	79%	0.19	2.5	92%	0.85	5	83%
Locked Rotor	1.26	2.5	50%	0.51	2.5	80%	1.30	5	74%
Rod Ejection	0.25	6.3	96%	0.13	6.3	98%	1.37	5	73%
WGDT Rupture	0.04	25	100%	0.01	25	100%	0.02	5	100%
VCT Rupture	<0.04	25	100%	<0.01	25	100%	<0.02	5	100%

6.0 NO SIGNIFICANT HAZARDS CONSIDERATIONS

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards are involved if the operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

This change does not involve a significant hazards consideration for the following reasons:

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

Alternative source term calculations have been performed for Salem Units 1 and 2 that demonstrate the dose consequences remain below limits specified in NRC Regulatory Guide 1.183 and 10 CFR 50.67. The proposed changes do not modify the design of the plant. The use of AST changes only the regulatory assumptions regarding analytical treatment of the design basis accidents and has no direct effect on the probability of any accident. The AST has been utilized in the analysis of limiting design basis accidents as described in Attachment 1. The results of the analyses, which include the proposed changes to the Technical Specifications, demonstrate that the dose consequences of these limiting events are within the regulatory limits.

The proposed full-implementation of an alternative source term at Salem Nuclear generating Station Units 1 and 2 and the proposed changes to TS Sections 1.10 and 3/4.7.7 for Units 1 and 2 do not involve a change to structures, systems, or components that would affect the probability of an accident previously evaluated in the Salem Generating Station Updated Final safety Analysis Report (SGS-UFSAR).

The proposed changes involve increasing allowable ESF leakage to provide for increased operational flexibility and eliminating credit for the Auxiliary Building Exhaust Air Filtration System in a revised analysis of the radiological consequences of a loss-of-coolant accident (LOCA) and also increasing the amount of inleakage into the control room envelope assumed in a revised analysis of the radiological consequences of a LOCA. The analysis of the effects of the proposed changes result in acceptable radiological consequences for the design basis LOCA previously evaluated in SGS-UFSAR Section 15.4.1. The revised analysis calculated offsite and control room operator dose consequences using an alternative source term and following the guidance provided in Regulatory Guide 1.183. The results demonstrate that dose consequences remain within acceptable limits (that is, the acceptance criteria specified in 10CFR50.67.

The table of results provided in Section 4.1 of this attachment contains the post-LOCA dose results that were based on Calculation S-C-ZZ-MDC-1945. It can be concluded from these results that with the changes proposed by this amendment application, the calculated doses following a LOCA for the control room, exclusion area boundary, and low population zone remain below the allowable regulatory limits.

Likewise, the tables of results provided in Sections 4.2 through 4.6 of this attachment demonstrate that the calculated doses following non-LOCA accidents for the control room, exclusion area boundary, and low population zone remain below the allowable regulatory limits.

The ABVS filtration system is not modeled in the Salem Units 1 and 2 Probabilistic Risk Assessment (PRA). Thus, the changes requested for the ABVS filtration system do not invalidate assumptions made in the Salem PRA and do not adversely impact the severe accident management program.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

There are no physical changes to the plant associated with this request, and the plant conditions for which PSEG evaluated design-basis accidents remain valid. The filtration system of the ABVS will be maintained in accordance with the UFSAR surveillance requirements and its operation will be at the discretion of the operations staff. The capability to place the filtration system in service is not modified by this amendment request. Consequently, this proposal introduces no new failure modes.

The elimination of the 24 hours Action Statement related to ABVS filtration and the relocation of the ABVS filtration surveillances to the Salem UFSAR meets the exclusion criteria for Technical Specifications contained in 10 CFR 50.36 (c)(2)(ii). The relocation of the filtration surveillances to the UFSAR maintains compliance with 10 CFR 20 and the ODCM requirements.

The ABVS filtration system is not modeled in the Salem Units 1 and 2 Probabilistic Risk Assessment (PRA). Thus, the changes requested for the ABVS filtration system do not invalidate assumptions made in the Salem PRA and do not adversely impact the severe accident management program.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

3) **The proposed amendment does not involve a significant reduction in the margin of safety.**

The proposed implementation of the alternative source term methodology is consistent with NRC Regulatory Guide 1.183. The Technical Specification and UFSAR changes result in operation in accordance with Regulatory guidelines and support the revisions to the radiological analyses of the limiting design basis accidents. Conservative methodologies, per the guidance of RG 1.183, have been used in performing the accident analyses. The radiological consequences of these accidents are all within the regulatory acceptance criteria associated with the use of the alternative source term methodology.

The proposed changes continue to ensure that the doses at the exclusion area, low population zone boundaries and in the control room are within the corresponding regulatory limits of RG 1.183 and 10 CFR 50.67. The results of the offsite and control room dose calculations demonstrate that the consequences of the design basis events are within the limits specified in 10CFR50.67 and the acceptance criteria identified in Regulatory Guide 1.183. Acceptable margins of safety are inherent in the acceptance criteria identified in the regulatory guide. Therefore, sufficient margin relative to regulatory limits is maintained as a result of the revised analyses (see the tables of results provided in sections 4.1 through 4.6 of this attachment).

The ABVS filtration system is not modeled in the Salem Units 1 and 2 Probabilistic Risk Assessment (PRA). Thus, the changes requested for the ABVS filtration system do not invalidate assumptions made in the Salem PRA and do not adversely impact the severe accident management program.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above discussion, PSEG has determined that the proposed changes do not involve a significant safety hazards consideration.

7.0 Environmental Considerations

This amendment request proposes to modify the design basis source term according to provisions set forth in 10 CFR 50.67, "Accident Source Term. Offsite or onsite radiation doses have been calculated to be within the regulatory guidelines of RG 1.183. The proposed amendment involves no significant hazard, no significant change in the types of effluents that may be released offsite, and there is no significant increase in the individual or cumulative occupational radiation exposure. Thus, this request causes no significant effect on the environment. This proposed amendment accordingly meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). In accordance with provisions of 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for this amendment.

8.0 References

1. USAEC Technical Information Document TID-14844, "Calculation of Distance Factors for Power and test Reactor Sites," by J. J. DiNunno, et al, for U. S. Atomic Energy Commission, dated 1962
2. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants, Final Report," by L. Soffer, S. B. Burson, C. M. Ferrell, R. Y. Lee, J. N. Ridgely, dated February 1995
3. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," dated July 2000
4. NUREG 0800, Standard Review Plan, SRP-15.0.1, Rev. 0 "Radiological Consequence Analyses Using Alternative Source Terms," dated July 2000
5. S-C-ZZ-MDC-1987, Input Parameters for Salem AST Dose Calculations
6. S-C-ZZ-MDC-1945, Post-LOCA EAB, LPZ, and Control Room Dose- AST
7. S-C-ZZ-MDC-1946, Post LOCA TSC Doses-AST
8. S-C-ZZ-MDC-1947, Post LOCA Vital Access Area Mission Doses-AST
9. S-C-ZZ-MDC-1948, EAB, LPZ, and Control Room Doses- Rod Ejection Accident
10. S-C-ZZ-MDC-1949, EAB, LPZ, and Control Room Doses- Steam Generator Tube Rupture Accident
11. S-C-ZZ-MDC-1950, EAB, LPZ, and Control Room Doses- Main Steam Line Break Accident
12. S-C-ZZ-MDC-1951, EAB, LPZ, and Control Room Doses- RCP Locked Rotor Accident
13. S-C-ZZ-MDC-2005, Hope Creek Control Room Habitability for LOCA Occurrence at Salem 2 Plant
14. S-C-ZZ-MDC-2008, Radiological Impact for AST on the Salem EQ Program
15. S-C-ZZ-MEE-1793, Post LOCA pH Confirmation Calculation for Salem AST Phase-2
16. S-C-ZZ-MEE-1797, ESF System Leakage to the Auxiliary Building and Refueling Water Storage Tank
17. S-C-ZZ-MEE-1799, Letdown Line Break Accident-AST
18. S-C-ZZ-MEE-1805, Waste Gas Decay Tank and Volume Control Tank Rupture Accidents.
19. Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency
20. Federal Guidance Reports 12, EPA-402-R-93-081, Environmental Protection Agency.
21. "RADTRAD, Version 3.02: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, Supplement 1, USNRC, June 8, 1999
22. S-C-ZZ-MDC-1959, CR χ/Qs Using ARCON96 Code – Non-LOCA Releases
23. S-C-ZZ-MDC-1912, Control Room c/Qs Using ARCON96 Code – Equipment Hatch, Plant Vent, FHB Rollup Door Releases

Salem Nuclear Generating Station Units 1 and 2
Facility Operating Licenses DPR-70 and DPR-75

Attachment 2

Revisions to the Technical Specifications
Implementation of Alternative Source Term
Regulatory Guide 1.183

SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

REVISIONS TO THE TECHNICAL SPECIFICATIONS
IMPLEMENTATION OF ALTERNATIVE SOURCE TERM
REGULATORY GUIDE 1.183

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Salem Unit 1, Facility Operating License No. DPR-70, are affected by this change request:

Technical Specification	Page
1.10	1-2, 1-3 (format only)
3/4.7.7	3/4 7-22
	3/4 7-23
	3/4 7-24
B3/4.7.7	B 3/4 7-5c
	B 3/4 7-5d

The following Technical Specifications for Salem Unit 2, Facility Operating License No. DPR-75, are affected by this change request:

Technical Specification	Page
1.10	1-2, 1-3 (format only)
3/4.7.7	3/4 7-18
	3/4 7-19
	3/4 7-20
B 3/4 .7.7	B 3/4 7-5c
	B 3/4 7-5d

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or 0-rings) is OPERABLE.

1.8 NOT USED

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of 1-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites."~~ **Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion".**

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 228 steps withdrawn, inclusive. FULLY WITHDRAWN will be specified in the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

4.7.7.1 The above required Auxiliary Building ~~exhaust air filtration~~ **Ventilation System** shall be demonstrated OPERABLE **by:**

a. At least once per 12 hours by verifying negative pressure in the Auxiliary Building.

~~a)~~.**b.** At least once per 31 days by **starting each fan** ~~initiating~~, from the control room, ~~flow through the HEPA filter and charcoal adsorber train~~ and verifying that ~~the filter train and~~ each fan operates for at least 15 minutes~~.~~

c. At least once per 18 months by verifying that the System starts following a Safety Injection Test Signal.

~~b. At least once per 18 months by: or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:~~

~~1. Verifying that with the system operating at a flow rate of 21,400 cfm ± 10 % and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the ventilation system to the facility vent, including leakage through the ventilation system diverting valves, is ≤ 1% when the system is tested by admitting cold DOP at the system intake.~~

~~2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.~~

~~3. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.~~

~~4. Verifying within 31 days after removal from the ABV unit, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 15.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, at a nominal face velocity of 74 ft/min, and a relative humidity of 95%.~~

~~5. Verifying a system flow rate of 21,400 cfm ± 10% during system operation~~

~~c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the ABV unit, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 15.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, at a nominal face velocity of 74 ft/min, and a relative humidity of 95%.~~

~~d. At least once per 18 monthsby:~~

~~1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4 inches Water Gauge while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.~~

~~2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers.~~

~~e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the ventilation system at a flow rate of 21,400 cfm ± 10%.~~

The OPERABILITY of this system in conjunction with control room design provisions is based **on** providing ~~on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent~~ **adequate radiation protection to permit access to and occupancy of the Salem control room for the entire duration of the postulated accident, with no person in the control room receiving radiation exposure that exceeds 5 ~~REM~~ rem TEDE. This limitation is consistent with the requirements of** ~~General Design Criterion 19 of Appendix "A", 10 CFR 50.~~ **Regulatory Guide 1.183.**

3/4.7.7 AUXILIARY BUILDING ~~EXHAUST AIR FILTRATION~~VENTILATION SYSTEM

The Auxiliary Building Ventilation System (ABVS) consists of two major subsystems. They are designed to; control Auxiliary Building temperature during normal and emergency modes of operation, **to maintain slightly negative pressure in the building to prevent unmonitored leakage out of the building** and, to contain Auxiliary Building airborne contamination during Loss of Coolant Accidents (LOCA).

The two subsystems are:

1. A once through filtration exhaust system, designed to contain particulate and gaseous contamination and prevent it from being released from the building in accordance with 10CFR20 **(no credit is taken for post-accident filtration)**, and

2. A once through air supply system, designed to deliver outside air into the building to maintain building temperatures **and negative pressure** within acceptable limits. For the purposes of satisfying the Technical Specification LCO, one supply fan must be administratively removed from service such that the fan will not auto-start on an actuation signal; however, the supply fan must be OPERABLE with the exception of this administrative control.

These systems operate during normal and emergency plant modes. Additionally, the system provides a flow path for containment purge supply and exhaust during Modes 5 and 6.

Appropriate filtration surveillances are contained in the UFSAR Section 9.4.2.4, Test and Inspections.

The **ventilation** exhaust ~~system~~ consists of three 50% capacity fans that are powered from vital buses. ~~These fans exhaust from a common plenum downstream from three High Efficiency Particulate Air (HEPA) filter banks, two of which, 11 & 12 can be interchangeably aligned to discharge to a single carbon adsorber bed. Filter unit 11 is limited in capacity and can only be aligned to the ECCS areas of the Auxiliary Building for HEPA only or HEPA + Carbon modes of filtration. Filter unit 12 can be used to ventilate the normal areas of the Auxiliary Building in HEPA only, or when used in conjunction with 13, may be used to ventilate the ECCS areas of the Auxiliary Building in HEPA + Carbon. Filter unit 13 does not communicate with the carbon adsorber housing and is used for exhausting air from the normal areas of the Auxiliary Building during any plant Mode or purging the Containment Building during Modes 5&6.~~ The fans are designed for continuous operation, to control the Auxiliary Building pressure at -0.10" Water Gauge with respect to atmosphere.

The **ventilation** supply ~~system~~ consists of two 100% capacity fans that are powered from vital buses, and distribute outdoor air to the general areas and corridors of the building through associated ductwork.

===

AUXILIARY BUILDING VENTILATION ALIGNMENT MATRIX

~~Unit 11 from ECCS HEPA only, with~~
~~Unit 12 from Aux. Normal HEPA only; or~~

NORMAL VENTILATION (Normal plant operations)*

~~Unit 11 from ECCS HEPA only, with~~
~~Unit 13 from Aux. Normal HEPA only; or~~

~~Unit 12 from ECCS HEPA only, with~~
~~Unit 13 from Aux. Normal HEPA only; and~~

Any two of the three exhaust fans and either of the two supply fans.

* The normal alignment is two exhaust fans and one supply fan. During cooler seasons, and with the absence of the system heating coils, it may be required to limit the amount of colder outside air entering the building. In this case, it is acceptable to secure both supply fans from operation and reduce the number of operating exhaust fans to one. There is sufficient capacity with the single exhaust fan to maintain the negative pressure within the auxiliary building boundary.

EMERGENCY VENTILATION (Emergency plant operations)

~~Unit 11 from ECCS HEPA + Unit 14, with~~
~~Unit 12 from Aux. Normal HEPA only; or~~

~~Unit 11 from ECCS HEPA + Unit 14, with~~
~~Unit 13 from Aux. Normal HEPA only; or~~

~~Unit 12 from ECCS HEPA + Unit 14, with~~
~~Unit 13 from Aux. Normal HEPA only; and~~

At least two of the three exhaust fans and either one of the two supply fans.

Note: During a Safety Injection (SI) all three exhaust fans and one of the supply fans will start. This is acceptable and will maintain the boundary pressure while supplying the required cooling to the building. Should access/egress become difficult with the three exhaust fans running, ~~then~~ one of the exhaust fans should be secured.

OPERABILITY of the Auxiliary Building ~~exhaust air filtration system~~**Ventilation System** ensures that air, which may contain radioactive materials leaked from ECCS equipment following a LOCA, is ~~filtered and~~ monitored prior to release from the plant **via the plant vent**. Operation of this system and the resultant effect on offsite **and control room** doseage calculations was assumed in the accident analyses. ~~Laboratory testing of the carbon adsorber is performed in accordance with ASTM D3803-1989 with an acceptance criteria that is determined by applying a minimum safety factor of 2 to the charcoal filter removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.~~ ABVS is discussed in Updated Final Safety Analysis Report (UFSAR) Section 9.4.2.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or 0-rings) is OPERABLE.

1.8 NOT USED

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Unit operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844 "Calculation of Distance Factors for Power and Test Reactor Sites"~~**Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion".**

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FULLY WITHDRAWN

1.13a FULLY WITHDRAWN shall be the condition where control and/or shutdown banks are at a position which is within the interval of 222 to 228 steps withdrawn, inclusive. FULLY WITHDRAWN will be established by the current reload analysis.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Water Injection) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

3/4.7.7 AUXILIARY BUILDING ~~EXHAUST AIR FILTRATION~~VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 At least ~~one Auxiliary Building exhaust air HEPA filter train, associated with the one charcoal adsorber bank,~~ two supply fans, and three exhaust fans shall be OPERABLE (*) **to maintain the Auxiliary Building at slightly negative pressure.**

-----------------------------NOTE-----------------------------

The intermittent opening of the Auxiliary Building pressure boundary causing a loss of negative pressure may be performed under administrative controls.

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APPLICABILITY: ~~MODES 1, 2, 3 and 4~~**At all times.**

ACTION:

Modes 1 thru 4

~~a)With the above required HEPA filter train inoperable, restore the HEPA filter train to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b)With the charcoal adsorber bank inoperable, restore the charcoal adsorber bank to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~c)~~**a)** With one supply fan and/or one exhaust fan inoperable, restore the fan(s) to OPERABLE status within 14 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b) With two **supply and/or two** exhaust fans inoperable restore **at least** one inoperable **supply and two** exhaust fans to operable status within 24 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c) **With the Auxiliary Building pressure not maintained slightly negative, restore the building to slightly negative pressure within the next 4 hours or**

~~d)With no supply fans operable,~~ be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~e)With no exhaust fans operable, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

During CORE ALTERATIONS

d) **With the Auxiliary Building pressure not maintained slightly negative, restore the Auxiliary Building to slightly negative pressure within the next 4 hours or suspend all operations involving CORE ALTERATIONS.**

At all times

e) **With the Auxiliary Building pressure not maintained slightly negative, suspend all operations involving radioactive gaseous releases via the Auxiliary Building immediately.**

(*) One of the supply fans may be considered OPERABLE with its auto start circuit administratively controlled (removed form service) to prevent more than one supply fan from operating at any time.

4.7.7 The above required Auxiliary Building ~~exhaust air filtration~~**Ventilation System** shall be demonstrated OPERABLE:

a) **At least once per 12 hours by verifying negative pressure in the Auxiliary Building.**

~~a.~~**b)** At least once per 31 days by ~~initiating~~**starting each fan**, from the control room, ~~flow through the HEPA filter and charcoal adsorber train and verifying that the filter train and~~ each fan operate for at least 15 minutes.

~~b.~~**c)** At least once per 18 months **by verifying that the system starts following a Safety Injection Test Signal.**~~or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system, by:~~

~~1. Verifying that with the system operating at a flow rate of 21,400 cfm ± 10 % and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the ventilation system to the facility vent, including leakage through the ventilation system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.~~

~~2. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas and that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place using the test procedure guidance of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978 (except for the provisions of ANSI N510 Sections 8 and 9), and the system flow rate is 21,400 cfm ± 10%.~~

~~3. Verifying within 31 days after removal from the ABV unit, that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows the methyl iodide penetration less than 15.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, at a nominal face velocity of 74 ft/min, and a relative humidity of 95%.~~

~~4. Verify that the system flowrate does not exceed the design limit of 23,540 cfm (21,400 cfm + 10%) when the HEPA + Charcoal adsorber filter train is aligned to the ECCS equipment areas.~~

~~c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal from the ABV unit, that a laboratory analysis of a representative carbon sample, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, shows a methyl iodide penetration less than 15.0% when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, at a nominal face velocity of 74 ft/min, and a relative humidity of 95%.~~

~~d. At least once per 18 months by:~~

~~1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than 4 inches Water Gauge while operating the system at a flow rate of 21,400 cfm ± 10%.~~

~~2. Verifying that the system starts on a Safety Injection Test Signal.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place while operating the system at a flow rate of 21,400 cfm ± 10%.~~

~~f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the system at a flow rate of 21,400 cfm ± 10%.~~

The OPERABILITY of this system in conjunction with control room design **features**~~provisions is based~~**provides adequate radiation protection to permit access to and occupancy of the Salem control room for the entire duration of the postulated accident with no person in the control room receiving radiation exposure that exceeds 5 rem TEDE.** ~~on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent.~~ This limitation is consistent with the requirements of ~~General Design Criterion 19 of Appendix "A", 10 CFR Part 50~~**Regulatory Guide 1.183.**

3/4.7.7 AUXILIARY BUILDING ~~EXHAUST AIR FILTRATION~~VENTILATION SYSTEM

The Auxiliary Building Ventilation System (ABVS) consists of two major subsystems. They are designed to control Auxiliary Building temperature during normal and emergency modes of operation, **to maintain slightly negative pressure in the building to prevent unmonitored leakage out of the building** and to contain Auxiliary Building airborne contamination during Loss of Coolant Accidents (LOCA). The two subsystems are:

1. A once through filtration exhaust system, designed to contain particulate and gaseous contamination and prevent it from being released from the building in accordance with 10CFR20 **(no credit is taken for post-accident filtration)**, and

2. A once through air supply system, designed to deliver outside air into the building to maintain building temperatures **and negative pressure** within acceptable limits. For the purposes of satisfying the Technical Specification LCO, one supply fan must be administratively removed from service such that the fan will not auto-start on an actuation signal; however, the supply fan must be OPERABLE with the exception of this administrative control.

These systems operate during normal and emergency plant modes. Additionally, the system provides a flow path for containment purge supply and exhaust during Modes 5 and 6.

Appropriate filtration surveillances are contained in the Updated Final Safety Analysis Report (UFSAR) Section 9.4.2.4, Test and Inspections.

The **ventilation** exhaust ~~system~~ consists of three 50% capacity fans that are powered from vital buses. ~~These fans exhaust from a common plenum downstream from three High Efficiency Particulate Air (HEPA) filter banks, two of which, 21 & 22 can be interchangeably aligned to discharge to a single carbon adsorber bed. Filter unit 21 is limited in capacity and can only be aligned to the ECCS areas of the Auxiliary Building for HEPA only or HEPA + Carbon modes of filtration. Filter unit 22 can be used to ventilate the normal areas of the Auxiliary Building in HEPA only, or when used in conjunction with 23, may be used to ventilate the ECCS areas of the Auxiliary Building in HEPA + Carbon. Filter unit 23 does not communicate with the carbon adsorber housing and is used for exhausting air from the normal areas of the Auxiliary Building during any plant Mode or purging the Containment Building during Modes 5&6.~~ The fans are designed for continuous operation, to control the Auxiliary Building pressure at -0.10" Water Gauge with respect to atmosphere.

The **ventilation** supply ~~system~~ consists of two 100% capacity fans that are powered from vital buses, and distribute outdoor air to the general areas and corridors of the building through associated ductwork.

3/4.7.7 AUXILIARY BUILDING EXHAUST AIR FILTRATION SYSTEM (cont'd)

AUXILIARY BUILDING VENTILATION ALIGNMENT MATRIX
NORMAL VENTILATION (Normal plant operations)*

~~Unit 21 from ECCS HEPA only, with~~
~~Unit 22 from Aux. Normal HEPA only; or~~

~~Unit 21 from ECCS HEPA only, with~~
~~Unit 23 from Aux. Normal HEPA only; or~~

~~Unit 22 from ECCS HEPA only, with~~
~~Unit 23 from Aux. Normal HEPA only; and~~

Any two of the three exhaust fans and either of the two supply fans.

* The normal alignment is two exhaust fans and one supply fan. During cooler seasons, and with the absence of the system heating coils, it may be required to limit the amount of colder outside air entering the building. In this case, it is acceptable to secure both supply fans from operation and reduce the number of operating exhaust fans to one. There is sufficient capacity with the single exhaust fan to maintain the negative pressure within the auxiliary building boundary.

EMERGENCY VENTILATION (Emergency plant operations)

~~Unit 21 from ECCS HEPA + Unit 24, with~~
~~Unit 22 from Aux. Normal HEPA only; or~~

~~Unit 21 from ECCS HEPA + Unit 24, with~~
~~Unit 23 from Aux. Normal HEPA only; or~~

~~Unit 22 from ECCS HEPA + Unit 24, with~~
~~Unit 23 from Aux. Normal HEPA only; and~~

At least two of the three exhaust fans and either one of the two supply fans.

Note: During a Safety Injection (SI) all three exhaust fans and one of the supply fans will start. This is acceptable and will maintain the boundary pressure while supplying the required cooling to the building. Should access/egress become difficult with the three exhaust fans running, ~~then~~ one of the exhaust fans should be secured.

OPERABILITY of the Auxiliary Building ~~exhaust air filtration system~~**Ventilation System** ensures that air, which may contain radioactive materials leaked from ECCS equipment following a LOCA, is ~~filtered and~~ monitored prior to release from the plant **via the plant vent**. Operation of this system and the resultant effect on offsite **and control room** doseage calculations was assumed in the accident analyses. ~~Laboratory testing of the carbon adsorber is performed in accordance with ASTM D3803-1989 with an acceptance criteria that is determined by applying a minimum safety factor of 2 to the charcoal filter removal efficiency credited in the design basis dose analysis as specified in Generic Letter 99-02.~~ ABVS is discussed in ~~Updated Final Safety Analysis Report (~~UFSAR~~)~~ Section 9.4.2.