February 17, 2006

Mr. William Levis
Senior Vice President & Chief Nuclear Officer
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: ALTERNATE SOURCE TERM (TAC NOS. MC3094 AND MC3095)

Dear Mr. Levis:

The Commission has issued the enclosed Amendment Nos. 271 and 252 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 26, 2004, as supplemented by letters dated September 16, 2004, September 23, 2004, February 25, 2005, and June 13, 2005.

These amendments revise the TSs to incorporate a full-scope application of an alternate source term methodology in accordance with Title 10 of the *Code of Federal Regulations*, Section 50.67.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Stewart N. Bailey, Senior Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 271 to License No. DPR-70
2. Amendment No. 252 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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Salem Nuclear Generating Station, Unit Nos. 1 and 2

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PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 271
License No. DPR-70

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

   A. The application for amendment filed by PSEG Nuclear LLC on behalf of PSEG Nuclear LLC and Exelon Generation Company, LLC (the licensees) dated April 26, 2004, as supplemented by letters dated September 16, 2004, September 23, 2004, February 25, 2005, and June 13, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 271, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 17, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 271

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following pages of the Appendix A, Technical Specifications, and associated Bases with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
VII	VII
1-2	1-2
1-3	1-3
3/4 7-22	3/4 7-22
3/4 7-23	3/4 7-23
3/4 7-24	3/4 7-24
3/4 7-25	3/4 7-25
B 3/4 7-5c	B 3/4 7-5c
B 3/4 7-5d	B 3/4 7-5d

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 252
License No. DPR-75

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

   A. The application for amendment filed by PSEG Nuclear LLC on behalf of PSEG Nuclear LLC and Exelon Generation Company, LLC (the licensees) dated April 26, 2004, as supplemented by letters dated September 16, 2004, September 23, 2004, February 25, 2005, and June 13, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 252, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 252

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following pages of the Appendix A, Technical Specifications, and associated Bases with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
VII	VII
1-2	1-2
1-3	1-3
3/4 7-18	3/4 7-18
3/4 7-19	3/4 7-19
3/4 7-20	3/4 7-20
B 3/4 7-5c	B 3/4 7-5c
B 3/4 7-5d	B 3/4 7-5d

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 271 AND 252 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PSEG NUCLEAR LLC

EXELON GENERATION COMPANY, LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated April 26, 2004, as supplemented by letters dated September 16, 2004, September 23, 2004, February 25, 2005, and June 13, 2005, PSEG Nuclear LLC (PSEG or the licensee) requested a license amendment for Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem). The licensee requested a full-scope implementation of the Alternate Source Term (AST) described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." As part of implementing the AST, the licensee requested, in part, to (1) increase the allowable leakage rate of the emergency core cooling systems (ECCS) outside containment, and (2) relocate to the Salem Updated Final Safety Analysis Report (UFSAR) the Technical Specification (TS) surveillance requirements (SRs) for the auxiliary building ventilation system (ABVS) exhaust air filtration system (TS 3/4.7.7). The supplements dated September 16, 2004, September 23, 2004, February 25, 2005, and June 13, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards determination as published in the Federal Register on June 22, 2004 (69 FR 34705).

2.0 REGULATORY EVALUATION

The licensee's request was pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, "Accident Source Term," which provides a mechanism for licensed power reactors to replace the traditional source term used in the radiological consequence analyses of design basis accidents (DBAs). Salem's current DBA radiological consequence analyses, other than that for the fuel handling accident (FHA), are based on the source term from Technical Information Document (TID) 14844. In License Amendment Nos. 251 and 232 for Salem Unit Nos. 1 and 2, respectively, the Nuclear Regulatory Commission (NRC or the Commission) staff previously approved selective application of the AST for the FHA.

The staff evaluated the radiological consequences of affected DBAs against the dose criteria specified in 10 CFR 50.67(b)(2). These criteria are 0.25 sieverts (Sv), or 25 rem, total effective

dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 0.25 Sv (25 rem TEDE) at the outer boundary of the low population zone (LPZ), and 0.05 Sv (5 rem TEDE) in the control room (CR). Except where the licensee has proposed a suitable alternative, the staff used the following regulations, regulatory guidance, and standards in its review:

- 10 CFR Part 50.36,"Technical Specifications"
- 10 CFR Part 50.67, "Accident Source Term"
- 10 CFR Part 50, Appendix A,"General Design Criterion [GDC] for Nuclear Power Plants": GDC 19, "Control Room"; GDC 60,"Control of Releases of Radioactive Materials to the Environment"
- RG 1.145,"Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"
- RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors"
- NUREG-0800, "Standard Review Plan": Section 2.3.4,"Short-Term Diffusion Estimates for Accidental Atmospheric Releases;" Section 6.4, "Control Room Habitability Systems;" Section 9.4.5,"Engineered Safety Feature Ventilation System;" and Section 15.0-1, "Radiological Consequence Analyses Using Alternative Source Term"
- Atomic Industrial Forum (AIF) General Design Criteria for Nuclear Power Plants dated October 2, 1967: Criterion 17, "Monitoring Radioactivity Releases"; and Criterion 70, "Control of Releases of Radioactivity to the Environment"
- Generic Letter (GL) 2003-01,"Control Room Habitability."

The staff considered the impact of the proposed changes on the previously-analyzed DBA radiological consequence analysis, and the acceptability of the revised analysis results.

3.0 TECHNICAL EVALUATION

As part of implementing the AST, the licensee proposed several changes to the Salem licensing basis. These changes include the following:

- Credit for the containment sprays in recirculation mode for iodine removal following a loss-of-coolant accident (LOCA).
- Use of control room envelope (CRE) unfiltered in-leakage values based on tracer gas testing, as discussed in the licensee's December 9, 2003, response to GL 2003-01.

- Increase in allowable leakage from the ECCS during normal operation.
- Removal of TS requirements for the ABVS exhaust filtration system, and relocation of the associated SRs to the UFSAR.
- Revision of the requirements in TS 3/4.7.7 to correspond to the assumptions in the radiological consequence analyses.

To justify these changes, the licensee analyzed the radiological consequences of DBAs and events and demonstrated that they satisfy the appropriate criteria.

3.1 Atmospheric Dispersion Factors

The licensee developed atmospheric dispersion factors (χ/Q values) for the range of release points and receptors considered in the DBA radiological consequence analysis. The release points included ground level, the main plant vent, the main steam safety valve (MSSV), the atmospheric relief valve (ARV), and the penetration area pressure relief panel (PAPRP). The receptors included the Salem EAB and LPZ, the Salem CR, and the Hope Creek Nuclear Generating Station (Hope Creek) CR. Hope Creek is adjacent to Salem, so Hope Creek's CR would potentially be impacted by a DBA at Salem.

The licensee used existing χ/Q values, listed in Chapter 2.3.4 of the Salem UFSAR, for the Salem EAB and LPZ. The licensee calculated new χ/Q values for use in evaluating the impact of Salem MSSV, ARV, and PAPRP releases on the Salem CR, and for Salem plant vent releases on the Hope Creek CR.

3.1.1 Meteorological Data

The licensee generated the new χ/Q values from site meteorological data collected between 1988 and 1994 from the site meteorological tower. The tower collects data for both Salem and Hope Creek. The NRC staff has previously accepted the use of these data for determining χ/Q values, and the staff concludes that the data are still valid. Therefore, based on its previous review, the staff concludes that the 1988–1994 database provides an acceptable basis for determining χ/Q values for use in the AST dose analyses.

3.1.2 CR Atmospheric Dispersion Factors

3.1.2.1 Previous Analyses

The licensee used existing χ/Q values to evaluate the impact of the Salem plant vent releases on the Salem CR. These χ/Q values, presented in Table 2, were accepted by the staff in License Amendment Nos. 251 and 232 for Salem Unit Nos. 1 and 2, respectively (Agencywide Documents Access and Management System (ADAMS) Accession No. ML022770181). Based on its previous review, the staff concludes that these χ/Q values are acceptable for use in the DBA dose assessments for AST.

3.1.2.2 New Analyses

The licensee calculated new χ/Q values for the impact of Salem MSSV, ARV, and PAPRP releases on the Salem CR, and also the impact of Salem plant vent releases on the Hope Creek CR. The licensee calculated these values using the guidance provided in RG 1.194. The licensee used ARCON96, the computer code that is described in NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes." RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological consequence analyses.

The licensee ran ARCON96 using the 1988–1994 hourly data from the site meteorological tower. Wind speed and wind direction data from the site tower's 33-foot and 150-foot levels were provided as input, and stability class was calculated using the temperature difference between the 300-foot and 33-foot levels. The resulting χ/Q values are presented in Table 2.

The staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangement, release characterization, source-receptor configuration, meteorological regime, or terrain condition considerations that would preclude use of the ARCON96 model for Salem. The staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and staff practice. The staff performed an independent evaluation, by running ARCON96, and obtained results similar to those of the licensee. On the basis of its review, the staff concludes that the new χ/Q values are acceptable for use in the DBA dose assessments for AST.

a. Salem CR χ/Q Values

The Salem units have a combined CR which normally draws air from two separate air intakes. There are two control room emergency air conditioning systems (CREACS) that each provide emergency filtration and air conditioning services to the combined CR. Each CREACS train takes outside air supplied through two independent ducts equipped with radiation monitors. The systems can be aligned to respond to a DBA with either a single train (Unit 1 or Unit 2) operating by itself or with both trains operating in parallel to pressurize the CR. In performing the CR habitability analysis, the licensee generally assumed that: (1) a one-minute delay is required for CR isolation to be fully operational; (2) only one CREACS train is operational in order to maximize resulting doses; and (3) the CR air intake radiation monitors select the less contaminated air intake during an accident condition. The licensee generated the CR χ/Q values using the CR air intakes as the receptors.

In its May 5, 2005, request for additional information (RAI), the NRC staff asked whether the CR air intake χ/Q values are appropriate for use in modeling unfiltered inleakage, since the "receptor location" may be different. By letter dated June 13, 2005, the licensee responded that there are no potentially unfiltered in-leakage pathways during pressurization mode that would result in χ/Q values higher than those that were used for CR air intake. The licensee's CRE in-leakage tests identified an in-leakage point in the auxiliary building (AB) in an area supplied by the ABVS. The supply to the ABVS is near the CR air intakes but farther from postulated release locations. Consequently, the licensee determined that using CR air intake χ/Q values to model unfiltered inleakage is appropriate.

The licensee modeled two release scenarios, MSSV/ARV releases and PAPRP releases, and two receptor locations (one for each CR air intake). The licensee calculated χ/Q values for each release-receptor combination to identify the more favorable air intake (i.e., the air intake with the lower χ/Q value) for each scenario. The χ/Q values for the more favorable CR air intake were generally used to evaluate filtered air makeup during the pressurization mode, while the χ/Q values for the less favorable (i.e., higher χ/Q value) CR air intake were generally used to evaluate unfiltered air makeup prior to the pressurization mode and the unfiltered inleakage during the pressurization mode.

MSSV and ARV Releases

Both Salem units are Westinghouse 4-loop pressurized water reactors with four steamlines for each unit. Each steamline has a set of five, self-actuated, spring-loaded MSSVs and one pneumatically-operated ARV. The MSSVs only open at or above their setpoint pressures, whereas the ARVs can be manually operated. Due to the clustering of the MSSVs and ARVs within each set, the licensee represented each set as one release point (the closest MSSV). Also, based on the proximity of the MSSV and ARV sets to the CR air intakes, the licensee modeled the two closest MSSV/ARV sets.

To model the MSSV/ARV releases using ARCON96, the licensee assumed the MSSV/ARV releases were ground-level point sources. This is conservative because the MSSV and ARV releases are energetic releases from uncapped and vertically oriented vent pipes, so that additional plume rise will occur due to the buoyancy and momentum effects. The licensee determined that the exit velocity from the MSSV with the lowest pressure setpoint is sonic at 448 meters per sec (1002 miles per hour). Nonetheless, the licensee based the MSSV/ARV release height on the height of the top of the MSSV/ARV vent pipes and the source-to-receptor distance on the closest MSSV vent pipe horizontal distance to the CR intake.

The closest release-receptor combination is the Salem Unit No. 1 MSSV to the Salem Unit No. 1 CR intake. The MSSV release point is approximately 6.38 meters (20.9 feet) horizontally and 5.09 meters (16.7 feet) vertically above the CR intake, for a total distance of approximately 8.16 meters (26.8 feet). RG 1.194 states that any release-receptor combination closer than approximately 10 meters should be addressed on a case-by-case basis. However, due to the conservatism in ignoring the plume rise effects, the staff finds the licensee's modeling to be acceptable.

RG 1.194 allows the ground level χ/Q values calculated with ARCON96 (on the basis of the physical height of the release point) to be reduced by a factor of 5 if (1) the release point is uncapped and vertically oriented, and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed. The sonic exit velocity from the MSSV will be considerably higher than the 10-meter (33-foot) 95th percentile wind speed value of 7.4 meters per second (16.5 miles per hour). Therefore, the licensee reduced the resulting ARCON96 MSSV χ/Q values by a factor of five.

In its RAI dated May 5, 2005, the NRC staff asked whether a stuck-open MSSV or ARV is part of Salem's licensing basis. By letter dated June 13, 2005, the licensee responded that Salem's current licensing basis does not consider stuck-open MSSVs or ARVs. Also, for certain accident scenarios, the operators can cool down the plant by releasing steam from the ARVs. As the transient progresses and secondary side pressure drops, the ARV plume will have a

lower exit velocity. In the same RAI, the staff asked whether such planned steam releases from the ARVs could result in higher doses as compared to MSSV releases (which assume a higher-pressure secondary). The licensee responded that releases from the ARVs will result in lower doses as compared to automatic releases from the MSSVs because the ARVs are located further from the CR intake. The staff noted that the operators would cool down the plant by releasing steam from multiple ARVs. The ARVs are located different distances and directions from the CR intakes, but the licensee models all of them using χ/Q values for the closest MSSV, which the staff finds acceptable.

PAPRP Releases

The PAPRPs are pressure-retaining barriers that are expected to rupture when the pressure in the penetration area exceeds the design value. The panels are vertical, such that the release is horizontal. The licensee assumed that the PAPRPs would rupture and provide a release point during a main steamline break (MSLB).

To model the PAPRP releases using ARCON96, the licensee assumed the PAPRP releases were ground-level diffuse area sources. In its RAI dated May 5, 2005, the NRC staff questioned this methodology for modeling the PAPRP release. By letter dated June 13, 2005, the licensee responded that releases from the PAPRPs were conservatively re-analyzed as point sources rather than diffuse area sources. Since the release is horizontal, the licensee did not take credit for plume rise. The release height was based on the centerline elevation of the PAPRPs and the source-to-receptor distance was based on the straight-line horizontal distance from the center of the PAPRPs to the CR air intake.

b. Hope Creek CR χ/Q Values

The Salem plant vents are on the same line-of-sight between the Salem and Hope Creek plants. The licensee used χ/Q values from the Salem Unit No. 2 plant vent to evaluate the Hope Creek CR. Salem Unit No. 2 is closer to Hope Creek and will have the higher χ/Q values. The Salem plant vent is centered on top of the containment building.

The licensee modeled the Salem Unit 2 plant vent releases using the ARCON96 computer code, assuming the plant vent releases were a ground-level point source. The release height was based on the elevation of the plant vent, and the source-to-receptor distance was based on the straight-line horizontal distance between the vent and the Hope Creek CR air intake.

3.1.3 EAB and LPZ Atmospheric Dispersion Factors

The licensee evaluated offsite doses using the EAB and LPZ χ/Q values provided in the Salem UFSAR, Table 2.3-21. These values, presented in Table 3 of this Safety Evaluation (SE), were calculated using a model described in RG 1.145 and onsite meteorological data from the period 1988 through 1994. Further details on the calculation of the licensee's EAB and LPZ χ/Q values can be found in Salem UFSAR Section 2.3.4, "Short-Term Diffusion Estimates."

The staff reviewed the licensee's use of existing EAB and LPZ χ/Q values and found them to be appropriate. On the basis of this review, the staff concludes that the EAB and LPZ χ/Q values presented in Table 3 are acceptable for use in the DBA offsite dose assessments for AST.

3.1.4 Summary

The licensee identified atmospheric dispersion factors for the range of release points and receptors considered in the AST radiological consequence analyses. The staff has reviewed the atmospheric dispersion factors and their application in the AST analyses, and finds them acceptable.

3.2 Radiological Consequences of DBAs

To support the implementation of an AST, the licensee analyzed the radiological consequences of the following five DBAs:

- LOCA
- MSLB
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor Accident (LRA)
- Rod Ejection Accident (REA)

In addition, the licensee analyzed the radiological consequences of the following two infrequent events:

- Waste Gas Decay Tank (WGDT) Rupture
- Volume Control Tank (VCT) Rupture

The radiological consequence analysis for the FHA was previously approved by the NRC staff in Amendment Nos. 251 and 232 for Salem Unit Nos. 1 and 2, respectively.

The licensee's submittal, as supplemented, reported the results of the radiological consequence analyses for the above DBAs to show compliance with10 CFR 50.67 for offsite doses and compliance with 10 CFR 50, Appendix A, GDC-19 for the CR.

To perform the dose analyses, the licensee used the computer code, "RADTRAD: Simplified Model for RADionuclide Transport and Removal And Dose Estimation," Version 3.02, as described in NUREG/CR-6604. RADTRAD was developed by the Sandia National Laboratories for the NRC. This code estimates transport and removal of radionuclides and radiological consequences at selected receptors. Since the licensee used an NRC code, the staff did not perform a separate calculation. Instead, the staff reviewed the RADTRAD input and output files and confirmed that the input parameters are consistent with the information in this SE.

3.2.1 LOCA

The current LOCA radiological consequence analysis, provided in Salem UFSAR Section 15.4.1, "Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)," is based on the accident source term described in TID-14844. The licensee re-analyzed the LOCA radiological consequences and determined that, after implementation of the proposed TS changes and use of AST, the existing engineered safety systems at Salem continue to provide reasonable assurance that the doses at the EAB, LPZ, and CR will meet the acceptance criteria in 10 CFR 50.67(b)(2).

As part of implementing AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and GDC 19. The licensee assumed that the inventory of fission products is based on the maximum power level of 3,632 megawatts thermal (MWt), which is 1.05 times the current licensed thermal power level of 3,459 MWt.

The staff reviewed the licensee's analyses for the following three potential fission product release pathways:

1. primary containment leakage
2. leakage from ECCS outside containment
3. containment pressure-vacuum relief line release

For potential back-leakage to the refueling water storage tank (RWST), the licensee calculated a back-leakage rate of 100 cubic centimeters per hour (a total of approximately 19 gallons for the 30-day accident period) through the double-isolation valves. The licensee used inspection testing criteria from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section II, to calculate the leakage rate based on pipe diameter and wall thickness. With this leakage rate, the licensee concluded that this release pathway has a negligible contribution to the overall radiological consequences from a LOCA. The NRC staff concurs with this conclusion.

3.2.1.1 Containment Leakage

Containment leakage provides a release path for fission products in the containment atmosphere. For the first 24 hours following the postulated LOCA, the analyses assume the containment leak rate is 0.1 percent by volume per day (percent per day), based on the leak rate specified in the Salem TSs and UFSAR. The leakage rate then decreases to 0.05 percent per day for the duration of the accident (30 days). These leakage rates are used in both the current analyses and AST. Since these assumptions are consistent with the guidelines provided in RG 1.183, the staff finds their use acceptable.

The amount of fission products in the containment atmosphere is reduced by natural deposition (of aerosols) and by operation of the containment spray system (CSS) and containment fan cooler units (CFCUs). For deposition, the licensee used the Powers Simplified Model in RADTRAD at the 10-percentile confidence interval (90-percent probability). This simplified model was derived by Monte Carlo analyses of detailed models of aerosol behavior in the containment under accident conditions. The staff finds that the use of this model in RADTRAD is acceptable. Removal by CSS and CFCUs are addressed in the following sections.

3.2.1.1.1 CSS

The CSS, in conjunction with the CFCUs, is designed to maintain containment pressure and temperature within the design limits and to remove fission products in the containment atmosphere. The CSS consists of two trains, each with a pump, spray headers, and associated valves and piping. Each train of the CSS is independently capable of delivering 2600 gallons per minute (gpm) of borated water from the RWST into 75 percent of the containment atmosphere. The spray pumps are automatically started whenever two out of four high-high containment pressure signals occur or a manual signal is given. The analyses only credit the

operation of one train of the CSS. The CSS initially takes suction from the RWST and is manually aligned to take suction from the containment sump when the RWST water level reaches a low level. The licensee assumed that spray flow starts 90 seconds after initiation of the event, which is conservatively 5 seconds later than 85 seconds assumed in the UFSAR. The RWST water reaches the low level at 48 minutes, and switchover is achieved at 58 minutes. CSS flow is assumed to be interrupted for the 10 minutes during manual realignment. During the recirculation phase, CSS flow is 1900 gpm. The licensee conservatively assumed that CSS flow is terminated at 4 hours. With the exception of the conservative increase in time to initiate CSS, the above CSS parameters were not affected by the implementation of AST. They were previously accepted by the staff and are reflected in Salem UFSAR Section 6.2.2.1, "Containment Spray System."

The analysis considers radioactive iodine in the containment atmosphere in three different forms: elemental, particulate, and organic. Iodine in elemental and particulate forms are removed by the CSS and by deposition in the containment. There is no effective mechanism for removing organic iodine from the containment atmosphere. Iodine removal by the CSS is modeled using parameters that describe the rate of removal (called removal coefficients) and parameters that describe the maximum fraction that can be removed (called decontamination factors, or DFs). Removal coefficients are a function of spray flow rate and the form of iodine being removed. As a result, there are two removal coefficients for each of the injection and recirculation phases: one for elemental iodine and one for particulate iodine. The licensee calculated the iodine removal coefficients in accordance with the methodology described in Section 6.5.2 of the Standard Review Plan (SRP).

For the injection phase, the licensee calculated an iodine removal coefficient of 29 per hour (or a factor of 29 per hour) for elemental iodine. However, the licensee used the upper-limit of 20 per hour, consistent with guidance provided in the SRP. The license calculated a removal coefficient of 4.44 per hour for particulate iodine.

For the recirculation phase, the licensee calculated a removal coefficient for elemental iodine of 14.62 per hour. This removal coefficient was used until the maximum DF of 100 was reached at 2.115 hours into the event. RG 1.183 specifies a maximum DF of 200 for elemental iodine; therefore, the licensee's assumption is conservative. For particulate iodine, the licensee calculated a removal coefficient of 3.24 per hour. This removal coefficient is used until the particulate iodine DF reaches 50, at 3 hours into the event. After this, the licensee used a removal coefficient of 0.324 per hour, consistent with the guidance in RG 1.183.

The staff performed an independent verification of the iodine removal coefficients and DFs used by the licensee. The staff finds the values to be conservative and acceptable.

3.2.1.1.2 CFCUs

The CFCUs are designed to remove heat and fission products from the containment atmosphere. The CFCUs work in conjunction with the CSS to maintain containment pressure and temperature within the design limits. There are five CFCUs, each of which is designed to supply a nominal 110,000 cfm during normal operation and 39,000 cfm during accident operation. Each CFCU has roughing filters and high-efficiency particulate (HEPA) filters, but the licensee's analyses conservatively assumed no fission product removal by the CFCUs.

The licensee conservatively assumed two units (total flow of 78,000 cfm) operate during the post-accident period. Based on the configuration in the containment, 55.45 percent of CFCU air flow (43,000 cfm) is distributed from the unsprayed to sprayed regions. RG 1.183 states that the analysis should assume a mixing rate of two turnovers of the unsprayed to sprayed regions per hour (21,800 cfm per hour), unless a higher flow rate is justified. This licensee's mixing rate (43,000 cfm) exceeds the flow rate criterion in RG 1.183. The previous analyses credited three CFCUs operating. The staff finds that, when crediting two CFCUs, the licensee has adequately justified its use of the higher air mixing (flow) between the sprayed and unsprayed regions, in accordance with RG 1.183.

3.2.1.2 Leakage From ECCS

As part of the AST license amendment, the licensee proposed to increase the allowable leakage from the ECCS outside containment. After the RWST is drained, the ECCS is aligned to take suction from the containment sump, such that contaminated sump water is circulated in the ECCS. At this point, ECCS leakage outside containment provides a path for the release of radionuclides to the environment.

The licensee conservatively assumed that the release of radionuclides due to ECCS leakage begins 20 minutes into the event. Twenty minutes is the earliest possible time to drain the RWST and initiate realignment of the ECCS suction to the containment sump. Consistent with the guidance in RG 1.183, the licensee conservatively assumed that all of the radioiodines released from the reactor coolant system (RCS) are instantaneously deposited in the containment sump water.

In its original application, dated April 26, 2004, the licensee requested to increase the allowable ECCS leak rate to 1 gpm. The pre-AST limit in the Salem TSs is 3790 cubic centimeters per hour (cc/hr), or approximately 1 gallon per hour. The basis for this increase was the licensee's determination that only 4.85 percent of elemental iodine released from the RCS is available for release from leaked ECCS fluid. The licensee had increased this number by an arbitrary factor of 3 to account for uncertainty. The staff did not accept this proposal. Appendix A to RG 1.183 states that the amount of iodine that becomes airborne from leaked ECCS leaked fluid should be assumed to be 10 percent of the total iodine activity (in contrast to the licensee's proposal to only consider elemental iodine), unless a smaller amount can be justified based on the actual sump pH history and ventilation rate. However, the licensee had not presented adequate technical justification for iodine speciation or the arbitrary uncertainty factor. This is reflected in the staff's RAIs dated May 2, 2005.

In response to the staff's RAIs, the licensee recalculated the fraction of total iodine that becomes airborne. By letter dated June 14, 2005, the licensee stated that when the containment sump temperature was above 170 EF, the licensee assumed ECCS leakage will flash until the water temperature decreases to 170 EF. The licensee used the constant enthalpy equation provided in RG 1.183. RG 1.183 guidance is to use 212 EF in the constant enthalpy equation; however, the licensee used 170 EF, which is more conservative. The licensee also used the Salem-specific maximum time-dependent temperature of the sump water (261.49 EF at 1399 seconds into the postulated LOCA) in its calculation. The licensee determined that the fraction of leakage that flashes to vapor (and hence the fraction of leaked iodine that becomes airborne in the AB) is 5.06 percent for these conditions. Further, although flashing terminates when the sump water temperature decreases below 212 EF, the licensee assumed flashing

would continue until the sump temperature reached 170 EF, 16.67 hours into the event. For sump temperature less than 170 EF, the licensee determined the fraction of iodine that becomes airborne to be no greater than 2.0 percent, based on Salem-specific sump pH history and AB ventilation rate. Also, the licensee revised its requested allowable ECCS leak rate to 0.7 gpm (instead of 1 gpm originally requested). The licensee's analyses doubled this leakage to 1.4 gpm, consistent with the guidance provided in RG 1.183. The licensee did not take any credit for iodine dilution or holdup within the buildings, or any credit for ABVS exhaust filtration systems.

In its review of the amount of iodine that becomes airborne (2 percent), the NRC staff considered the justifications proposed by the licensee and performed its own evaluations. In support of its evaluations, the staff visited Salem on June 14, 2005, to verify the design, layout, configuration, and operation of the ECCS in the Salem AB. The staff performed the following five deterministic engineering evaluations specifically applicable to Salem.

(1) ECCS Leakage Pathway

All active components of the ECCS are located outside of the containment and inside of the AB. The ECCS potential leakage sources include valve packing glands, pump shaft seals, flanged connections, and other similar components. The Salem ECCS includes two centrifugal charging pumps, two safety injection pumps, two containment spray pumps, two residual heat removal (RHR) system pumps, and associated valves and piping. The containment spray pump operation is expected to terminate at 4 hours into the postulated LOCA. For a large-break LOCA, the high-pressure injection pumps (charging pumps and safety injection pumps) may be operated for a relatively short period. The RHR components in the RHR rooms will be most actively used during the 30-day period that the radiological consequence analysis covers, and the staff believes that the ECCS leakage from the RHR pumps and RHR components will result in more leakage than the other expected leakage sources.

Most of the ECCS pumps are located in the AB rooms at the 84 foot elevation (EL), except RHR pumps which are located at the 45 foot EL. During its site visit, the staff observed that the pumps are equipped with a built-in drip pan or a measurable (indexed) glass container to collect leaked fluid. These drainage collection devices have hard-piped drain lines that are designed to gravity-drain to the waste holdup tanks (WHTs) in the AB at the 55 foot EL.

Three WHTs have a total storage capacity of 75,000 gallons (25,000 gallons each). During normal plant operation, the WHTs receive liquid radioactive waste from equipment and floor drains. While the first WHT receives liquid radioactive waste, the liquid waste in the second WHT is processed (mixing, sampling, and discharging), and the third tank is in reserve (no liquid waste in the tank). One WHT has adequate volume (25,000 gallons) to receive the requested maximum allowable ECCS leakage, 0.7 gpm, for 25 days out of the 30-day analysis period. The liquid waste in the other WHT(s) can be processed, if needed, to receive leaked ECCS fluid for the remaining 5 days (or longer). The licensee stated during the staff's visit that the WHT liquid waste are always sampled and analyzed for pH prior to processing or discharging, and that the pH has always been near neutral. The WHTs are vented to the plant vent through the AB ventilation system. The AB ventilation system includes a HEPA filter, but the licensee did not take any credit for fission product removal by the HEPA filter.

There are two RHR rooms in the Salem AB. Each RHR room is equipped with a sump with a capacity of approximately 1400 gallons. The ECCS leakage in the RHR room and other cubicle floor drains are gravitationally collected in the sump through closed, hard-piped drain lines. Each sump is equipped with two, redundant safety-related sump pumps and level transmitters that are powered from a separate vital 230-volt power control center. The RHR sump pumps discharge into the WHTs through hard-piped drain lines when the sump water reaches a preset level.

Therefore, the ECCS leakage is not expected to form a pool of water on any floor in the AB (see item 5 below) or to evaporate to dryness. This minimizes or eliminates the amount of potential gaseous elemental iodine release to the AB atmosphere.

(2) ECCS Leakage Water Chemistry

After realignment of the ECCS suction, the ECCS leakage will be contaminated sump fluid. As stated in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plant,” the fission product releases to the containment atmosphere may contain as much as 5 percent of elemental iodine (volatile gaseous form). Once the iodine enters the containment sump, it dissolves in the aqueous environment as cesium iodide. Subsequent iodine behavior, such as transport to the atmosphere, depends on time and water pH.

The licensee’s analysis demonstrates that containment sump water pH will remain greater than seven for at least 30 days following the postulated LOCA. The staff reviewed the licensee’s analysis and concurs with these results (Section 3.4). It follows that the pH of the ECCS leakage will remain greater than seven. NUREG-1465 states that if the pH is maintained at seven or greater, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine, which can become airborne. Therefore, the potential release of radioiodine from the leaked ECCS fluid to the AB atmosphere is minimized or eliminated.

(3) Iodine Deposition in the AB

Gaseous iodine in elemental and particulate forms deposits on surfaces. Elemental iodine deposits by chemical adsorption, while iodine in particulate form deposits through a combination of sedimentation and thermophoretic and diffusiophoretic (molecular) deposition. Basic chemical and physical principals predict this behavior, and many laboratories, experimental facilities, and in-plant measurements have demonstrated this deposition. Elemental iodine is the most reactive and has the largest deposition rate among other iodine species on surfaces. The staff allows credit in radiological consequence analyses for iodine removal by these deposition processes in the reactor containment and main steamlines. During its visit to Salem, the staff noticed large surface areas available for elemental iodine to deposit on the pump room walls and ceilings, pumps and equipment surfaces, fan coolers, and air ducts.

The licensee calculated that the fraction of elemental iodine that deposits on the RHR pump room piping, walls, and ceiling, ranges between 23 to 97 percent. However, the staff did not accept this analysis due to the uncertainty associated with air mixing and ventilation flow rate in the AB. The licensee has withdrawn the deposition fraction that was proposed in the original amendment request. The licensee (and staff) did not credit iodine deposition the radiological consequence analysis. This is conservative in estimating the fraction of the iodine in the leaked ECCS fluid that is released by the AB ventilation system.

(4) Interactions of ECCS leakage fluid with paint, concrete, and impurities affecting the pH

Nitric and carbonic acids may be formed by irradiation of water and interaction with air. The staff considered these effects for the leaked ECCS fluid exposed to air in the open sumps and the WHTs. The staff concluded that the contribution that these acids could have in the pH is minimal, once the fluid is buffered with sodium hydroxide (Section 3.4).

The licensee stated that the ECCS pump cubicles are painted with a surface coat of 10 mil (dry thickness) Phenoline 300 surfacer, and the walls and ceilings are painted with 10 mil (wet film thickness) of Carboline 300 surfacer. The licensee stated during the staff's visit that the paint specification includes a high temperature (212 EF) resistance for thermal degradation of paint. In its response to the staff's RAI, the licensee stated that the individual room coolers, in conjunction with the once-through ventilation air flow (1600 cfm), are designed to maintain the AB area temperatures at or below 110 EF following a LOCA.

The licensee further stated that the pH of the concrete is maintained between seven and eight before the surfaces are applied. NUREG/CR-5950 states that aerosols from limestone concrete will contain basic oxides, and the staff believes that the interactions of ECCS fluid with concrete, if any, will not reduce the water pH.

During the staff's walk-down in the Salem AB, the staff noticed that the pump cubicles and adjacent areas were kept clean with no material stored in the pump rooms and cubicles. In response to the staff's RAI, the licensee stated that combustible and flammable liquids (such as lubricants and cleaning solvents) are used during periods of maintenance, but require use of approved containers for transit and temporary storage. These transient materials are controlled by station housekeeping, chemical use, and fire protection procedures, as applicable, to limit frequency and duration of use in areas outside of the approved storage cabinets. The staff notes that there are lubricants present in the ECCS pumps, motors, and motor operated valves; however, the staff believes any potential leakage of lubricants and their affect on pH are negligible.

Therefore, the staff concludes that the amount of elemental iodine which becomes airborne due to the interaction between leaked ECCS fluid and air, paint, concrete, and impurities in the AB will be negligible.

(5) ECCS Leakage Characteristics and Evaporation

During its visit to Salem, the staff observed ECCS leakage from the shaft seal of an operating charging pump (positive displacement pump) during normal power operation. This pump is in the chemical and volume control system and was running with 24 psig suction pressure, 133 EF water temperature, and 2400 psig discharge pressure. The leakage was collected in a glass container designed to measure its leakage rate. The licensee estimated the observed leakage rate to be 620 cc/hr (2.73E-3 gpm), and stated that the similar leakage on the other unit was approximately 330 cc/hr (1.47E-3 gpm). The current Salem TS limit for ECCS leakage is 3790 cc/hr, and Salem's administrative control limit is 2100 cc/hr. The leakage was in the form of slow water droplets with no flashing or splashing. Similarly, the staff expects little or no flashing or splashing to occur from recirculated sump fluid once the containment sump water temperature decreases to below 170 EF and the containment pressure decreases to approximately atmospheric pressure (approximately 16.67 hours into the postulated LOCA).

(6) Conclusion for Release Fraction

Based on the above evaluations, the staff concludes that the licensee's assumption that 2 percent of the elemental iodine in the leaked ECCS fluid becomes airborne and available for release is sufficient to provide reasonable assurance that the uncertainties associated with elemental iodine behavior are bounded. Therefore, the staff finds that this assumption acceptable.

3.2.1.3 Containment Relief Line Release

The containment pressure vacuum release line is a potential pathway for the release of radionuclides. Following a LOCA and before containment isolation, fission products in the primary coolant may be released through this line. Containment isolation occurs in 5 seconds, which is before the start of the gap release at 30 seconds (assumed for AST). The licensee conservatively assumed the entire RCS inventory is instantaneously released and homogeneously mixed in the containment atmosphere. The licensee assumed a primary coolant iodine activity corresponding to one percent failed fuel (3.58 μCi/gm dose-equivalent iodine 131 (DE I-131)), which is greater than the TS limit of 1.0 μCi/gm DE I-131. The licensee further estimated the volume of containment air released during 5 seconds is 1000 cubic feet. Using these source terms, the licensee calculated the contribution to the total LOCA dose is less than 0.01 percent (4E-4 rem TEDE for CR, 3.34E-5 rem TEDE for the EAB, and 5.67E-6 rem TEDE for the LPZ). The staff verified the resulting doses by a hand calculation and concludes that the radiological consequence for this pathway is negligible.

3.2.1.4 Radiological Consequence of a LOCA

The licensee re-evaluated the postulated LOCA, using the AST, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose criteria specified in 10 CFR 50.67. The results of the licensee's evaluation are provided in Table 1. The major parameters and assumptions used by the licensee, and found acceptable by the staff, are listed in Table 4. The radiological consequences in the CR are described in Section 3.3. The staff found that the EAB, LPZ, and CR doses meet the applicable criteria and are acceptable.

3.2.2 MSLB

The MSLB accident is described in the Salem UFSAR Section 15.4.2, "Major Secondary system Pipe Rupture." For radiological consequence analyses, the MSLB assumes complete severance of the largest main steamline outside containment, since this will bound the consequences of a break inside containment. The faulted steam generator (SG) rapidly depressurizes and releases its contents to the environment. A reactor trip occurs, main steam isolation occurs, safety injection actuates, and a loss of offsite power (LOOP) occurs concurrently with the reactor trip. Since the LOOP renders the main condenser unavailable, the operators cool the plant by releasing steam from the atmospheric relief valves associated with the intact SGs.

Appendix E of RG 1.183 identifies acceptable assumptions for radiological analysis of an MSLB. The radiological releases are from the initial blowdown of the faulted SG, subsequent primary-to-secondary leakage in the faulted SG, and steaming of the intact SGs during cooldown. The licensee assumed that the faulted SG boils dry releasing its entire initial

inventory and entrained radionuclides through the faulted steamline. Primary-to-secondary leakage to the faulted SG is assumed immediately to flash to steam and be released to the environment without holdup or dilution. The leakage in the intact SGs mixes with the bulk water and is released at the assumed steaming rate. Primary-to-secondary leakage is assumed to be the maximum value permitted by TSs, 1 gpm for all four SGs, and is conservatively apportioned between faulted and intact SGs. The primary-to-secondary leakage for the faulted SG is assumed to be 0.35 gpm, with 0.65 gpm split between the other three SGs. Steaming from the intact SGs is assumed to continue for 32 hours. The licensee conservatively assumed no iodine partitioning in either the faulted or the intact SGs (partition coefficient of one).

The licensee stated that no fuel damage is postulated to occur in the MSLB event. This is supported by Salem UFSAR, Section 15.4.2, which states that the departure from nucleate boiling (DNB) ratio limit is not violated for any steamline rupture, assuming the most reactive control rod is stuck in its fully-withdrawn position. Consistent with the guidance in RG 1.183, the licensee assumed the released activity is the maximum reactor coolant activity specified in the TSs. Two radioiodine spiking cases were considered. The first assumed that a radioiodine spike occurred just before the event (pre-incident) and the RCS radioiodine inventory was at the maximum value permitted by the TSs. The second case assumed the event initiated a radioiodine spike (coincident) where the radioiodine was released from the fuel to the RCS at a rate 500 times the normal rate for 8 hours after event initiation.

The licensee re-evaluated the postulated MSLB, using the AST, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines specified in SRP 15.0.1 and dose criteria specified in 10 CFR 50.67. The results of the licensee's evaluation are provided in Table 1. The staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE, and with those stated in the Salem UFSAR. The assumptions found acceptable to the staff are presented in Table 5. The staff found that the EAB, LPZ, and CR doses meet the applicable criteria and are acceptable.

3.2.3 SGTR

The SGTR analysis assumes a complete severance of a single tube in one of the SGs. The primary-to-secondary flow through the ruptured tube contaminates the secondary side, making the radionuclides available for release in subsequent steaming of the SG. A reactor trip occurs, safety injection actuates. The licensee assumes a LOOP, since this renders the main condenser unavailable, and the operators cool the plant by releasing steam to the environment through the ARVs. The licensee assumes that primary-to-secondary flow continues for 30 minutes, at which time the effected SG is isolated and the RCS is depressurized. Appendix F of RG 1.183 identifies acceptable radiological analysis assumptions for an SGTR.

Two iodine-spiking cases are considered. The first assumes a pre-incident spike with the RCS radioiodine inventory at the TS maximum value. The second case assumes the event initiates a coincident radioiodine spike where iodine is released from the fuel at 335 times the normal rate for 8 hours. In addition to the break flow, the licensee assumed there is primary-to-secondary leakage at the maximum value permitted by TSs of 1 gpm. Primary-to-secondary leakage is assumed to be 0.347 gpm into the bulk water of the ruptured SG and 1 gpm total into the bulk water of the unaffected three SGs.

The iodine activity from the primary-to-secondary leakage is assumed to be directly released to the environment and partitioning of iodine is not credited. The licensee conservatively assumed that the iodine activity in the faulted SG water is released to the environment with an iodine partitioning factor of 10 (which is conservative compared to the partitioning factor of 100 provided in RG 1.183).

The radionuclides in the intact SG bulk water are assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. The licensee conservatively assumed that 10 percent (rather than 1 percent given in RG 1.183) of the radionuclides in the unaffected SG bulk water enter the vapor space and are released to the environment. The steam release continues until the RHR system can be used to complete the cooldown at approximately 32 hours.

The licensee re-evaluated the postulated SGTR accident, using the AST, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines specified in SRP 15.0.1 and dose criteria specified in 10 CFR 50.67. The results of the licensee's evaluation are provided in Table 1. The staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Salem UFSAR. The assumptions found acceptable to the staff are presented in Table 6. The staff found that the EAB, LPZ, and CR doses meet the applicable criteria and are acceptable.

3.2.4 LRA

The LRA assumes the instantaneous seizure of a reactor coolant pump rotor, which causes a rapid reduction in the flow through the affected RCS loop. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. As the LOOP renders the main condenser unavailable, the operators cool the plant by releasing steam to the environment. Appendix G of RG 1.183 identifies acceptable radiological analysis assumptions for an LRA.

The licensee conservatively assumed that 5 percent of the fuel rods will experience DNB and release their gap activity into the RCS. This is consistent with Salem UFSAR Section 15.4.5, "Single Reactor Coolant Locked Rotor and Reactor Coolant Pump Shaft Break." A radial peaking factor of 1.65 was applied. The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage. The licensee assumed the leakage rate was 1 gpm for all four SGs for 32 hours. The licensee further assumed that this leakage mixes with the bulk water in the SGs and that the radionuclides become vapor at a rate that is a function of the SG steaming rate and the partition coefficient.

The SG tubes would remain covered by the bulk water, which reduces the release of radionuclides from primary-to-secondary leakage. The licensee conservatively assumed that 10 percent of the radionuclides enter the vapor space (rather than 1 percent given in RG 1.183) and are released to the environment. The steam releases from the SGs continue until 32 hours into the event, when the RHR system can be aligned to complete the cooldown.

The licensee re-evaluated the postulated LRA, using the AST, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines specified in SRP 15.0.1 and dose criteria specified in 10 CFR 50.67. The results of the licensee's evaluation are provided in Table 1. The staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Salem UFSAR. The assumptions found acceptable to the staff are presented in Table 7. The staff found that the EAB, LPZ, and CR doses meet the applicable criteria and are acceptable.

3.2.5 REA

The REA assumes the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a control rod and its drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected to occur due to the resultant reactivity spike, while the failure breaches the reactor pressure vessel head resulting in a LOCA. A reactor trip occurs, safety injection actuates, and a LOOP occurs concurrently with the reactor trip. As the LOOP renders the main condenser unavailable, the operators cool the plant by releasing steam to the environment.

The radiological release is assumed to occur through two separate pathways: (1) release by containment atmosphere leakage; and (2) release of RCS inventory by primary-to-secondary leakage and steaming the SGs. The actual doses are a composite of the two pathways. However, modeling each pathway separately, assuming 100 percent of the radionuclides are available for release from that pathway, and showing an acceptable dose from each pathway would also show that the composite dose would be acceptable. Appendix H of RG 1.183 identifies acceptable radiological analysis assumptions for an REA.

The licensee assumed that 10 percent of the fuel rods fail releasing the radionuclide inventory in the fuel rod gap. The licensee further assumed that 10 percent of the core inventory of radioiodines and noble gases are in the fuel rod gap. A radial peaking factor of 1.75 was applied. In addition, localized heating is assumed to cause 0.25 percent of the fuel to melt, releasing 100 percent of the noble gases and 25 percent of the radioiodines contained in the melted fuel to the containment. For the secondary side release path, 100 percent of the noble gases and 50 percent of the radioiodines contained in the melted fuel are released.

For the containment leakage case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the containment free volume. The licensee assumed that the containment leaks at the TS value of 0.1 percent volume per day for the first 24 hours and 0.05 percent volume per day for days 2 through 30, consistent with the guidance in RG 1.183. The licensee has taken no credit for removal of fission products in the containment due to deposition, the CSS, or the CFCUs.

For the secondary release case, the radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage. Primary-to-secondary leakage is assumed to be the TS maximum of 1 gpm total for four SGs. The licensee assumed that this leakage mixes with the bulk water of the SGs and that the radionuclides become vapor at a rate that is a function of the steaming rate and the partition coefficient. The licensee conservatively assumed that the chemical form of the radioiodine would be 97 percent elemental and 3 percent organic,

consistent with the guidance in RG1.183. The licensee conservatively assumed all noble gasses are immediately released to the SG steam space, and no partitioning of iodine in the SG water (even though the SG tubes would remain covered in this event). These assumptions are also consistent with RG 1.183. The steam release continues until the systems depressurize at approximately 110 seconds.

The licensee re-evaluated the postulated REA, using the AST, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines specified in SRP 15.0.1 and dose criteria specified in 10 CFR 50.67. The results of the licensee's evaluation are provided in Table 1. The staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Salem UFSAR. The assumptions found acceptable to the staff are presented in Table 8. The staff found that the EAB, LPZ, and CR doses meet the applicable criteria and are acceptable.

3.2.6 WGDT and VCT Ruptures

The WGDT and VCT rupture events assume the immediate release of all radioactive gasses stored in these tanks. These tanks are the major sources of gaseous radioactivity during normal operation. The WGDT and VCT rupture event analyses are included in Salem UFSAR Section 15.3, "Condition III - Infrequent Faults." By definition, Condition III occurrences are faults which may occur very infrequently during the life of the station.

The licensee developed WGDT and VCT source terms based on 1 percent fuel cladding defects while operating at 3600 MWt for 497 effective full-power days. The licensee conservatively assumed that the fission product inventory in the WGDT and VCT are released through the plant vent as an instantaneous puff with no credit for hold-up or decontamination. The licensee assumed that the CR ventilation system continued to operate in normal mode (no CR isolation or CR switch to emergency mode) following the release.

The licensee re-evaluated the postulated WGDT and VCT ruptures, using the AST, and concluded that the radiological consequences at the EAB, LPZ, and CR are within the dose guidelines specified in SRP 15.0.1 and dose criteria specified in 10 CFR 50.67. The EAB and LPZ doses are also less than 0.1 rem TEDE specified in 10 CFR 20.1301. The results of the licensee's evaluation are provided in Table 1. The staff found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.0 of this SE and with those stated in the Salem UFSAR. The staff found that the EAB, LPZ, and CR doses meet the applicable criteria and are acceptable.

3.3 CR Habitability

The Salem CRs are located within a common CRE. The CR ventilation system provides cool filtered air to maintain ambient room temperatures. The CREACS provides isolation, pressurization, and filtration of the CRE atmosphere to reduce the radiological doses to plant operators and other personnel. The CREACS is designed to initiate on either a safety injection signal or by CR duct radiation monitors. When initiated by the CR duct monitors, the CREACS selects the less contaminated intake. When initiated on safety injection signal, the CREACS selects the intake associated with the non-accident unit. All events that credit the CREACS

assume initiation is from the CR duct monitors, except for a LOCA which assumes initiation is from the safety injection signal.

CREACS pressurizes the CRE to at least 1/8-inch higher pressure than adjacent areas (including the environment) by providing a total of 2200 cfm (2000 cfm nominal plus 10 percent) of filtered makeup air from the selected air intake. For the first 60 seconds following a postulated DBA, the licensee assumed the maximum design air intake flow rate of 1320 cfm (1200 cfm nominal plus 10 percent) and treated it as unfiltered air in-leakage. The licensee also assumed a LOOP occurs when the reactor trips, and power to the CREACS fans is lost. The CREACS fans are assumed to take 60 seconds to gain full speed to fully pressurize the CR, closing the normal air intake dampers. The makeup and recirculation air filter efficiencies are 95 percent for elemental and organic iodine and 99 percent for particulate iodine. The licensee conservatively used a minimum design air recirculation air flow rate of 5000 cfm.

The revised analysis used CRE unfiltered in-leakage values based on tracer gas testing. As discussed in the licensee's December 9, 2003, response to GL 2003-01, the tracer gas testing was performed from May 31 to June 4, 2003, and for the worst-case conditions the unfiltered inleakage was 90 to 100 cfm. The licensee assumed 150 cfm in the radiological consequence analyses. The licensee reflected this change in UFSAR Table 6.4-3, "Control Area Ventilation System Parameters."

Based on information provided by the licensee, the NRC staff understands that tracer testing showed that the Salem CRE has 96 scfm (±91) of unfiltered in-leakage. Regulatory Position C.1.4 of RG 1.197 provides guidance indicating that it is optional to include the uncertainty for facilities that demonstrate a CRE in-leakage rate less than 100 scfm. Since the tested unfiltered in-leakage rate is less than 100 scfm, the licensee would be consistent with RG 1.197 if it used 96 scfm in the calculations. The licensee assumed 150 scfm, which is greater than the tested value. Therefore, the staff finds that the licensee's use of 150 scfm unfiltered air in-leakage in the dose analysis is acceptable.

The licensee's DBA radiological consequence analyses used 150 scfm unfiltered air in-leakage and concluded that the radiological consequences in the CR are within the dose criteria specified in 10 CFR 50.67. The results are in Table 1.

3.4 Containment Sump Water Chemistry

Maintaining sump water in a neutral or alkaline condition (pH at least seven) prevents dissolved radioactive iodine from being released to the containment atmosphere during the recirculation of containment sump water. Most of the iodine leaves the damaged core in an ionic form which is readily dissolved in the sump water. However, in an acidic environment, some of it becomes converted into elemental form which is much less soluble, causing re-evolution of iodine to the containment atmosphere. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," states that the iodine entering the containment is at least 95 percent cesium iodide (CsI) with the remaining iodine (not less than 1 percent) as iodine plus hydriodic acid. In order to prevent release of elemental iodine to the containment atmosphere after a LOCA, the sump pH has to be maintained equal to or higher than seven.

A variety of acids and bases are produced in containment after a LOCA. The pH value of the containment sump will depend on the chemical species dissolved in the containment sump

water. After a LOCA, most of the sump water comes from the systems containing boric acid, including the RWST, safety injection accumulators, and the RCS. Salem uses sodium hydroxide (NaOH) as a buffer to raise the pH above seven. In addition, the following chemical species are introduced into the containment sump in a post-LOCA environment: hydriodic acid (HI), nitric acid (HNO_3), hydrochloric acid (HCl), and cesium hydroxide (CsOH). CsOH and HI enter the containment directly from the RCS, HCl is produced by radiolytic decomposition of cable jacketing, and HNO_3 is synthesized in the radiation field existing in the containment. The resultant containment sump pH will depend on their relative concentrations and on the buffering action of NaOH.

The licensee used the Electric Power Research Institute's MULTEQ code to determine the containment sump pH 30 days after a LOCA. The calculation uses the volumes and boron concentrations from the RWST, accumulators, and the RCS, as well as the NaOH from the spray additive tank volume and concentration. The code calculates the concentrations of boron and sodium in the containment sump after injection and then calculates the sump pH. It then takes into consideration the radiolytic decomposition of the cable jacketing to calculate how much HCl is produced inside containment. The licensee assumed that all the cable in the cable tray is broken down by radiolysis entering the sump. Once the concentration of HCl is calculated, it is used to determine the new sump pH. The methodology used by the code to calculate the amounts of strong acid generated is consistent with NUREG/CR-5950, "Iodine Evolution and pH Control." The licensee's calculation did not consider the formation of HI, HNO_3 and CsOH. Although HI and HNO_3 are strong acids, the contribution that these acids could have on the pH is minimal. In addition, by not taking credit for the addition of CsOH into containment, which would increase pH, the calculation is more conservative.

The staff reviewed the licensee's methodology and assumptions, and performed hand calculations to verify the resulting pH value after 30 days. The staff's independent verification demonstrated the containment sump pH would remain above seven for at least 30 days. Based on the above, the staff concludes that the iodine will stay in solution as assumed in the licensee's radiological consequence analyses.

3.5 TS Changes

3.5.1 TS 1.10

The licensee proposed to change the definition in TS 1.10, "Dose Equivalent [DE] I-131." The intent of this definition is to ensure that assumptions made in the DBA radiological consequence analyses remain bounding. As such, the specification should have a basis consistent with the analyses. The licensee currently calculates DE I-131 using thyroid dose conversion factors, since the limiting analysis result was the thyroid dose. The AST analyses, however, determine the TEDE, rather than the whole-body dose and thyroid dose as done previously. Therefore, the licensee proposed to use the Federal Guidance Report No.11 (FGR No. 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submission, and Ingestion," and to delete reference to TID-14844. The staff finds the proposed use of FGR No.11 to be consistent with the AST analyses and, therefore, acceptable.

3.5.2 TS 3/4.7.7

The licensee proposed numerous changes to TS Section 3/4.7.7. The purpose of this TS is to specify the equipment that is required to be operable to mitigate the consequences of a DBA (i.e., to support the assumptions in the radiological consequence analyses). As part of implementing the AST, the licensee no longer takes credit for the ABVS exhaust air filtration system (i.e., the charcoal filter and HEPA filters); however, the analyses still credit the operation of the ABVS to (1) maintain the proper environmental conditions for vital plant equipment, and (2) ensure that airborne radioeffluents (such as those from containment leakage or leaked ECCS fluid) are held up in the AB and released through the plant vent. Since the licensee is not crediting the filtration function of the ABVS, the licensee proposed to change the requirements in TS 3/4.7.7 and relocate the requirements for the charcoal and HEPA filtration units to the Salem UFSAR.

The ABVS is a once-through heating and ventilation system, consisting of a supply air subsystem and an exhaust air subsystem. There is a separate system for each Salem unit, with no connection or sharing between the units except for the drumming and boiling area and the AB elevator shaft. The ABVS can maintain design conditions with one of the two 100-percent capacity air supply units and two of the three 50-percent capacity exhaust fans operating. The starting, stopping, and mode of operation of the system are manually controlled from the CR, except the pump room coolers that will auto-start on rising temperatures in their respective pump rooms. 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, the current TS Bases state that both supply fans can be secured and the number of exhaust fans can be reduced to one. There is sufficient capacity with the single exhaust fan to maintain a negative pressure within the AB.

The exhaust air subsystem includes filtration to reduce the release of contamination. For the AST radiological consequence analyses, no credit is taken for post-accident filtration. The system is used to limit releases from the plant, in accordance with 10 CFR Part 20, and all exhaust air from the AB and containment (during purge operation) is processed through HEPA filters. However, if there is indication of excessive radiation levels in the AB, the charcoal filter can be aligned through HEPA filters, as required. The HEPA and charcoal filters are credited to control releases to offsite dose calculation manual limits.

At least two of the three exhaust fans and either one of the two supply fans is aligned during emergency ventilation (emergency plant operations). During a safety injection, all three exhaust fans and one of the supply fans will start. This will maintain a negative boundary pressure while supplying the required cooling to the AB. Should access/egress become difficult with the three exhaust fans running, then one of the exhaust fans can be secured, as provided in the current TS Bases.

The following TS changes were made to reflect the AST. As previously stated, under AST the licensee credits the function of the AVBS to maintain appropriate temperatures and pressures in the AB. However, the licensee no longer takes credit for the ABVS exhaust system filtration functions to mitigate the consequences of DBAs.

1) The title of TS 3/4. 7.7 was changed from “Auxiliary Building Exhaust Air Filtration” to “Auxiliary Building Ventilation System.”

2) The Limiting Condition for Operation (LCO) was revised to state, “At least two supply fans, and three exhaust fans shall be OPERABLE to maintain the auxiliary building at slightly negative pressure.” Also, a “NOTE” was added to state, “The intermittent opening of the auxiliary building pressure boundary causing a loss of negative pressure may be performed under administrative controls.”

3) The applicability was modified from "MODES 1, 2, 3, and 4” to "At all times.”

4) The action statements were modified to reflect the availability of the supply or exhaust fans (instead of the HEPA and charcoal filters) and to reflect different requirements depending on the plant Mode. Specific changes are as follows:

- Existing Actions a), b), d), and e), all of which relate to filtration, were deleted.

- Existing Action c) was retitled a).

- New Action b) was added to state, “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.”

- New Action c) was added to state, “With the auxiliary building pressure not maintained slightly negative, restore the building to slightly negative pressure within the next 4 hours or "be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.”

- New Action d) was added to state, "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.”

- New Action e) was added to state,"With the auxiliary building pressure not maintained slightly negative, suspend all operations involving radioactive gaseous releases via the auxiliary building immediately.”

5) The surveillance requirements were changed such that only the operation of the supply and exhaust fans are tested. The requirements related to the filtration were relocated to the Salem UFSAR. The specific changes are as follows:

- The first sentence is revised to state,"The above required auxiliary building ventilation system shall be demonstrated OPERABLE,” and reference to “exhaust air filtration” is deleted.

- Existing SR 4.7.7.b (including SR 4.7.7.b.1 through SR 4.7.7.b.5), SR 4.7.7.c, SR 4.7.7.d (including SR 4.7.7.d.1 and SR 4.7.7.d.2), SR 4.7.7.e, and SR 4.7.7.f were deleted and relocated to the Salem UFSAR.

- Existing SR 4.7.7.a was renamed SR 4.7.7.b and revised to state, "At least once per 31 days by starting each fan, from the CR, and verifying that each fan operates for at least 15 minutes."

- New SR 4.7.7.a was added to state, "At least once per 12 hours by verifying negative pressure in the auxiliary building."

- New SR 4.7.7.c was added to state, "At least once per 18 months by verifying that the system starts following a Safety Injection Test Signal."

By letter dated May 5, 2005, the staff asked the licensee to justify (1) the removal of the ABVS exhaust air filtration system from the requirements of TS 3/4.7.7, and (2) the new requirement to maintain negative pressure in the AB. By letter dated June 13, 2005, the licensee responded that these TS changes are based on the criteria in 10 CFR 50.36, as follows:

1. 10 CFR 50.36 Criteria (c) (3) states "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier". The corresponding Loss of Coolant Accident (LOCA) calculations performed to support this amendment request do not credit the ABVS filtration system. Regardless, PSEG proposes to relocate the charcoal/HEPA filtration surveillance requirements (SRs) to the UFSAR in order to maintain compliance with the occupational exposure guidelines of 10 CFR 20. At this time, PSEG does not intend to remove the filtration system and the use of the filtration system remains an option for plant operators, as reflected in Salem Emergency Procedures, following a design basis accident. Should removal of the filtration system be considered in the future, the provisions of 10 CFR 50.59 will be followed.

2. 10 CFR 50.36 Criteria (c) (4) states "A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

   As stated in page 62 of PSEG's submittal, the ABVS filtration system is not modeled in the Salem Units 1 and 2 Probabilistic Risk Assessment (PRA). PSEG has determined that the ABVS filtration is no longer required for accident mitigation and is therefore not on the primary success path for any DBA. As such, the need for maintaining these TSs LCOs and SRs has been eliminated. The proposed changes in the TS are acceptable and are consistent with the maintenance of public health and safety.

With respect to the new requirements, the licensee's June 13, 2005, letter provided the following justification:

> As described in Salem UFSAR Section 9.4.2.1, the Auxiliary Building Ventilation System (AVBS) operates to limit the average temperature of the Auxiliary Building to 110 EF or less, and to maintain the Auxiliary Building boundary at a

slight negative pressure. Auxiliary Building equipment has been evaluated to operate at area temperatures to 110 EF. Individual room coolers, in conjunction with the once-through ventilation air, are designed to limit the ambient room temperature at the vital pumping equipment. These temperatures help to assure long-term and reliable operation of the pumps, motors, controls and instrumentation and accessibility to this equipment for maintenance as required.

Maintaining the technical specifications for the Auxiliary Building Ventilation falls under the criteria of 10 CFR 50.36 which states in Criterion 1, “Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary”. As proposed, Technical Specification 3/4.7.7 for Auxiliary Building Ventilation contains equipment that must be OPERABLE following a Design Basis Accident since ECCS leakage outside containment is assumed and must be contained within the bounds of the Auxiliary Building (slightly negative pressure) and released via the plant vent which has installed accident range radiation monitors to alert the unit operators and accident response personnel of the amount of radioactivity released and to ensure that the accident dose assumption (atmospheric dispersion factors, χ/Q) values are maintained in accordance with design basis assumptions.

The licensee has revised the current licensing bases for the ABVS. The licensee credits the ABVS exhaust system filtration for maintaining releases within the guidelines of 10 CFR Part 20 during normal operation, and maintains the ability to use this system following an event, but does not credit the system for mitigating the consequences of a DBA. Therefore, the staff concludes that Salem continues to conform to the requirements of 10 CFR 50.67 for offsite doses and 10 CFR Part 50, Appendix A, GDC 19 for the CR. Acceptability of the current design of ABVS is unchanged and meets the intent of specific GDC and regulatory guides, stated in SRP Section 9.4.5, that are applicable to Salem. The system continues to conform with GDC 60, “Control of Releases of Radioactive Materials to the Environment,” with respect to the capability of the system to suitably control the release of gaseous radioactive effluents to the environment. Also, the licensee will relocate the ABVS exhaust filtration system requirements to the UFSAR, where future changes are required to comply with the requirements of the 10 CFR 50.59 process.

The proposed TS 3/4.7.7 contains requirements for equipment that must be operable following a DBA. The ECCS leakage outside containment that is assumed in the accident dose analysis must be contained within the AB and released by the monitored plant vent. This alerts the operators and accident response personnel of the amount of radioactivity released, and also ensures that the accident dose assumptions (χ/Q values) are maintained. Therefore, the NRC staff finds the requirements of Criterion 17, “Monitoring Radioactivity Releases,” and Criterion 70, “Control of Releases of Radioactivity to the Environment,” of the AIF GDC for nuclear power plants continue to be met.

Therefore, based on the above review, the staff finds that the above proposed TS changes are acceptable.

3.6 TS Bases and UFSAR Changes

In addition to the above TS changes, the licensee proposed to change the TS Bases and the Salem UFSAR. These changes correspond to the TS changes and the changes that were made to the plant licensing basis for AST.

The changes to the Bases for TS Section 3/4.7.7 reflect that the ventilation system is credited for maintaining negative pressure in the AB. Information related to the AVBS exhaust filtration system, including system lineups and SRs, has been deleted.

The UFSAR changes reflect the AST analyses and include the information that was relocated from the TSs. Table 6.2-9 was revised to include containment spray flow rates and areas of coverage, because the AST analyses take credit for iodine removal by the containment spray system. Table 6.4-3 was revised to specify the unfiltered inleakage in the CREACS, consistent with the inleakage values used in the AST analyses. Section 9.4.2 was revised to discuss the revised design basis and use of the ABVS exhaust filtration system, as discussed above, and Section 9.4.2.4 includes the SRs that were relocated from TS 3/4.7.7. New Section 9.4.8 lists references for Section 9.4. Chapter 15 descriptions were revised to reflect the AST assumptions and results, and Table 15.4-5B was revised to reflect the allowable ECCS leakage.

The staff reviewed the proposed changes and concludes that they are consistent with the radiological consequence analyses performed for AST and are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. The State official responded by letter dated August 10, 2005. The State official had no technical comments, but was concerned that the licensee could use the provisions in 10 CFR 50.59 to remove the ABVS exhaust ventilation system HEPA and charcoal filters without prior NRC approval. The licensee committed to place a requirement in the Salem commitment tracking system to notify the State official if they plan to remove the filtration. This commitment satisfied the State official's concern.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments changed requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changed SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding 69 FR 34705. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

As described above, the staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological consequences of the proposed full implementation of an AST and the TS changes requested. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 of this SE. The staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The staff also finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and CR doses will comply with these criteria. The staff finds reasonable assurance that Salem, as modified by this license amendment, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters. Therefore, the staff finds the proposed changes to the TSs and the implementation of an AST to be acceptable.

This licensing action is considered a full implementation of the AST. With this approval, the previous accident source term in the Salem design basis is superseded by the AST proposed by the licensee. The previous offsite and CR accident dose criteria expressed in terms of whole body, thyroid, and skin doses are superseded by the TEDE criteria of 10 CFR Part 50.67 or fractions thereof, as defined in SRP 15.0.1. All future radiological accident analyses performed to show compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as defined in the Salem design basis, and modified by the present amendment.

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

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Date: February 17, 2006

Table 1

Radiological Consequences Expressed as TEDE [1] (rem)

Design Basis Accidents	EAB [2]	LPZ [3]	Control Room (CR)
Loss-of-Coolant Accident	4.08	1.35	4.17
Dose criteria[4]	25	25	5.0
Main steamline break accident [4]	8.68E-2	2.88E-2	1.27E-1
Dose criteria	2.5	2.5	5.0
Main steamline break accident [5]	5.28E-1	1.87E-1	8.93E-1E-1
Dose criteria	25	25	5.0
Steam generator tube rupture [4]	9.70E-2	1.40E-2	1.40E-1
Dose criteria	2.5	2.5	5.0
Steam generator tube rupture [5]	3.20E-1	4.30E-2	8.90E-1
Dose criteria	25	25	5.0
Locked rotor accident	1.0	3.03E-1	1.88
Dose criteria	2.5	2.5	5.0
Rod ejection accident	6.64E-1	2.03E-1	1.06
Dose criteria	6.3	6.3	5.0
Waste Gas Decay Tank Rupture	2.16E-2	4.79E-3	3.55 [6]
Dose criteria	2.5	2.5	5.0

[1] Total effective dose equivalent
[2] Exclusion area boundary
[3] Low population zone
[4] Accident initiated iodine spike
[5] Pre-accident iodine spike
[6] No credit for CR isolation

Table 2

Control Room Atmospheric Dispersion Factors

Release Pathway

Salem Main Steam Safety Valves
Salem Atmospheric Relief Valves

Accidents

Main Steam Line Break (intact steam generator releases)
Reactor Coolant Pump Locked Rotor
Rod Ejection (secondary system releases)
Steam Generator Tube Rupture

Time Interval (hrs)	Salem Control Room χ/Q Values (sec/m^3)	
	Less Favorable Control Room Air Intake[(a)]	More Favorable Control Room Air Intake[(b)]
0 – 2	$1.57\times10^{!2}$	$6.98\times10^{!4}$
2 – 8	$1.13\times10^{!2}$	$5.66\times10^{!4}$
8 – 24	$4.24\times10^{!3}$	$2.38\times10^{!4}$
24 – 96	$3.08\times10^{!3}$	$1.65\times10^{!4}$
96 – 720	$2.26\times10^{!3}$	$1.32\times10^{!4}$

[(a)]For the main steamline break (intact steam generator), reactor coolant pump locked rotor, and steam generator tube rupture accidents, the χ/Q values for the less favorable control room air intake are used to evaluate unfiltered air makeup prior to the pressurization mode and unfiltered inleakage during the pressurization mode. For the rod ejection accident, the 0!2 hour χ/Q value for the less favorable control room air intake is used to evaluate the entire 110-second secondary system release; that is, no credit is taken for the control room emergency air conditioning system (CREACS) ability to align with the less contaminated air intake.

[(b)]For the main steamline break (intact steam generator), reactor coolant pump locked rotor, and steam generator tube rupture accidents, the χ/Q values for the more favorable control room air intake are used to evaluate filtered air makeup during the pressurization mode.

Table 2 (cont'd)

Control Room Atmospheric Dispersion Factors

Release Pathway

Salem Penetration Area Pressure Relief Panels

Accidents

Main Steamline Break (faulted steam generator releases)

Time Interval (hrs)	Salem Control Room χ/Q Values (sec/m^3)	
	Less Favorable Control Room Air Intake[c]	More Favorable Control Room Air Intake[d]
0 – 2	$6.17\times10^{!3}$	$2.15\times10^{!3}$
2 – 8	$4.19\times10^{!3}$	$1.70\times10^{!3}$
8 – 24	$1.57\times10^{!3}$	$6.65\times10^{!4}$
24 – 96	$1.04\times10^{!3}$	$4.60\times10^{!4}$
96 – 720	$8.13\times10^{!4}$	$3.52\times10^{!4}$

[c]The χ/Q values for the less favorable control room air intake are used to evaluate unfiltered air makeup prior to the pressurization mode and unfiltered inleakage during the pressurization mode.

[d]The χ/Q values for the more favorable control room air intake are used to evaluate filtered air makeup during the pressurization mode.

Table 2 (cont'd)

Control Room Atmospheric Dispersion Factors

Release Pathway

Salem Plant Vent

Accidents

Loss of Coolant
Rod Ejection (containment leakage releases)
Waste Gas Decay Tank Rupture
Volume Control Tank Rupture

Time Interval (hrs)	Salem Control Room χ/Q Values (sec/m^3)		Hope Creek Control Room χ/Q Values (sec/m^3)
	Less Favorable Control Room Air Intake[e]	More Favorable Control Room Air Intake[f]	
0 – 2	$1.78\times10^{!3}$	$8.84\times10^{!4}$	$4.86\times10^{!5}$
2 – 8	$1.31\times10^{!3}$	$6.60\times10^{!4}$	$3.35\times10^{!5}$
8 – 24	$5.22\times10^{!4}$	$2.64\times10^{!4}$	$1.34\times10^{!5}$
24 – 96	$3.77\times10^{!4}$	$1.93\times10^{!4}$	$9.59\times10^{!6}$
96 – 720	$3.17\times10^{!4}$	$1.62\times10^{!4}$	$7.63\times10^{!6}$

[e]For the loss of coolant and rod ejection (containment leakage releases) accidents, the χ/Q values for the less favorable control room air intake are used to evaluate unfiltered air makeup prior to the pressurization mode and unfiltered inleakage during the pressurization mode. For the waste gas decay tank rupture and volume control tank rupture accidents, the 0 ! 2 hour χ/Q value for the less favorable control room air intake is used to evaluate the entire instantaneous puff release; that is, no credit is taken for the CREACS ability to align with the less contaminated air intake.

[f]For the loss of coolant and rod ejection (containment leakage releases) accidents, the χ/Q values for the more favorable control room air intake are used to evaluate filtered air makeup during the pressurization mode.

Table 3

Exclusion Area Boundary and Low Population Zone Atmospheric Dispersion Factors

Receptor	Time Interval (hrs)	χ/Q Value (sec/m^3)
Exclusion Area Boundary	0 - 2	$1.30\times10^{!4}$
Low Population Zone	0 – 2	$1.86\times10^{!5}$
	2 – 8	$7.76\times10^{!6}$
	8 – 24	$5.01\times10^{!6}$
	24 – 96	$1.94\times10^{!6}$
	96 – 720	$4.96\times10^{!7}$

Table 4

Parameters and Assumptions Used in Radiological Consequence Calculations for Loss-of-Coolant Accident

Parameter	Value
Reactor power	3632 MWt
Containment volume	2.62E+6 ft^3
Sprayed area	1.965E+6 ft^3
Unsprayed area	6.55E+5 ft^3
Containment leak rates	
0 to 24 hour	0.1% per day
24 to 720 hours	0.05% per day
Containment mixing rates	
Sprayed to unsprayed	4.3251E+4 cfm
Unsprayed to sprayed	4.3251E+4 cfm
Aerosol removal rates by containment spray (per hour)	
Injection Phase (20 to 48 minutes)	4.44
Recirculation Phase	
58 minutes to 3 hours	3.24
3 to 4 hours	0.324
Elemental iodine removal rates by spray (per hour)	
Injection Phase (20 to 48 minutes)	20
Recirculation Phase	
58 minutes to 2.115 hours	14.62
2.115 to 4 hours	0
Containment sump volume	4.39E+4 ft^3
Emergency core cooling system leak rates	
Time	Rates
0 to 20 minutes	0
20 minutes to 30 days	0.7 gpm (1.4 gpm used)
Iodine partition factors	2 to 5.06%
Release points	
Containment leakage	plant vent
ECCS leakage	plant vent

Table 5

Parameters and Assumptions Used in Radiological Consequence Calculations for Main Steamline Break Accident

Parameter	Value
Pre-incident iodine spike activity	60 µCi/gm dose equivalent I-131
Coincident spike appearance rate, based on	
Reactor coolant systems (RCS) letdown flow rate, gpm	165
RCS letdown demineralizer efficiency	100
Coincident spike multiplier	500
Iodine spike duration, hrs	8
Primary-to-secondary leakage per SG, gpd	500
Duration, hours	32
Liquid Masses	
RCS	2.5E+8 gm
Steam Generator (SG) (each)	1.911 ft^3
Steam release from faulted SG, lbm	
0 to 2 hours	128,000
2 to 8 hours	0
Steam release from intact SGs, lbm	
0 to 2 hours	5.00E+5
2 to 8 hours	4.52E+5
8 to 32 hours	2.00E+6
Steam iodine partition coefficient in SGs	
Faulted SG (elemental and organic)	1.0
Unaffected SG	1.0
Elemental	1.0
Organic	1.0
Release points	penetration area pressure relief panels

Table 6

Parameters and Assumptions Used in Radiological Consequence Calculations for Steam Generator Tube Rupture Accident

Parameter	Value
Pre-incident iodine spike activity	60 µCi/gm dose equivalent I-131
Coincident spike appearance rate, based on	
RCS letdown flow rate, gpm	165
RCS letdown demineralizer efficiency	100
Coincident spike multiplier	335
Iodine spike duration, hrs	8
Primary-to-secondary leakage per SG, gpd	500
Duration, hours	32
Liquid Masses	
RCS	2.5E+8 gm
SG (each)	1.19E+5 lbm (Unit 1) 1.28E+5 lbm (Unit 2)
Steam release from faulted SG, lbm	
0 to 0.5 hours	5.65E+4 lbm (flashed)
Steam release from intact SGs, lbm	
0 to 2 hours	4.65E+5
2 to 8 hours	1.055E+6
8 to 24 hours	1.50E+6
24 to 30 hours	4.77E+5
30 to 32 hours	1.50E+5
Steam iodine partition coefficient in SGs	
Faulted SG	1.0
Unaffected SG	10
Release points	main steam safety valves atmospheric relief valves

Table 7

Parameters and Assumptions Used in Radiological Consequence Calculations for Locked Rotor Accident

Parameter	Value	
Fraction of failed fuel	0.05	
Fraction of Core Inventory in Gap		
Kr-85	0.10	
I-131	0.08	
Alkali metals	0.12	
Other noble gases / iodines	0.05	
Iodine speciation	CNMT	Secondary
Aerosol	0.95	0
Elemental	0.0485	0.97
Organic	0.0015	0.3
Primary-to-secondary leakage per SG, gpd	500	
Primary-to-secondary leakage duration, hours	32	
Steam partition coefficient in SGs	0.1	
Steam release from all 4 SGs, lbm		
0 to 2 hours	6.55E+5	
2 to 8 hours	5.40E+5	
8 to 32 hours	2.40E+6	
Release points	main steam safety valves atmospheric relief valves	

Table 8

Parameters and Assumptions Used in Radiological Consequence Calculations for Control Rod Ejection Accident

Parameter	Value	
Radial peaking factor	1.65	
Fraction of rods that exceed DNB (melted fuel)	0.25	
Gap fraction, all nuclide groups	0.10	
Melt isotopic composition		
	CNMT	SG
Noble gases	1.0	1.0
Iodine	0.25	0.5
Iodine species fraction	CNMT	SG
Particulate/aerosol	0.95	0
Elemental	0.0485	0.97
Organic	0.0015	0.03
Containment free volume, ft^3	2.62E+6	
Containment Sprays	Not credited	
Containment release		
0-24 hours, %/day	0.1	
24-720 hours, %/day	0.05	
Containment Particulate deposition 1/hr	0.023	
Duration of release, days	30	
Primary-to-secondary leakage per SG, gpd	500	
Primary-to-secondary leakage duration, hours	8	
Steam generator mass (each)	1.19E+5 lbm (Unit 1) 1.28E+5 lbm (Unit 2)	
Steam partition coefficient in SGs	1	
Steam release from 4 SGs, lbm		
0- 110 seconds	5.12E+5	
Release point		
Containment leakage	containment	
Secondary	atmospheric relief valves	