October 10, 2002

Mr. Harold W. Keiser
Chief Nuclear Officer & President
PSEG Nuclear LLC - X04
Post Office Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENT RE: REFUELING OPERATIONS - FUEL DECAY TIME PRIOR TO COMMENCING CORE ALTERATIONS OR MOVEMENT OF IRRADIATED FUEL (TAC NOS. MB5488 AND MB5489)

Dear Mr. Keiser:

The Commission has issued the enclosed Amendment Nos. 251 and 232 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 28, 2002, as supplemented on August 15, August 16, and October 2, 2002.

These amendments change the Salem TS requirements for Fuel Decay Time prior to commencing movement of irradiated fuel. TS Limiting Condition for Operation 3/4.9.3, "Decay Time," is revised to allow fuel movement in the containment to commence 100 hours after the reactor has become subcritical between October 15th through May 15th. Should refueling occur between May 16th and October 14th, the current 168 hours decay time limit will remain in place. These requirements are valid through the year 2010.

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/

Robert J. Fretz, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosures:
1. Amendment No. 251 to License No. DPR-70
2. Amendment No. 232 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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PDI-2 Reading	RFretz	ACRS	SLittle	GHill(4)
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ACCESSION NUMBER: **ML02770181** ATTACHMENTS: **ML** PACKAGE: **ML**

* SE Input provided. No major changes made. ** See previous concurrence

OFFICE	PDI-2/PM	PDI-2/LA	SPLB**	SPSB*	OGC**	PDI-2/SC(A)
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DATE	109/02	10/9/02	10/07/02	09/12/02	09/08/02	10/9/02

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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. 251
License No. DPR-70

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

   A. The application for amendment filed by the PSEG Nuclear LLC and Exelon Generation Company, LLC (the licensees) dated June 28, 2002, as supplemented on August 15, August 16, and October 2, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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. 251, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by REnnis for/

James W. Andersen, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 10, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 251

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Replace the following pages of the Appendix A, Technical Specifications, 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
3/4.9.3	3/4 9-3
B 3/4.9.3	B 3/4 9-3

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. 232
License No. DPR-75

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

   A. The application for amendment filed by the PSEG Nuclear LLC and Exelon Generation Company, LLC (the licensees) dated June 28, 2002, as supplemented on August 15, August 16, and October 2, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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. 232, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by REnnis for/

James W. Andersen, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 10, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 232

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Replace the following pages of the Appendix A, Technical Specifications, 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
3/4.9.3	3/4 9-3
B 3/4.9.3	B 3/4 9-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 251 AND 232 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 June 28, 2002, as supplemented on August 15, August 16, and October 2, 2002, PSEG Nuclear LLC (PSEG/the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2 (Salem), Technical Specifications (TSs). The proposed amendments requested revision to TS requirements for Fuel Decay Time prior to commencing movement of irradiated fuel. TS Limiting Condition for Operation (LCO) 3/4.9.3, "Decay Time," would be revised to allow fuel movement in the containment to commence 100 hours after the reactor has become subcritical between the dates of October 15th through May 15th. Should refueling occur between May 16th and October 14th, the current decay time limit of 168 hours will remain in place. These requirements are valid through the year 2010.

The proposed amendment also replaces the current accident source term used in selected design basis radiological analyses with an alternative source term (AST) pursuant to Section 50.67 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.67), “Accident Source Term.” The licensee requested a selective implementation of the AST limited to fuel-handling accident (FHA) analyses.

PSEG provided additional information in letters dated August 15, August 16, and October 2, 2002. The August 16, 2002, supplement was provided in response to the staff’s request for additional information dated August 12, 2002. The August 15, August 16, and October 2, 2002, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The following specific TS changes were requested:

TS 3/4.9.3, “Decay Time”

Replace LCO stating the reactor shall be subcritical for at least 168 hours applicable during movement of irradiated fuel in the reactor vessel with:

3.9.3 The reactor shall be subcritical for at least:

a. 100 hours - Applicable through year 2010.
b. 168 hours

APPLICABILITY: Specification 3.9.3.a - From October 15th through May 15th, during movement of irradiated fuel in the reactor pressure vessel.

Specification 3.9.3.b - From May 16th through October 14th, during movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than the required time, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

Modify associated surveillance requirements to verify the reactor has been subcritical for the required period of time prior to movement of irradiated fuel in the reactor vessel.

Bases for TS 3/4.9.3, "Decay Time"

Add the following information to the bases (Unit No. 1):

The 100-hour decay time is consistent with the assumptions used in the fuel handling accident analyses and the resulting dose calculations using the Alternative Source Term described in [Regulatory] Guide 1.183.

The minimum requirement for reactor subcriticality also ensures that the decay time is consistent with that assumed in the Spent Fuel Pool cooling analysis. Delaware River water average temperature between October 15th and May 15th is determined from historical data taken over 30 years. The use of 30 years of data to select maximum temperature is consistent with [Regulatory] Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants".

A core offload has the potential to occur during both applicability time frames. In order not to exceed the analyzed Spent Fuel Pool cooling capability to maintain the water temperature below 180 °F, two decay time limits are provided. In addition, PSEG has developed and implemented a Spent Fuel Pool Integrated Decay Heat Management Program as part of the Salem Outage Risk Assessment. This program requires a pre-outage assessment of the Spent Fuel Pool heat loads and heatup rates to assure available Spent Fuel Pool cooling capability prior to offloading fuel.

Similar language was proposed for the Salem Unit No. 2 TS Bases.

2.0 REGULATORY EVALUATION

2.1 Spent Fuel Pool

Salem has two separate and independent spent fuel pools (SFPs), with one pool associated with each operating reactor. The proposed amendment involves a change in minimum decay time that increases the maximum decay heat generation that could be placed within each SFP during the first several days of a refueling outage. Under otherwise identical conditions, a larger decay heat generation rate leads to a higher peak SFP temperature with the cooling systems in operation and a greater rate of temperature increase if the cooling system fails.

Decay heat is removed from each SFP by the spent fuel cooling system. Each spent fuel cooling system consists of one seismically qualified cooling train, which includes two full-capacity pumps and one heat exchanger. The pumps are supplied power from the vital electrical distribution system that is backed by the emergency diesel generator. In the event one cooling system fails, the remaining system can be periodically swapped between the two SFPs to provide intermittent cooling to both pools. Heat is removed from the spent fuel cooling system heat exchangers by the safety-related component cooling water system.

Section 3.1 of the Salem Updated Final Safety Analysis Report (UFSAR) describes that the plant has been designed to meet the intent of General Design Criteria contained in Appendix A to 10 CFR Part 50. Criterion 61 specifies, in part, that fuel storage systems shall be designed with:

(1) a residual heat removal system capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and

(2) the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

Section 9.1.3 of the UFSAR describes that the spent fuel cooling system provides cooling to remove residual heat from the fuel stored in the SFP. The system is designed with reliability and redundancy to assure continued heat removal under normal, abnormal, and accident conditions. Section 9.1.3 of the UFSAR also describes SFP temperature limits of 149 °F when both heat exchangers are available and 180 °F when only one spent fuel cooling system heat exchanger is available to cool both pools. The SFP structure, including the liner, has been evaluated for performance of its design function at temperatures of 180 °F. These descriptions provide criteria applicable to evaluation of proposed increases in decay heat generation within the SFP.

2.2 Alternate Source Term - Fuel Handling Accident

In December 1999, the Nuclear Regulatory Commission (NRC) issued a new regulation, 10 CFR 50.67, "Accident Source Term," which provided a mechanism for licensees of power reactors to voluntarily replace the traditional accident source term used in its design-basis accident (DBA) analyses with ASTs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an AST to apply for a license amendment, and requires that the application contain an evaluation of the consequences of DBAs. The present amendment request addresses these requirements in proposing selectively to use an AST in evaluating the offsite and control room radiological consequences of an FHA. This re-analysis involved several changes in selected analysis assumptions including revised values for atmospheric dispersion values for the control room outside air intakes.

3.0 TECHNICAL EVALUATION

3.1 Spent Fuel Pool

The proposed change to TS 3.9.3, "Decay Time," which reduces the minimum decay time from 168 hours to 100 hours, will result in the potential for increased decay heat loads within the SFP. In its license amendment request dated June 28, 2002, PSEG described a conservative evaluation of SFP cooling capability and administrative controls implemented through Salem's Integrated Decay Heat Management Program. As attachments to its letter dated October 2, 2002, PSEG provided Calculation S-C-SF-MEE-1679, Revision 0, "SFP Cooling System Capability with Core Offload Starting 100 Hours after Shutdown," and non-proprietary and proprietary sections of its Critical Software Document for the Crosstie (HOLTEC International) software. This calculation provided the basis for the evaluation of SFP cooling capability, and the Crosstie software is used for the Integrated Decay Heat Management Program.

Calculation S-C-SF-MEE-1679 evaluated the SFP peak temperature by analysis using NRC Branch Technical Position ASB 9-2 to determine decay heat load and scaled heat exchanger performance from the manufacturer's design performance values. The analyses used the following assumptions:

(1) A full core offload begins at the minimum decay time and is completed 44 hours later.

(2) The total decay heat load of 44 MBTU/hour is based on a full core discharge for refueling plus a background decay heat load of 6.8 MBTU/hour representative of discharges from 18-month operating cycles expected prior to the Salem Unit No. 1 refueling planned for the year 2010.

(3) The component cooling water temperature is 71 °F based on the 30-year average Delaware River water temperature of 63 °F for the month of October, which has the highest monthly average river water temperature of the months between October and May.

(4) A single SFP cooling train is in service at its maximum SFP cooling water flow rate of 2,500 gallons per minute (gpm) with heat exchanger fouling and tube plugging at or above their design values.

(5) Evaporative cooling and other passive heat removal paths were neglected.

Using the above methods and assumptions, PSEG calculated a peak SFP temperature of 161 °F.

The NRC staff performed independent calculations of decay heat load and heat exchanger performance to verify the accuracy of the analyses provided by PSEG. The decay heat load calculations used the method described in Branch Technical Position ASB 9-2 from NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," and the heat exchanger performance evaluation used the temperature effectiveness method and heat exchanger performance data from the Salem UFSAR. These independent calculations, with consideration for the differing analytical methods and assumptions, confirmed that the results provided by PSEG were accurate.

Although the calculated peak temperature of 161 °F exceeds the UFSAR criterion of 149 °F with one heat exchanger cooling each pool, the staff identified conservative methods and values used in this calculation. The use of NRC Branch Technical Position ASB 9-2 for decay heat determination is conservative relative to other approved methods of determining decay heat. In addition, using that method, the staff calculated a background decay heat value of 5.0 MBTU/hour, which is significantly lower than the 6.8 MBTU/hour value used in the calculation. Lastly, the design heat exchanger performance is based on nominal service conditions, which is conservative for the purified water service conditions actually seen in the SFP cooling heat exchanger. In addition, other actual heat removal paths such as evaporative cooling, which provides significant heat removal at higher SFP temperatures, were conservatively neglected from the calculation.

To more accurately predict SFP temperature under various conditions, PSEG's contractor, HOLTEC International, developed and validated the Crosstie software program. The Crosstie software is used in PSEG's Integrated Decay Heat Management Program. In its letter dated October 2, 2002, PSEG committed to use the decay heat management program methodology prior to each refueling to:

(1) Calculate that the SFP temperature will not exceed 149 °F following full core offload, using one and only one heat exchanger for each SFP and to provide to the Operations staff the required component cooling water (CCW) temperature to achieve such results, and

(2) Calculate that the SFP temperature will not exceed 180 °F following full core offload with one heat exchanger available for both SFP's and to provide to the Operations staff the required CCW temperature to achieve such results.

Consistent with its integrated operating procedures for movement of spent fuel, PSEG also committed to validate the assumptions used in the Integrated Decay Heat Management Program calculations prior to initiating core offload activities. These validation activities include the following actions:

(1) Ensuring the availability of both SFP heat exchangers, each with an available spent fuel pit pump, to support spent fuel cooling for a full core offload, and

(2) Verifying that actual CCW supply temperatures are consistent with the decay heat management calculation input values.

The staff reviewed the critical software document and audited the decay heat management program evaluation of the most recent Salem refueling outage. The staff found that the software was calibrated against actual plant data, and that validation against SFP transients, including loss and restoration of cooling, produced acceptable agreement. The staff also found that, with representative input data, the predicted SFP temperature profile was in close agreement with that of the previous refueling outage. The staff also reviewed the partitioning of heat removal between the SFP cooling system and evaporative heat losses. The staff concluded that the evaporative heat losses were consistent with the SFP temperature and represented a small fraction of the total heat loss up to temperature of 180 °F. The staff performed independent calculations to verify that the rate of evaporation would represent a small fraction of the total ventilation system exhaust flow and, therefore, would be within the capacity of the ventilation system to remove. Accordingly, the staff concluded that the Crosstie software, as implemented in the decay heat management program, will provide an accurate representation of peak SFP temperature.

The Salem SFP's high temperature alarm setpoint is 125 °F. The alarm setpoint is also an entry condition for abnormal operating procedure S1(2).OP-AB.SF-0001(Q), "Loss of Spent Fuel Pool Cooling." Actions directed by this procedure include suspension of fuel movement into the SFP, periodic monitoring of SFP temperature, restoration or increase of SFP cooling, verification of SFP level, and operation of the fuel handling building ventilation system. If peak SFP temperature, as predicted by the decay heat management program, exceeds 125 °F for a refueling outage, then exceeding the alarm setpoint is an expected condition, and the alarm would not be indicative of an actual loss or degradation of SFP cooling. Because the normal operating configuration of the SFP cooling system is a single circulating water pump flowing through a single heat exchanger, loss of the operating pump or loss of heat removal through the heat exchanger would require operator action to restore forced SFP cooling. To ensure a high probability that operators are alerted to conditions that require operator action to restore forced cooling, PSEG committed to maintain SFP high temperature alarm capability to alert the operators in the event that SFP temperature exceeds the peak temperature predicted by the decay heat management program for each refueling outage. The availability of several SFP cooling pumps and the provision of an alarm to alert operators to a loss of forced cooling provides adequate assurance that SFP cooling will be maintained consistent with its importance to safety and that accident conditions involving a loss of SFP cooling will not result in a significant loss of SFP coolant inventory.

3.1.1 NRC Staff's Conclusion

Based on its review, the staff concludes that the proposed revisions to TS 3/4.9.3, in conjunction with the specified operational controls, ensure that the available decay heat removal capability will be maintained consistent with its importance to safety and that the SFP cooling system provides the capability to prevent a significant reduction in coolant inventory under accident conditions. Specifically, the decay heat removal capability is acceptable because: (1) the SFP cooling system will be capable of maintaining an appropriate pool temperature consistent with the current design basis during planned refueling evolutions; and, (2) with the failure of a single cooling train, the cooling system will maintain SFP temperature within analyzed limits for SFP structural integrity with the remaining cooling system in operation to cool both trains. Therefore, the proposed revisions to TS 3.9.3 are acceptable with respect to the resulting increase in decay heat. The proposed modifications to Surveillance Requirement 4.9.3 and the Bases for TS 3/4.9.3 are consistent with the revised LCO, and are, therefore, acceptable. The existing SFP cooling system and associated administrative controls ensure the potential increase in decay heat load, resulting from reducing the minimum decay time from 168 hours to 100 hours (TS 3/4.9.3) in a SFP with the maximum number of fuel assemblies provided in TS 5.6.3, can be removed with reliability consistent with the importance to safety of decay heat removal.

3.2 Fuel-Handling Accident Radiological Consequences

The staff reviewed the proposed changes submitted by PSEG in its June 28, 2002, application. The FHA is the limiting event with regard to the proposed TS change. Since only the FHA was revised to use the AST, the Salem implementation of the AST is considered a selective application applicable only to the FHA analyses. The following sections of this Safety Evaluation (SE) provide the results of the staff's review of the licensee's analyses. Table 1 to this SE tabulates the analysis inputs and assumptions found acceptable to the staff. Although the staff did confirmatory analyses, the staff's approval of this amendment is based on the information docketed by the licensee as well as on the staff's finding that the methods, inputs, and assumptions used in the licensee's analyses are acceptable.

3.2.1 FHA Evaluation

The licensee evaluated the consequences of an event in which a spent fuel assembly is dropped during refueling, damaging all of the fuel rods in the assembly. This accident is postulated to occur inside the containment or in the fuel-handling building (FHB). The licensee considered two potential release points for the containment release and three potential release points for the FHB. The licensee reported the limiting case for an FHA in either the containment or the FHB.

The inventory of fission products in the reactor core is a function of the reactor power, the duration of the at-power operation, and the time after shutdown prior to spent fuel movement. PSEG determined the core inventory assuming a power level of 3600 megawatts thermal (MWt) (greater than 102% of the rated thermal power), an extended period of operation sufficient for significant radionuclides to reach equilibrium, and a decay period of 96 hours following shutdown. To account for differences in power distribution across the core, a peaking factor of 1.7 is applied to the average inventory. The majority of the fission products produced during operation are contained within the fuel pellet; however, some migrate to void spaces, known as "gap," within the fuel rods. PSEG assumed that 8% of the I-131 inventory of the core was in the fuel rod gap, along with 10% of the Kr-85, 12% for alkali metals, and 5% of all other iodines and noble gases.

PSEG assumes that a single fuel assembly is dropped over the reactor vessel, or over the SFP. The Salem reactor cores contain 193 assemblies each. The radionuclides are assumed to be released from the damaged fuel rods, pass through the water in the reactor cavity or SFP, and enter the building atmosphere instantaneously. As the released gases rise through the overlaying water, halogens are scrubbed by the water column, resulting in an effective halogen decontamination factor of 200. No decontamination of noble gases or organic iodine forms was assumed. The fission products are released to the environment for a 2-hour period via the open containment equipment hatch (CEH), personnel air locks, and other penetrations. The CEH provides a direct release path to the environment. Releases via the other paths are collected and released via the plant vent (PV). Since PSEG assumes a 100% release over 2 hours and has taken no credit for engineered safeguards features for isolation or filtration of releases to the environment, the only parameter that differentiates the release points in the containment and FHB cases is the atmospheric dispersion. To simplify the calculational effort, PSEG selected the limiting χ/Q values for each case and performed the dose analysis once. The releases in both cases were treated as ground level releases for determining the exclusion area boundary (EAB), low population zone (LPZ), and control room doses. PSEG assumes a release rate based on the release of 99% of the radionuclides in the containment or FHB to the environment over a 2-hour period. PSEG determined this rate to be equivalent to a release flow rate of 99,800 cubic feet per minute (cfm) for the containment and 21,439 cfm for the FHB.

The licensee evaluated the dose to operators in the control room. For this assessment, PSEG assumed a normal outside air makeup flow rate of 1320 cfm. PSEG assumes that the control room ventilation system automatically realigns within 2 minutes into an emergency configuration. In this mode, the normal outside air makeup is halted and filtered emergency pressurization is started at a flow rate of 2200 cfm. The flow rate through control room recirculation filters is 5000 cfm.

PSEG assumed an unfiltered inleakage flow rate of 4000 cfm in lieu of a measured value. In response to staff questions, the licensee stated in its August 16, 2002, supplement that the 4000 cfm value was an arbitrary value that they believe reasonably bounds expected unfiltered inleakage. PSEG stated, for purposes of comparison, that a recent tracer gas test at the Hope Creek units measured an unfiltered inleakage of 200 cfm, with a makeup flow rate limit of 1000 cfm. Although the staff would not accept this statement in justifying an unfiltered inleakage of 200 cfm for Salem, it is indicative of the conservatism of the 4000 cfm assumed in the Salem analyses. The staff believes that, with the assumption of the increased inleakage flow rate, there is adequate assurance that the radiation doses to the control room personnel will not impede response actions necessary to protect the public. The staff has issued, for public comment, four draft regulatory guides (DGs) on control room habitability issues and a generic communication that will request licensees to provide information related to control room habitability issues. The staff's acceptance of the PSEG's assumed unfiltered inleakage does not exempt the licensee from future regulatory actions that may become applicable due to the generic initiative.

Details on the assumptions found acceptable to the staff are presented in Table 1. The staff performed a confirmatory analysis. The doses estimated by the licensee for the postulated FHAs were found to be acceptable.

3.2.2 Atmospheric Relative Concentration Estimates

PSEG calculated new χ/Q estimates for the CEH at both units for both control room intakes and calculated new values for the Unit No. 1 plant vent and the Unit No. 1 FHB roll up door for both control room intakes. Values were not determined for the Unit No. 2 plant vent and the Unit No. 2 FHB roll up door as these would be bounded by the values for Unit No. 1. PSEG used the guidance of DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," in developing these values. PSEG did not credit the ability to preferentially select a control room intake even though that capability is available.

All of the values were determined as ground level releases using meteorological data collected for the years 1988 through 1994, using the NRC-sponsored ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). The meteorological data were obtained from the site meteorological tower that services both Salem and Hope Creek. PSEG stated that the meteorological measurements program meets the guidance in RG 1.23, “Onsite Meteorological Programs,” and includes quality assurance provisions consistent with Appendix B to 10 CFR Part 50. The staff performed a series of statistical checks on the meteorological data to evaluate its suitability. This evaluation compared stability class, wind speed, and wind direction distributions observed in each year between 1988 through 1994 with the other years and with corresponding distributions provided in the Salem UFSAR. The staff compared the ARCON96 code inputs used by PSEG to the site release point and intake configuration and found the inputs to be acceptable. Since the licensee used an NRC-sponsored computer code and provided its code printouts for review, the staff only ran one confirmatory case. The staff finds the χ/Q values listed in Table 1 to be acceptable.

3.2.3 NRC Staff’s Conclusion

The NRC staff has reviewed the AST implementation proposed by PSEG for Salem. The staff also reviewed the proposed changes to TS 3/4.9.3. In performing its review, the staff relied upon information placed on the docket by licensee, staff experience in doing similar reviews and, where deemed necessary, on staff confirmatory calculations.

This licensing action is considered a selective implementation of the AST. While the licensee adopted all characteristics of the AST, PSEG’s assessment was limited to the consequences of an FHA. With the approval of this amendment, the AST, the total effective dose equivalent (TEDE) criteria, and the analysis methods, assumptions and inputs become the licensing basis for the assessment of radiological consequences of FHA DBAs. All future FHA radiological analyses done to show compliance with DBA dose acceptance criteria will use this approved licensing basis. This approval is limited to this specific application. The AST and TEDE criteria may not be extended to other aspects of plant design or operation without prior NRC review pursuant to 10 CFR 50.67.

The staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183. The staff compared the radiation doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff finds, with reasonable assurance, that the licensee’s estimates of the TEDE due to FHA accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183.

The staff finds reasonable assurance that Salem 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 in analysis assumptions and parameters. The staff concludes that the proposed AST implementation and the proposed TS 3/4.9.3 change are acceptable from the standpoint of radiological consequences.

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 had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the 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 (67 FR 55887). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Attachment: Table 1 - FHA Accident Analysis Assumptions

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Date: October 10, 2002

TABLE 1

ANALYSIS ASSUMPTIONS

Core power (includes 2% uncertainty penalty), MWt	3,600
Radial peaking factor	1.7
Number of damaged fuel assemblies	1
Number of fuel assemblies in core	193
Decay time, hours	96
Fuel rod gap fractions	
I-131	0.08
Kr-85	0.10
All other noble gases, iodines	0.05
Alkali metals	0.12
Iodine species fractions	
Elemental	0.9985
Organic	0.0015
Particulates	none
Water depth, ft	23
Pool scrubbing factor, effective	200
Release modeling	
Immediate release from fuel through pool to building	
100% release from buildings in 2 hours	
No credit for building holdup or filtration prior to release; no containment closure	
Control room volume, ft^3	81,400
Control room emergency ventilation start, minutes	2
Control room ventilation flow rates, cfm	
Unfiltered outside air makeup prior to isolation	1,320
Filtered outside air makeup after isolation	2,200
Recirculation flow rate	5,000
Unfiltered inleakage	4,000
Control room filter efficiency, %	95
Control room occupancy factors	
0-24 hr	1.0
24-96 hr	0.6
96-720 hr	0.4
Control room breathing rate, m^3/s	3.5E-4
Offsite breathing rate, m^3/s	
0-8 hrs	3.5E-4

Atmospheric dispersion factors, s/m^3

Time Interval Hr	Control Room χ/Q, sec/m3			
	U1 Equipment Hatch to U1 CR Intake*	U1 Equipment Hatch to U2 CR Intake	U2 Equipment Hatch to U2 CR Intake	U2 Equipment Hatch to U1 CR Intake
0-2	2.86E-3	9.47E-4	2.85E-3	9.38E-4
2-8	2.22E-3	7.55E-4	2.33E-3	7.72E-4
8-24	9.15E-4	3.26E-4	9.68E-4	3.28E-4
24-96	6.60E-4	2.33E-4	6.51E-4	2.20E-4
96-720	5.62E-4	1.96E-4	5.07E-4	1.70E-4

Time Interval Hr	Control Room χ/Q, sec/m3			
	U1 Plant Vent to U1 CR Intake*	U1 Plant Vent to U2 CR Intake	U1 FHB Door to U2 CR Intake*	U1 FHB Door to U1 CR Intake
0-2	1.78E-3	8.84E-4	1.50E-3	8.59E-4
2-8	1.31E-3	6.60E-4	1.20E-3	7.10E-4
8-24	5.22E-4	2.64E-4	4.48E-4	2.70E-4
24-96	3.77E-4	1.93E-4	3.22E-4	1.88E-4
96-720	3.17E-4	1.62E-4	2.50E-4	1.55E-4

* Used in dose analysis.

Time Interval Hr	χ/Q, sec/m3	
	Exclusion Area	Low Population Zone
0-2	1.30E-3	1.86E-5
2-8		7.76E-6
8-24		5.01E-6
24-96		1.94E-6
96-720		4.96E-7

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