May 4, 1994

Docket Nos. 50-272/311

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: SPENT FUEL POOL RERACKING, SALEM NUCLEAR GENERATING STATION, UNITS 1 AND 2 (TAC NOS. M85797 AND M85798)

The Commission has issued the enclosed Amendment Nos. 151 and 131 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 28, 1993, as supplemented by letters dated August 12, 1993, November 17, 1993, February 2, 1994, and April 7, 1994.

These amendments increase the spent fuel pool capacities for Salem 1 and 2 from the current 1170 fuel assemblies to 1632 fuel assemblies. Also, the decay time for refueling operations is extended from 100 hours to 168 hours.

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

You are requested to notify the NRC, in writing, when these amendments have been implemented at Salem 1 and 2.

Sincerely,
/S/
James C. Stone, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000272
P PDR

Enclosures:
1. Amendment No. 151 to
License No. DPR-70
2. Amendment No. 131 to
License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

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

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James C. Stone

James C. Stone, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 151 to
   License No. DPR-70
2. Amendment No. 131 to
   License No. DPR-75
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Steven E. Miltenberger
Public Service Electric & Gas
Company

Salem Nuclear Generating Station,
Units 1 and 2

cc:

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
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 Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated April 28, 1993, as supplemented by letters dated August 12, 1993, November 17, 1993, February 2, 1994, and April 7, 1994, 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. 151 , 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 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 4, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 151

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 9-3	3/4 9-3
5-5	5-5
5-6	5-6
-	5-6a

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 168 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 168 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 168 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

a. In accordance with the code requirements specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,811 ± 100 cubic feet at a nominal Tavg of 576.7°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The new fuel storage racks are designed and shall be maintained with:

a. A maximum Keff equivalent of 0.95 with the storage racks flooded with unborated water.

b. A nominal 21.0 inch center-to-center distance between fuel assemblies.

c. A maximum unirradiated fuel assembly enrichment of 4.5 w/o U-235.

5.6.1.2 The spent fuel storage racks are designed and shall be maintained with:

a. A maximum Keff equivalent of 0.95 with the storage racks filled with unborated water.

b. A nominal 10.5 inch center-to-center distance between fuel assemblies stored in Region 1 (flux trap type) racks.

c. A nominal 9.05 inch center-to-center distance between fuel assemblies stored in Region 2 (non-flux trap) racks.

d. Fuel assemblies stored in Region 1 racks shall meet one of the following storage constraints.

1. Unirradiated fuel assemblies with a maximum enrichment of 4.25 w/o U-235 have unrestricted storage.

2. Unirradiated fuel assemblies with enrichments greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235, that do not contain Integral Fuel Burnable Absorber (IFBA) pins, may only be stored in the peripheral cells facing the concrete wall.

3. Unirradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235, that contain IFBA rods with a nominal 2.35 mg B-10/linear inch loading, and a number of IFBA rods equal to or greater than the number determined by the equation below, have unrestricted storage.

   $N = 42.67\ (E-4.25)$

4. Irradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o, that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

   $BU\ (MWD/kg\ U) = -26.212 + 6.1677E$

e. Fuel assemblies stored in Region 2 racks shall meet one of the following storage constraints.

1. Unirradiated fuel assemblies with a maximum enrichment of 5.0 w/o U-235 may be stored in a checkerboard pattern with intermediate cells containing only water or non-fissile bearing material.

2. Unirradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 may be stored in the central cell of any 3x3 array of cells provided the surrounding eight cells are empty or contain fuel assemblies that have attained the minimum burnup (BU) as determined by the equation below.

   $BU\ (MWD/kg\ U) = -15.48 + 17.80E - 0.7038E^2$

   In this configuration, none of the nine cells in any 3x3 array shall be common to cells in any other similar 3x3 array. Along the rack periphery, the concrete wall is equivalent to 3 outer cells in a 3x3 array.

3. Irradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

   $BU\ (MWD/kg\ U) = -32.06 + 25.21E - 3.723E^2 + 0.3535E^3$

4. Irradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 that have attained the minimum burnup (BU) as determined by the equation below, may be stored in a peripheral cell facing the concrete wall.

   $BU\ (MWD/kg\ U) = -25.56 + 15.14E - 0.602E^2$

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 124'8".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1632 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

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

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
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 Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated April 28, 1993, as supplemented by letters dated August 12, 1993, November 17, 1993, February 2, 1994, and April 7, 1994 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) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 131, 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 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 4, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 9-3	3/4 9-3
5-5	5-5
-	5-5a
-	5-5b

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 168 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 168 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 168 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The new fuel storage racks are designed and shall be maintained with:

a. A maximum Keff equivalent of 0.95 with the storage racks flooded with unborated water.

b. A nominal 21.0 inch center-to-center distance between fuel assemblies.

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3. Unirradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o U-235, that contain IFBA rods with a nominal 2.35 mg B-10/linear inch loading, and a number of IFBA rods equal to or greater than the number determined by the equation below, have unrestricted storage.

$$N = 42.67 (E-4.25)$$

4. Irradiated fuel assemblies with enrichments (E) greater than 4.25 w/o U-235 and less than or equal to 5.0 w/o, that have attained the minimum burnup (BU) as determined by the equation below, have unrestricted storage.

   BU (MWD/kg U) = $-26.212 + 6.1677E$

e. Fuel assemblies stored in Region 2 racks shall meet one of the following storage constraints.

   1. Unirradiated fuel assemblies with a maximum enrichment of 5.0 w/o U-235 may be stored in a checkerboard pattern with intermediate cells containing only water or non-fissile bearing material.

   2. Unirradiated fuel assemblies with a maximum enrichment (E) of 5.0 w/o U-235 may be stored in the central cell of any 3x3 array of cells provided the surrounding eight cells are empty or contain fuel assemblies that have attained the minimum burnup (BU) as determined by the equation below.

      BU (MWD/kg U) = $-15.48 + 17.80E - 0.7038E^2$

      In this configuration, none of the nine cells in any 3x3 array shall be common to cells in any other similar 3x3 array. Along the rack periphery, the concrete wall is equivalent to 3 outer cells in a 3x3 array.

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      BU (MWD/kg U) = $-25.56 + 15.14E - 0.602E^2$

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 151 AND 131 TO FACILITY OPERATING

LICENSE NOS. DPR-70 AND DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated April 28, 1993, as supplemented by letters dated August 12, 1993, November 17, 1993, February 2, 1994, and April 7, 1994, the Public Service Electric & Gas Company (the licensee, PSE&G) submitted a request for changes to the Salem Nuclear Generating Station (SGS), Unit Nos. 1 and 2, Technical Specifications (TS). The requested changes would increase the spent fuel pool capacities for Salem 1 and 2 from the current 1170 fuel assemblies to 1632 fuel assemblies. Also, the decay time for refueling operations is being extended from 100 hours to 168 hours. The April 7, 1994, letter provided clarifying information and did not change the initial proposed no significant hazards consideration determination.

Each SGS Unit currently has a total storage capacity of 1170 cells in their spent fuel pool storage racks. These racks provide adequate capacity for storage of spent fuel while maintaining an operational full core reserve discharge capacity of 300 storage locations. Operational full core reserve includes both a full core fuel assembly reserve (193 storage locations) plus additional locations typically required for storage of non-fuel bearing components and maneuverability during refueling. Unit 1 will lose its operational full core reserve by March 1998, and Unit 2 by March 2002. Therefore, to preclude this situation and to ensure that sufficient spent fuel storage capacity continues to exist at SGS, PSE&G plans to install poisoned maximum density spent fuel storage racks whose design incorporates Boral as a neutron absorber in the cell walls thereby allowing for more dense storage of spent fuel. The reracking would provide an ultimate storage capacity of 1632 cells and extend the date of loss of operational full core reserve to September of 2008 and September of 2012 for Unit 1 and Unit 2, respectively.

2.0 EVALUATION

2.1 Materials

2.1.1 Structural Materials

The licensee has selected the following structural materials for use in the proposed storage rack modification:

- American Society of Mechanical Engineers (ASME) Section II SA240-304 stainless steel for fabrication of the racks,
- ASME SA240-304 for the internally threaded support legs,
- ASME SA564-630 for the externally threaded support spindle - this is a precipitation hardened stainless steel, heat treated to 1100°F, and
- Weld material - type R308L stainless steel conforming to ASME specification SFA 5.9.

ASME Section II, SA240, Type 304 stainless steel is a common austenitic alloy frequently used in nuclear applications. The choice of type 304 stainless steel for fabrication of the rack assembly legs is reasonable. The high chromium content imparts reasonable corrosion resistance to oxidizing effects of most electrolytes when at low concentration levels. The steel is, however, susceptible to corrosion in acidic solutions ($pH < 7.0$) containing chloride or fluoride anions. These anions can lead to pitting of the material. The corrosion effects by chloride or fluoride anions is not as pronounced in basic media ($pH > 7.0$).

The licensee has opted to use a Type 630 martensitic, precipitation hardened, stainless steel for the externally threaded support spindle. Type 630 stainless steels have increased strength, without suffering considerable loss of ductility. The corrosion resistance, however, is not quite as good as that of austenitic stainless steels. The Type 630 stainless steel has been heat treated at 1100°F to increase its resistance to stress corrosion cracking.

It should be noted that control of water impurities in nuclear plant spent fuel pool water is typically provided by the spent fuel pool demineralizers in the spent fuel cooling system. The demineralizers function to keep the chemistry of the spent fuel pool water approximately the same as that of the reactor coolant system, in order to minimize the probability of abnormal chemistry incursions during refueling operations when the two systems link together. Control of spent fuel pool chemistry, however, also serves to reduce corrosion effects by keeping the concentrations of water impurities at low levels. Therefore, stress corrosion cracking or pitting, induced by residual chloride or fluoride ions in the fuel pool, should not be a problem with the SA240-304 stainless steel.

2.1.2 Poison Material

- Boral - patented material produced by AAR Brooks and Perkins

The Boral panels used in the proposed rack modifications are manufactured in accordance with AAR Brooks and Perkins certified procedures. Production of Boral falls within the scope of the manufacturer's quality assurance program (10 CFR 50 Appendix B) for nuclear grade materials. The licensee intends to install the Boral sheets by freely inserting them between the 304 stainless steel walls of the rack assemblies and the 304 stainless steel sheaths which are to be welded to the wall.

It is evident that the insertion of the Boral panels into the sheathed areas will create a tight fit. Independent studies by industry organizations and by NRC contractors have shown that Boral may react with water or moisture to generate hydrogen gas. Production of hydrogen may result in deformation of the rack cells by imparting additional stresses on the walls. Information Notice 83-29, "Fuel Binding Caused by Fuel Bundle Deformation," was issued to alert the industry to this concern. The licensee's submittal indicates that holes at the corners of the sheath areas will create a sufficient vent path for any potential hydrogen which may be produced by a water-aluminum reaction.

The licensee has also created an accelerated Boral surveillance program to characterize the performance of the Boral panels during the remaining lifetime of the plant. This program is in accordance with the NRC Letter of April 14, 1978 to all nuclear power licensees, which stated that "Methods for verification of long-term material stability and mechanical integrity of special poison materials utilized for neutron absorption should include actual tests."

The licensee's accelerated Boral Surveillance Program calls for placing ten Boral test coupons (mounted on a "tree") in each of the spent fuel pool rack areas of SGS Units 1 & 2. At the end of the first five operating cycles following the modification, the coupon tree will be surrounded with eight freshly discharged fuel assemblies. This is done to assure that the coupons experience a higher radiation dose than the Boral panels in the storage racks. Beginning with the fifth spent fuel load, the fuel assemblies surrounding the test coupon tree may remain in place for the remaining life of the racks.

The accelerated Boral Surveillance Program calls for removing and testing one Boral test coupon at the following refueling outages for each unit after the rack modifications are complete: 2nd, 3rd, 5th, 10th, 15th, 20th, 25th, 30th, 35th, and 40th. Each test panel, upon its removal, will be analyzed according to the following tests:

- Visual Observation and Photography
- Neutron Attenuation
- Dimensional Measurements (length, width, and thickness)
- Weight and Specific Gravity Analyses
- Wet Chemical Analysis (Optional)

The neutron attenuation and the dimensional measurements are the more important tests of the group since they are used to determine whether or not the coupons are exhibiting any signs of boron loss or structural deformation, respectively.

The licensee's contractor has established an acceptable set of screening criteria for evaluating the Boral test coupons. The results of testing on the Boral test coupons will be compared to identical tests run on the Boral control coupons.

2.1.3 Conclusion

The PSE&G license amendment request submittal indicates that material selection for the SGS Units 1 & 2 spent fuel rack modifications have been satisfactorily thought out. The racks are to be constructed from a Type 304 stainless steel fabricated according to an approved ASME Section II specification. Boral is an acceptable poison material; however, since the Boral may generate hydrogen when in contact with water or moisture, care must be taken to provide a sufficient path to allow potential hydrogen generation to vent from the sheath area. The Boral surveillance program will provide a reliable method of assessing the potential deformation or degradation of Boral panels which are exposed to radiation in the spent fuel area over time. Following the review of the licensee's submittal, the staff concludes that the licensee's selection of structural, welding and poison materials meets current industry and regulatory standards and that these materials are acceptable for construction of the new rack modules.

2.2 Criticality

2.2.1 Criticality Evaluation

After reracking, two separate storage regions will be provided in the spent fuel pool with independent criteria defining the highest potential reactivity in each of the two regions. Region 1 will utilize the three existing Exxon Nuclear Corporation (now Siemens Nuclear Corporation) flux trap type, high density racks and Region 2 will contain the nine new Holtec International non-flux trap type, maximum density racks. Region 1 is designed to accommodate fresh fuel with a maximum enrichment of 4.25 weight percent (w/o) U-235. Unirradiated and irradiated fuel with initial enrichments up to 5.0 w/o U-235 can also be stored in Region 1 with some restrictions. These restrictions are stated in proposed TS 5.6.1.2d. Region 2 is designed to accommodate unirradiated and irradiated fuel with stricter controls as compared to Region 1. These controls are stated in proposed TS 5.6.1.2e.

The analysis of the reactivity effects of fuel storage in Regions 1 and 2 was performed with the two-dimensional transport theory code, CASMO-3. Independent verification calculations were made with the KENO-5a Monte Carlo computer code using the 27-group SCALE cross-section library. Since the KENO-5a code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the SGS spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber thickness. These two independent methods of analysis (KENO-5a and CASMO-3) showed good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a calculations, a minimum of 500,000 neutron histories in 1,000 generations of 500 neutrons each were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-5a reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the SGS storage racks with a high degree of confidence.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Unborated pool water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures.

(2) Assumption of infinite array of storage cells in all directions (except for the assessment of peripheral effects and certain abnormal conditions where neutron leakage is inherent).

(3) Neutron absorption effect of structural material is neglected.

The design basis fuel assembly was a standard Westinghouse 17x17 array of fuel rods containing UO_2 at a maximum initial enrichment of 4.5 w/o U-235. In addition, a Westinghouse Vantage-5H fuel assembly, identical in dimensions to the standard assembly but with a burnable poison (IFBA) coating on some fuel rods, was considered with enrichments up to 5.0 w/o U-235. Since the burnable poison may burn up more rapidly than the fuel, the reactivity effect of these assemblies was evaluated at the point of highest reactivity over burnup.

The staff concludes that appropriately conservative assumptions were made.

For the nominal storage cell design, uncertainties due to boron loading tolerances, boral width tolerances, tolerances in cell lattice spacing, stainless steel thickness tolerances, and fuel enrichment and density tolerances were accounted for. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations and the effect of the axial distribution in burnup in those cases where burnup credit is used.

For the existing storage racks in Region 1, analysis has shown that fresh fuel of 4.25 w/o U-235 enrichment results in a maximum k_{eff} of 0.949, including calculational and manufacturing uncertainties (95%/95%). This meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable. Because of neutron leakage from the rack, fresh fuel of 5.0 w/o U-235 enrichment stored in the peripheral cells of Region 1 resulted in a maximum k_{eff} of 0.946, including uncertainties. In addition, Westinghouse Vantage-5H fuel assemblies of 5.0 w/o enrichment containing a minimum of 32 IFBA rods resulted in a maximum k_{eff} of 0.935. Vantage-5H fuel with a larger number of IFBA rods would provide a greater margin below the 0.95 k_{eff} limit. For enrichments between 4.25 w/o and 5.0 w/o U-235, the minimum required number of IFBA rods may be determined by interpolation, as given in proposed TS 5.6.1.2d.3. Spent fuel assemblies with initial enrichments up to 5.0 w/o U-235 were found to result in a maximum k_{eff} no greater than 0.949 provided the minimum burnup as a function of initial enrichment falls within the acceptable domain of Figure 4.2.1 of Reference 1. For convenience, the limiting burnup data in Figure 4.2.1 of Reference 1 is incorporated in proposed TS 5.6.1.2d.4 as a fitted polynomial expression that conservatively bounds the curve.

For the Region 2 racks, fresh fuel with a maximum U-235 enrichment of 5.0 w/o stored in a checkerboard pattern with intermediate cells containing only water or non-fissile bearing material resulted in a maximum k_{eff} of 0.846, including uncertainties. Fresh fuel assemblies with a maximum enrichment of 5.0 w/o U-235, stored in the central cell of any 3x3 array of cells in which the surrounding eight cells are empty or contain fuel that have attained the minimum burnup shown in the upper curve of Figure 4.2.2 of Reference 1, resulted in a maximum k_{eff} of 0.942, including uncertainties. In addition, 5.0 w/o fuel that has attained the minimum burnup shown in the middle curve of Figure 4.2.2 of Reference 1 resulted in a maximum k_{eff} of 0.933, including uncertainties. Irradiated fuel with a maximum initial enrichment of 5.0 w/o U-235 that has attained the minimum burnup shown in the bottom curve of Figure 4.2.2 of Reference 1 resulted in a maximum k_{eff} of 0.929, including uncertainties, when stored in a peripheral cell in Region 2. These acceptable burnup domains are presented in TS 5.6.1.2e by fitted polynomial expressions that conservatively bound the curves shown in Figure 4.2.2 of Reference 1.

This reactivity equivalencing method is the standard one used for storage rack reactivity evaluations and is acceptable. Although not included in the burnup dependent criticality analyses, subsequent decay of Pu-241 with long-term storage results in a significant decrease in reactivity. This will provide an increasing subcriticality margin and further compensate for any uncertainty in the depletion calculations.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the inadvertent misloading of an assembly with a burnup and enrichment combination outside of the acceptable areas in Figures 4.2.1 or 4.2.2 of Reference 1, which could lead to an increase in reactivity. However, for such events credit may be taken for the presence of soluble boron in the pool water which is assured by administrative procedures during fuel handling operations since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The plant procedures require that the boron concentration in the pool be maintained between 2300 and 2500 ppm during operating modes, which is confirmed by weekly surveillance measurements. The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. In fact, the licensee has confirmed that a minimum boron concentration of only 600 ppm boron would be adequate to assure that the limiting k_{eff} of 0.95 is not exceeded.

The following Technical Specification changes have been proposed as a result of the requested spent fuel pool reracking. Based on the above evaluation,the staff finds these changes acceptable as well as the associated Bases changes.

(1) TS 5.6.1 has been separated into two specifications. New TS 5.6.1.1 reflects the storage requirements for fresh fuel storage in the new (fresh) fuel storage racks. New TS 5.6.1.2 reflects the new requirements for fuel storage in Region 1 and Region 2.

(2) TS 5.6.3 has been modified to reflect the increased fuel pool storage capacity to 1632 fuel assemblies.

2.2.2 Conclusion

Based on the review described above, the staff finds the criticality aspects of the proposed modifications to the SGS spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

2.3 Spent Fuel Pool Cooling

The Salem spent fuel pool (SFP) cooling system for each unit consists of two 100% capacity pumps and one heat exchanger for normal decay heat removal. The SFP cooling system has its maximum duty during the refueling operation when the decay heat from the spent fuel is the highest. The system is normally placed in operation prior to the transfer of any fuel and continues operating as long as required to maintain temperature at the required level. The licensee plans to continue the Salem operating practice of maintaining the maximum spent fuel pool temperature (currently 120°F) at values that maintain the physical integrity of the spent fuel pool demineralizer resin.

Installed piping and valves allow the Units 1 and 2 heat exchangers to be cross connected. During normal plant operation, the heat exchangers operate independently to meet the cooling requirements of the individual units. The cross connect also allows one heat exchanger to be used to alternately cool the spent fuel pools in both units during periods when one heat exchanger is out of service for maintenance.

No modifications to the SFP cooling system are planned with the proposed expansion. Therefore, the spent fuel pool cooling system was only reviewed for decay heat removal capability and for makeup capability during loss of all cooling.

The licensee calculated decay heat loads of 25×10^6 BTU/hr after a normal discharge of spent fuel during refueling (fuel transfer takes place 168 hours after reactor shutdown). This is based on a proposed storage capacity of 1632 spent fuel assemblies. The heat load value was compared to Branch Technical Position 9-2 and found to be conservative. The calculated pool temperature rises to a maximum of less than 150°F at 195 hours after shutdown for a normal discharge of 88 fuel assemblies. The pool temperature is calculated to remain above 140°F for over 200 hours. If cooling fails, the time-to-boil for a normal off-load and 150°F pool is 4.6 hours with no makeup water. Although this maximum temperature is above the guideline of 140°F for a normal discharge, it is acceptable because it is well below the boiling temperature and the maximum temperature will not damage the spent fuel cooling system.

When a full core is off-loaded into the spent fuel pool, the maximum decay heat load is calculated to be 40×10^6 BTU/hr. This heat load results in a calculated maximum temperature of 180°F at 205 hours after shutdown. The maximum temperature of the pool for the abnormal condition of full-core off-load is acceptable because it is below boiling.

The SFP cooling system is not seismic category I. There are four redundant makeup water sources available to the SFP. The normal source of makeup water to the spent fuel pool is the Demineralized Water System which distributes water from two 500,000 gallon Demineralized Water Storage Tanks. The tanks and the distribution system do not have seismic classification. Makeup is also available from the Primary Water Storage Tank via the makeup pumps (seismic class II) and from the Chemical and Volume Control System holdup tanks via the holdup tank recirculation pump (seismic class II) rated at 500 gpm. For the fourth source of makeup water, valves have been installed on the existing 6-inch spare nozzles on both Refueling Water Storage Tanks (RWSTs). These tanks are seismic class I. A portable pump, with appropriate suction and discharge connections and hose are provided with the capability to deliver approximately 100 gpm makeup water flow from one of the RWSTs directly to the spent fuel pool. The valves installed on the RWSTs are locked, closed and capped, and under administrative control. The portable pump and hose are also under administrative control to ensure constant and timely availability.

While Salem Units 1 and 2 SFP cooling systems were not designed to comply with the NRC Standard Review Plan, the design was reviewed and approved by the NRC in Salem Safety Evaluation Report Section 9.5 dated October 11, 1974. The

SFP cooling system was found to be acceptable because there are four redundant pool makeup water sources to ensure a reliable supply of makeup water. The proposed rerack does not alter the SFP cooling system design or the associated makeup water sources.

The rerack does not change the design of the SFP cooling system, which was previously accepted by the NRC. The maximum calculated temperature for the pool is below boiling. There are diverse sources of makeup water available. Therefore, the decay heat removal for normal and abnormal conditions and the provisions for makeup capability are acceptable.

Technical Specification 3.9.3 currently requires the reactor to be subcritical for a minimum of 100 hours before beginning fuel movement. This is being changed to require the reactor to be subcritical a minimum of 168 hours before beginning fuel movement. This provides additional conservatism in the decay heat removal requirements for the spent fuel pool cooling system. The staff finds this acceptable.

2.4 Heavy Load Handling

A spent fuel storage rack is considered to be a heavy load because it weighs more than a spent fuel assembly and its handling tool. NUREG-0612 outlines the guidelines for heavy load handling. The general guidelines are as follows: (1) safe load paths should be defined, (2) load handling procedures should be developed, (3) crane operators should be trained, (4) special lifting devices should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials," (5) other lifting devices should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings," (6) the crane should be inspected, tested, and maintained in accordance with ANSI B30.2-1976, "Overhead and Gantry Cranes."

Lifting and installation of the spent fuel racks will be performed in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The licensee has met both the general guidelines for heavy load handling and the specific guidelines for the spent fuel pool area. All reracking activities will be carried out along defined safe load handling paths. The crane operator will be required to follow specific load handling procedures. All crew members involved in the use of the lifting and upending equipment will be given training. The maximum weight of any storage rack and its associated handling tool is 17 tons. A new hoist with a rated load of 41.3 tons will be installed for the rerack. The rig complies with all provisions of ANSI 14.6-1978, including compliance with the primary stress criteria, load testing at 150% of maximum lift load, and dye examination of critical welds. The lifting crane and rig will meet the NUREG-0612 stress and inspection criteria.

For load handling operations in the spent fuel pool area, the licensee can either have a single-failure-proof handling system or the effects of heavy load drops should be analyzed and shown to satisfy the following criteria: (1) a heavy load drop will not produce releases of radioactive material that approach 10 CFR Part 100 limits, (2) damage to fuel and fuel storage racks does not result in a configuration of the fuel such that k_{eff} is larger than 0.95, (3) damage to the spent fuel pool will not result in water leakage that could uncover the fuel, and (4) damage to equipment will be limited so as not to result in loss of required safe shutdown functions.

Since the licensee will not be using a single-failure proof crane for the reracking, the licensee has evaluated the consequences of a heavy load drop and has established a program to minimize the potential for such a drop. The potential for a heavy load drop is minimized through the licensee's compliance with the guidelines of NUREG-0612, as described above. Heavy loads will not be carried over racks with stored spent fuel. Thus, a heavy load drop will not produce releases of radioactive material that approach 10 CFR Part 100 limits, and fuel and fuel storage racks will not be damaged in a heavy load drop such that k_{eff} exceeds 0.95. A heavy load drop in the SFP may damage the concrete structure, but the potential water loss resulting from this damage will not uncover stored spent fuel. Safe-shutdown equipment is not located in the vicinity of the defined safe load path, so a heavy load drop will not result in damage to this equipment. Therefore, the licensee meets the criteria for heavy load handling in the spent fuel pool area.

Based on the above, the staff finds that the heavy load handling will be performed in accordance with the guidelines of NUREG-0612 to ensure that an unacceptable release of radioactivity or criticality accident will not result from a heavy load drop, and is therefore acceptable.

2.5 Structural Design Considerations

The licensee has chosen to utilize the rack design of Holtec International, which has been implemented in several other nuclear power facilities regulated by the NRC. Holtec has performed both a whole-pool multi-rack (WPMR) and single rack analyses using the DYNARACK structural code.

The licensee submitted an update to the original submittal dated August 12, 1993 (Reference 2), participated in a teleconference (Reference 3), and responded to a staff request for additional information (Reference 4). The primary areas of staff review associated with the structural aspects of the proposed application are the seismic design considerations including spent fuel assembly and rack structural integrity, fuel handling accident analysis, and spent fuel pool integrity.

2.5.1 Seismic Design Considerations

2.5.1.1 Seismic Input Motion

Both the existing spent fuel racks and the new Holtec racks are designed to be seismic Category I equipment. Multiple time histories for both horizontal and vertical seismic input motions were utilized as outlined in Standard Review Plan Section (SRP) 3.7.1.

To construct an adequate input motion for the seismic analysis, the licensee incorporated four sets of statistically independent time histories. The resulting separate response spectra were examined to insure that the average of the four sets envelope the original floor design response spectra. Lastly, the time history set which generated the largest stresses and maximum displacements in a typical rack were used as the controlling set, increased by 10% for conservatism, and used to develop the stresses and displacements in the seismic analysis of all structures and components in the model. The highest correlation coefficient observed between the time histories in the analysis is 0.1, below the 0.15 criteria for judging statistical independency among components of input motions established by the staff. The staff accepts the licensee's input motion as a conservative set of synthetic time histories because (1) the procedure used to develop the controlling input time histories is in accordance with SRP Section 3.7.1, (2) the average spectra generated by the time histories envelope the target design floor response spectra, (3) the controlling time history set has been increased 10% at all frequencies for conservatism, and (4) the input time histories are statistically independent.

Thus, the staff concludes that the licensee has provided acceptable seismic input by generating input motion that is in compliance with SRP Section 3.7.1 and conservative in the manner in which it was developed.

2.5.1.2 Three-Dimensional Single Rack Analysis

The three-dimensional single rack analysis developed by the Holtec Corporation for this re-rack models the free-standing structures with a 22 degree-of-freedom system. The rack response to seismic input motions is very complex, since the free-standing racks can undergo displacement through over-turning, sliding, rocking, and twisting concurrently. Assumptions made in the submitted calculations include the modeling of the rattling of the assemblies by five lumped masses distributed along the height of the rack, movement of all assemblies in-phase, consideration of fluid coupling for rack-to-rack interaction as an anti-symmetric motion and simulation of the rack-to-assembly and rack-to-wall coupling by inertial coupling. Other conservative assumptions given credit by the staff include the omission of pool damping and fluid drag effects.

The single rack analysis results after 28 iterations are shown in Table 6.8.19 of Reference 1 and are in compliance with ASME Code, Section III, Subsection NF criteria. The calculated impact loads and the corresponding stresses are much less than the allowable values. The analysis shows there are no rack-to-wall impacts, and the limiting result of the controlling single rack analysis indicates a maximum in-phase rack displacement of 1.2 inches, which corresponds to a safety margin of 1.25, or simply that the existing gap between the racks is 1.25 times the expected maximum displacement. The weld loading analysis is provided in Table 6.7.32, and margins of safety for cell/cell welds, rack/baseplate welds, and pedestal/baseplate welds to be 7.9, 2.0, and 1.2 respectively. These margins of safety are acceptable.

The calculated impact loading for the fuel assemblies and the rack walls are also displayed in Table 6.7.32 of Reference 1. The impact safety margin for this impact in the single rack analysis is 3.4. This safety margin associated with the impact loading of the fuel assemblies assures the staff that the assembly has the ability to absorb the collision energy, and that the fuel cladding will remain intact during a seismic event. With respect to the safety margins associated with the calculated "x" and "y"-axis shear and bending stresses, the controlling margins of safety, as reported by the licensee (see Tables 6.7.4 through 6.7.31 of Reference 1), are 6.7 and 2.8 respectively.

The small displacements and significant safety margins indicate low levels of component response to the seismic input, and these results form the basis of the staff acceptance of the single rack analysis.

2.5.1.3 <u>Whole Pool Multi-Rack Analysis (WPMR)</u>

The WPMR analysis models each rack and the fuel contained in the rack cells, and allows for a much more accurate depiction of the fluid flow and fluid coupling effects during a seismic event. The licensee stated that scaled laboratory experiments were conducted by Holtec International to validate some aspects of the multi-rack fluid coupling theory used in the computer code DYNARACK. These experiments do offer a somewhat limited indication that the whole-pool analysis would adequately model the multi-rack fluid coupling effects and provide reasonably acceptable results from the engineering application standpoint. Since the DYNARACK computer code, which is used to model the nonlinear behavior of the racks and fuel assemblies under seismic motion, has not been extensively verified through large scale testing, the staff conducted independent assessments of similar rack responses as part of the re-rack amendment reviews of other applications in order to enhance staff confidence on the validity of the code. These independent assessments form part of the basis for the staff's conclusion that the racks of Salem Units 1 and 2 would function properly and maintain structural integrity during and after the postulated design basis seismic event.

The WPMR analysis code includes an experimentally established variation in the coefficient of friction (COF) between the fuel rack pedestal and the pool floor. Because this COF is indeterminate, the calculation is performed with randomly generated friction values bounded on the upper limit by 0.8 and the lower limit by 0.2. This method of generating the stress factors and rack displacements by incorporating bounding friction values, which is considered a conservative assumption, has been accepted by the staff in several previous re-rack applications.

The WPMR analysis results are presented in Tables 6.8.1 through 6.8.18 of Reference 1, and compared to the single rack analysis in Table 6.8.19. The staff finds that the limiting safety margin in the WPMR analysis is 1.4 and occurs in the pedestal region. This result indicates that adequate conservatism has been incorporated in the rack design. The overall minimum safety margins associated with the calculated "x" and "y"-axis shear and bending stresses reported by the licensee in the WPMR analysis are 6.1 and 3.1, respectively (see Tables 6.8.1 - 6.8.18 of Reference 1). These safety factors are acceptable to the staff.

The overall comparison of the WPMR analysis to the single-rack one shows agreement between the two models, as shown in Table 6.8.19 of Reference 1. Although the evaluations are not exact, the two analyses have adopted different assumptions, parameters, and details of their modelling in an effort to bound the uncertainties associated with the complex rack seismic response. The staff considers the extent of the agreement between the models as good and acceptable.

2.5.2 Fuel Handling Accident Analysis

The licensee submittal postulates two different fuel handling accidents in the submittal which are in accordance with the Salem Updated Final Safety Analysis Report (UFSAR). The staff accepts the assumed cases as conservative representation of the possible in-pool fuel handling accident loadings. Both accidents involve dropped fuel assemblies, the difference being the impact point and failure mode.

The first postulated accident is the deep drop scenario where a 3810 lb. assembly is dropped from 36 feet and impacts the baseplate of the module. This design accident allows for local pool slab deformation but the liner must remain unaffected and catastrophic structural failure is unacceptable. Concrete bearing pressures calculated in the seismic qualification assure the staff that the impact loading can be safely transmitted to the pool slab without rupture to the liner.

The second postulated accident is the same fuel handling accident, but impact occurs at the top of the rack. In this case the licensee states, based on the results of the analysis, that the small deformations which do not affect the structural ruggedness and the criticality levels of the stored spent fuels can be tolerated.

The licensee provides an indication of the margin of safety by determining the amount of permanent deformation if an assembly were to be dropped in each scenario. In the deep drop scenario, the safety margin based on permanent baseplate displacement toward the liner is 4.9. This safety factor states that the space between the baseplate and the liner is 4.9 times the maximum postulated accident induced displacement. The second fuel assembly accident has a safety margin of 3.1, based on the rack deformation distance above the stored fuel. The staff believes that the submitted safety margin calculations and postulated fuel handling accidents are conservative and demonstrate the ability of the components to withstand effects of the postulated impacts. The staff, therefore, concludes that the above discussed fuel drop analyses are acceptable.

2.5.3 Spent Fuel Pool Integrity

The structural integrity of the spent fuel pool was evaluated using a combination of static, dynamic, and thermal loads, combined in accordance with SRP Section 3.8.4. The structural analysis was performed using the ANSYS three-dimensional finite element code. The model incorporates the pool structure, the liner, the steel bearing pads, the effects of localized failures (concrete cracking), and the interaction between adjacent pool walls. The ANSYS computer code used in the structural analysis of the spent fuel pool is acceptable to the staff because the code is a public domain code and has been extensively used in previous analyses of nuclear structures.

Tables 8.5.2 and 8.5.3 of Reference 1 outline the limiting load combinations for the analysis of shear and bending stresses in the ANSYS model. The licensee reported that considering the entire structure with all wall and floor sections, the most limiting safety margin determined was 1.2 for bending strength and 2.2 for shear strength.

Based upon the review of the analysis procedure the loading combinations selected by the licensee and the results of the analysis, the staff concludes that the above described spent fuel analysis results are acceptable.

2.5.4 Conclusions

Based on the information presented in the original submittal and the licensee's responses to staff requests for additional information, the staff finds that the proposed design is adequate to withstand the normal, seismic and accident loading outlined by the Standard Review Plan (SRP), and the USNRC Regulatory Guides 1.13 and 1.92. The staff also concludes that the proposed amendment is in compliance with the licensing commitment set forth in the Salem Generating Station UFSAR Section 9.1, "Fuel Storage and Handling".

It is therefore concluded that the licensee's design and analysis of the proposed re-racking are adequate to withstand the necessary normal, seismic, and accident loadings and are acceptable provided that the licensee commits to implement a post earthquake inspection of the rack configurations (including the gaps) and, as needed, to restore the gaps between the racks and between the racks and walls after occurrence of an earthquake exceeding the Operating Basis Earthquake (OBE).

2.6 Occupational Dose Control

The licensee estimated in its April 28, 1993, application (Reference 1) that total occupational dose for planned reracking operations would be between 6 and 12 person-rem, including any necessary diving activities.

This overall estimate is based on individual dose estimates for each of the series anticipated activities to be performed during the reracking operation. These activities include removing and decontaminating (hydrolasing) the current racks once they are emptied and removed from the fuel pool; removing underwater appurtenances; installing new racks; and preparing the old racks for shipping.

The licensee has indicated that the underwater appurtenances will be removed using remote handling tools to the greatest extent possible. If diving operations are required, careful monitoring and adherence to procedures should ensure that the radiation dose to the divers is as low as reasonably achievable (ALARA). Further, if divers are used, the licensee has committed (Reference 7) to the guidance provided in Appendix A ("Procedures for Diving Operations in High and Very High Radiation Areas") to Regulatory Guide 8.38 (Reference 8).

The radiation protection program at Salem is adequate for the reracking operations. Where there is a potential for significant airborne activity, continuous air samplers will be in operation. Personnel will wear protective clothing and, if necessary, will use respiratory protective equipment. Work activities will be governed by a radiation work permit, and personnel-monitoring equipment will be issued to each individual as needed. As a minimum, this will include thermoluminescence dosimeters and pocket dosimeters. Additional personnel-monitoring equipment (i.e., extremity badges or alarming dosimeters) may be utilized as required. All work activities, personnel traffic, and the movement of equipment will be monitored and controlled to minimize contamination and to ensure that exposures are maintained ALARA. Based on our review of the licensee's application, the staff finds the proposed radiation protection aspects of the spent fuel pool reracking activity acceptable.

2.7 Solid Radioactive Waste

The licensee stated that the spent fuel storage racks will be removed and washed in preparation for packaging and shipment. Estimates of the collective doses associated with this operation were included and found to be acceptable. Shipping containers and procedures will conform to the U.S. Department of Transportation (DOT) regulations and to the requirements of the state through which the shipment may pass, as determine by the State DOT office.

On the basis of its review, the staff finds that the licensee's plan for handling and disposing of solid radioactive waste generated during the planned reracking operation meets regulatory requirements and is, therefore, acceptable.

2.8 Design Basis Accidents (DBA)

In its application, the licensee evaluated the possible consequences of postulated accidents, including means for avoiding them in the design and operation of the facility, and recommended means for mitigating their consequences should they occur. The licensee has evaluated the effect of the changes on the calculated consequences of a spectrum of postulated design basis accidents (i.e, fuel handling accidents) and concludes that the effect of the proposed TS change is small and that the calculated consequences are within regulatory requirements and staff guidelines on dose values. The addition of poison pins or removal of blocking devices will not have any effect on the probability of occurrence of a fuel handling accident. Since the licensee proposes to utilize extended burnup fuel, the staff reevaluated the fuel handling accident for Salem to consider the effect of increased burnups.

In its evaluation for Salem, issued on October 11, 1974, the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in SRP Section 15.7.4.

Since the licensee intends to utilize extended burnup fuel, the staff reanalyzed the fuel handling DBA for this case. The licensee proposes to increase fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T. The licensee had requested approval to extend burnup to 65,000 MWD/T (Reference 1). However, by letter dated April 7, 1994 (Reference 9), the licensee reduced the requested fuel burnup to 60,000 MWD/T. In Table 1, the new and old DBA doses are presented and compared to the guideline doses in SRP Section 15.7.4 (established on the basis of 10 CFR Part 100).

Table 1

Radiological Consequences of Fuel Handling Design Basis Accident (rem)

Thyroid

	Exclusion Area	Low Population Zone
Staff Evaluation October 11, 1974	11	1
Bounding Estimates for Extended Burnup Fuel[1]	13	1.2
Regulatory Guideline (NUREG-0800 Section 15.7.4)	75	75

The staff concludes that the only potential increased doses resulting from the fuel handling accidents with extended burnup fuel is the thyroid doses; these doses remain well within the dose limits given in NUREG-0800 (Reference 5) and are, therefore, acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. By letter dated March 24, 1994,the State official notified the NRC that they had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environemntal assessment and finding of no significant impact have been prepared and published in the Federal Register on May 3, 1994 (59 FR 22871). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

[1] According to NUREG/CR-5009 (Reference 6), increasing fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T increases the doses for a fuel handling accident by a factor of 1.2.

5.0 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.

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REFERENCES

1. "Salem Generating Station, Unit Nos. 1 and 2, Request for Amendment," from S. LaBruna, PSE&G, to U.S. NRC, Document Control Desk, dated April 28, 1993.

2. "Salem Generating Station, Unit Nos. 1 and 2, Update to Spent Fuel Pool Rerack Licensing Report," from S. LaBruna, PSE&G, to U.S. NRC, Document Control Desk, dated August 12, 1993.

3. Teleconference between Dr. A. Solar of Holtec International and Mr. D. Jeng and Mr. J. Stone of the U.S. NRC, January 26, 1994.

4. "Salem Generating Station, Unit Nos. 1 and 2, Response to Request for Additional Information," from S. LaBruna, PSE&G, to U.S. NRC, Document Control Desk, dated February 2, 1994.

5. NUREG-0800, Section 12.6, "Occupational Exposure Associated with the Spent Fuel Pool," November 1984.

6. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Pacific Northwest Laboratory (PNL), 1987.

7. "Salem Generating Station, Unit Nos. 1 and 2, Response to Request for Additional Information," from S. LaBruna, PSE&G, to U.S. NRC, Document Control Desk, dated February 2, 1994.

8. Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," June 1993.

9. "Salem Generating Station, Unit Nos. 1 and 2, Update to Spent Fuel Pool Rerack Licensing Report," from S. LaBruna, PSE&G, to U.S. NRC, Document Control Desk, dated April 7, 1994.