PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Nuclear Department

APR 28 1993

NLR-N93058
LCR 93-02

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

REQUEST FOR AMENDMENT
SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

In accordance with the requirements of 10CFR50.90, Public Service Electric and Gas Company (PSE&G) hereby transmits a request for amendment of Facility Operating Licenses DPR-70 and DPR-75 for Salem Generating Station (SGS), Unit Nos. 1 and 2. Pursuant to the requirements of 10CFR50.90 (b) (1), a copy of this request has been sent to the State of New Jersey as indicated below.

The proposed changes increase the Spent Fuel Pool capacities for Unit Nos. 1 and 2 from the current 1170 to 1632 fuel assemblies, and extend the decay time for Refueling Operations from 100 to 168 hours. This increased capacity is expected to delay the loss of Operational Full Core Reserve capability (defined as 300 storage locations) from 1998 to 2008 and from 2002 to 2012 for Salem Units 1 and 2 respectively.

Attachment A contains further discussion and justification for the proposed changes. Attachment B contains a markup of the existing Unit 1 Technical Specifications to reflect the requested changes. Attachment C contains a markup of the existing Unit 2 Technical Specifications to reflect the requested changes. Attachment D contains the associated Licensing Report.

PSE&G has reviewed the implementation requirements for the proposed amendment and requests a 60 day implementation period after amendment approval.

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Should you have any questions on this transmittal, please contact us.

Sincerely,

S. LaBruna
Vice President -
Nuclear Engineering

Attachments (4)

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REF: NLR-N93058

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

S. LaBruna, being duly sworn according to law deposes and says:

I am Vice President - Nuclear Engineering of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning the Salem Generating Station, Unit Nos. 1 and 2, are true to the best of my knowledge, information and belief.

Subscribed and Sworn to before me
this 28th day of April, 1993

Sherry L Cagle
Notary Public of New Jersey

SHERRY L. CAGLE
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires March 5, 1997

My Commission expires on

ATTACHMENT A

PROPOSED LICENSE CHANGE LCR 93-02
SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

I. Background

Each Salem Unit currently has a wet storage capacity of 1170 cells. Salem's spent fuel storage management program seeks to maintain an Operational Fuel Core Reserve (OFCR) of approximately 300 storage locations in each Spent Fuel Pool (SFP). With the existing storage capacity, loss of OFCR is expected to occur in 1998 and 2002 for Unit 1 and 2 respectively. SFP reracking will increase each SFP's capacity from 1170 cells to 1632 cells and provide an additional 10 years of storage. The expansion entails the retention of (3) existing Exxon Nuclear Corporation modules, containing 300 cells, and adding (9) new Holtec modules containing 1332 cells. The existing Exxon racks and the new Holtec racks are free-standing, austenitic stainless steel modules using Boral as the neutron absorber material. PSE&G plans to realize the SFP capacity of 1632 cells through on-site reracking efforts in 1994 for Unit 1 and 1995 for Unit 2.

After reracking, the Salem SFPs will use a multi-region storage design previously employed in over (20) Pressurized Water Reactor (PWR) SFPs. Region 1 will utilize the (3) existing flux-trap type, high density racks and Region 2 will contain the (9) new non-flux-trap type, maximum density racks. Region 1 can unconditionally accommodate the storage of fresh fuel with a maximum enrichment of 4.25 w/o U-235. Unirradiated and irradiated fuel with initial enrichments up to 5.0 w/o U-235 can be stored in Region 1 with some restrictions. Region 2 can accommodate the storage of unirradiated and irradiated fuels with stricter controls as compared to Region 1.

The latest criticality analysis methodology was used to characterize the acceptable fuel enrichment for the existing Exxon racks and the new Holtec racks. The maximum permissible enrichment, with and without integral fuel burnable absorber (IFBA) rods was established. The maximum allowable enrichment and the number of IFBA rods has been developed to support future fuel management needs with anticipated higher fuel enrichments.

II. Description of Changes

This amendment request extends the decay time for Refueling Operations, LCO 3.9.3, from 100 hours to 168 hours. The

Design Features Section 5.6, "Fuel Storage" is revised as follows:

1. Criticality Section 5.6.1 is replaced with information associated with the storage of unirradiated and irradiated fuel assemblies in: the new fuel vault, existing spent fuel pool high density racks (Region 1), and the proposed spent fuel pool maximum density fuel racks (Region 2).

2. Capacity Section 5.6.3 is revised to indicate a maximum Spent Fuel Pool capacity of 1632 fuel assemblies.

III. Justification for the Proposed Changes

Attachment D contains our Licensing Report, which describes the design and analyses performed to ensure that the new Holtec racks comply with all applicable regulations, codes, and standards, including "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications," USNRC (1978) and 1979 addendum. The safety assessment of the proposed rack modules considered their adequacy in the following areas: criticality, thermal-hydraulic, seismic, structural, and radiological.

Criticality was evaluated for compliance to the USNRC effective neutron multiplication factor limit of 0.95 (OT Position Paper) with assumed 95% probability and 95% confidence. The criticality evaluation of the existing Exxon racks also established the new permissible maximum initial enrichment with and without IFBA credit. Licensing Report Section 4.0 provides a detailed account of this evaluation.

Thermal-hydraulic adequacy is confirmed if the fuel cladding does not fail due to excessive thermal stress, and the steady-state bulk pool temperature remains within the limits that satisfy pool structural strength constraints. The decay time for Refueling Operations limit is revised from the existing 100 hours to 168 hours to support the thermal-hydraulic calculation assumptions. Licensing Report Section 5.0 describes the analyses performed and results.

Structural adequacy primarily involves confirmation, through analyses, that the free-standing rack modules will not impact each other in the cellular region or with the pool walls during a postulated Design Basis Earthquake (DBE) or Operating Basis Earthquake (OBE) event. Rack module primary stresses must remain below the ASME B&PV Code (subsection NF) allowables. Structural qualification also includes analytical demonstration that stored fuel subcriticality is maintained under all UFSAR postulated accident scenarios. The structural/seismic considerations of the new racks, the accident analyses, and the spent fuel

pool structural integrity analyses are presented in Sections 6.0, 7.0 and 8.0 of the Licensing Report.

The Radiological analyses and Boral Surveillance Program are described in Licensing Report Sections 9.0 and 10.0 respectively. A Cost/Benefit analysis that establishes reracking as the most cost effective approach to increasing the on-site spent fuel storage capacity, is included in Section 11.0.

All computer programs that were used to perform the documented analyses are identified in the appropriate Licensing Report Sections. All computer codes were benchmarked and verified per Holtec International's Nuclear Quality Assurance Program. Computer programs and codes have been utilized at numerous plants to support rerack applications.

The completed analyses demonstrate that the rack module arrays possess large margins of safety in the significant review areas: criticality, thermal-hydraulic, seismic, structural, and radiological.

IV. <u>Significant Hazards Analysis Consideration</u>

The proposed Technical Specification changes:

1. Do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   PSE&G has evaluated the following postulated accident scenarios:

   1. A spent fuel assembly drop in the SFP.
   2. Loss of SFP cooling.
   3. A seismic event.
   4. An installation accident during reracking.

   The Salem SFP has been analyzed considering fuel handling equipment, operating procedures, SFP cooling system, and seismic events. Reracking involves replacing 9 out of the 12 existing high density racks with 9 new maximum density racks. It does not require any system modifications or modifications to the cask handling crane, which by its physical location and design is prevented from moving over the SFP. Results confirm that the proposed modification does not increase the probability of the first three postulated accident scenarios.

   NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," sections 5.1.1, 5.1.2, and 5.1.6, provide guidance for heavy load handling operations during

spent fuel storage rack replacement. Section 5.1.2 lists (4) alternatives for assuring safe heavy load handling during a fuel storage rack replacement. Alternative (1) satisfies the control of heavy loads guidelines through the implementation of defense-in-depth measures. These measures ensure that the potential for a heavy load drop is extremely small. PSE&G intends to utilize the defense-in-depth concept during reracking activities.

NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," provides guidance for the design, fabrication, installation, and testing of highly reliable new cranes. NUREG-0612, Appendix C, "Modification of Existing Cranes," provides guidance for the implementation of NUREG-0554 at operating plants. We have evaluated anticipated fuel handling crane movements for compliance with the guidelines specified in alternative (1) of Appendix C, and determined that alternative (1) was satisfied based on the extremely small probability of a storage rack drop. The maximum weight of any storage rack and its associated handling tool is 17 tons. The fuel handling crane will be upgraded to a 20 ton lifting capacity and a design safety factor, with respect to ultimate strength, of five times the lifting capacity (i.e., 100 tons). The uprated fuel handling crane has ample safety factor margin for storage rack movement. This applies to non-redundant load-bearing components. Special redundant lifting devices, which have a rated capacity sufficient to maintain safety factors, will be utilized for storage rack movements. Per NUREG-0612, Appendix B, the substantial safety factor margin ensures that the probability of a load drop is extremely low. Additionally, a load drop analysis was performed to ensure the integrity of the pool structure. The analysis results were acceptable.

Based on the actions discussed above, the proposed modification does not increase the probability of an installation accident.

PSE&G evaluated the consequences of a spent fuel assembly drop in the SFP and determined that the criticality acceptance criterion, Keff less than or equal to 0.95, was not exceeded. The radiological consequences of a fuel assembly drop did not change significantly from those previously analyzed. The calculated doses are well within 10CFR100 requirements. A spent fuel assembly dropped on the racks, will not cause rack distortion that would prevent the performance of their safety function. Thus, the consequences of this postulated accident are not significantly changed from those previously evaluated.

The consequences of a loss of SFP cooling were evaluated. The evaluation concluded that sufficient time is available to establish an alternate means of cooling following a complete failure of the normal SFP cooling system. Calculations show that under a normal discharge scenario, if all indirect forced cooling paths (i.e., heat removal by heat exchangers) are lost at the instant the pool water reaches its maximum value, the pool will not begin bulk boiling for at least 4.61 hours. This time interval is sufficient to allow plant personnel to establish alternate heat removal methods. A piped cross-connection exists between Unit 1 and Unit 2's SFP heat exchangers. This allows for use of the opposite Unit's heat exchanger during emergencies, or when a given Unit's Service Water header or Component Cooling System are out-of-service. Thus, the consequences of this postulated accident are not significantly changed from those previously evaluated.

The new racks are designed and fabricated to meet applicable NRC requirements and industry standards. Seismic analyses were performed on the new racks and the existing racks using 3-D single rack (opposed phase motion) and Whole Pool Multi-Rack (WPMR) models. Kinematic and shear analyses conclude the existence of large margins of safety. The kinematic margin against rack-to-rack or rack-to-wall impact is at least 1.5 for all SFP racks. Maximum rack primary stresses, under SSE conditions, are less than 50% of the allowable ASME Code value. Maximum supporting pool structure bending moments and thru-thickness shear, under factored load conditions, are less than 80% of the allowables. All racks (new and existing) are designed as free-standing racks, to ensure that rack and pool structure integrity is maintained during and after a seismic event. Thus, the consequences of a postulated seismic event are not increased from previously evaluated events.

The consequences of an installation accident were considered. All fuel in the SFP will have decayed for a minimum of (3) months prior to any heavy load movement in the SFP area. This allows sufficient time for decay of gaseous radionuclides in the fuel (gap activity). A postulated accidental gaseous release from all stored fuel assemblies would result in a potential offsite dose less than 10% of 10CFR100 limits. No equipment essential to safe reactor shutdown or employed to mitigate the consequences of an accident is located beneath, adjacent to, or within the area of influence of any load handling to support the SFP modification. Thus, the consequences of a postulated installation accident are not significantly

increased from those previously evaluated.

The only postulated accident affected by decay time is a Loss of SFP cooling. The proposed increase in decay time prior to refueling operations is conservative and decreases the decay heat removal requirements. All thermal-hydraulic calculations used 168 hours as the assumed decay time and concluded that adequate heat removal capability existed. Thus, the probability and consequences of a loss of SFP cooling accident are not significantly increased from those previously evaluated.

Therefore, it may be concluded that the proposed changes do not increase the probability or consequences of an accident previously evaluated.

2. Do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed modification has been reviewed and analyzed for possible accidents. The criteria used in the analyses, design, and installation of the new spent fuel racks account for anticipated loadings and postulated conditions that may be imposed upon the structure during its lifetime, and is in conformance with established codes, standards, and specifications acceptable to the NRC.

Factors that could affect the SFP neutron multiplication factor have been addressed conservatively. PSE&G concluded that the maximum SFP neutron multiplication, with the addition of the maximum density racks, will not exceed the subcriticality limit of Keff less than or equal to 0.95.

The increase in decay time prior to refueling operations reduces the initial heat load and SFP cooling requirements. The addition of new racks and associated spent fuel will produce an incremental heat load in the SFP. However, analysis has shown that the existing SFP cooling system is sufficient to absorb this incremental heat load. The peak bulk pool temperature will be maintained below the threshold value to preclude bulk boiling. The incremental heat load does not alter SFP cooling safety considerations from those previously reviewed and found acceptable.

Rack impact analysis was performed to investigate possible impact during seismic events (i.e., rack-to-rack and rack-to-wall impacts). The analysis concluded that the proposed SFP modification does not result in rack-to-rack impact in the cellular region or

rack-to-wall impact during postulated seismic events.

The basic SFP reracking technology has been reviewed and approved by the NRC in numerous applications for spent fuel capacity increases. The safety function and operation of the SFP cooling system, makeup, and structural systems are unchanged by this modification. No new failure modes are created.

Therefore, it may be concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do not involve a significant reduction in a margin of safety.

The safety function of the SFP and the racks is to preclude inadvertent criticality in a safe, specially designed, underwater storage location for spent fuel assemblies that require shielding and cooling during storage and handling. The NRC Staff has established that the issue of margin of safety, when applied to reracking modifications, should address the following areas:

1. Nuclear criticality considerations.
2. Thermal-hydraulic considerations.
3. Mechanical, material, and structural considerations.

Assessment in these areas assures that the SFP and racks will withstand specified design conditions, without impairment of the structural integrity or performance of required safety functions.

The criticality analysis confirms that the new and existing rack designs meet the NRC acceptance criterion of Keff less than or equal to 0.95 under all conditions. The criticality analysis methods conform to applicable industry codes, standards, specifications and NRC guidance. Keff calculations include uncertainties at a 95%/95% probability confidence level. Thus, the proposed amendment does not involve a significant reduction in the nuclear criticality margin of safety.

Conservative methods and assumptions were used to calculate the maximum fuel temperature and the increase in SFP water temperature. The thermal-hydraulic evaluation employed methods previously used to evaluate the existing spent fuel racks. The results demonstrate that the temperature margins of safety are maintained. The proposed modification, with the fuel inventory,

will increase the heat load in the SFP. However, the decay time prior to refueling operations was increased from 100 to 168 hours to reduce the initial SFP cooling requirements. Evaluation results indicate that the existing SFP cooling system can maintain the bulk pool water temperature at or below 149 F under normal discharge scenarios. The maximum allowable temperature for bulk boiling is not exceeded for the calculated increase in pool heat load. Maximum local water temperatures, along the hottest fuel assembly, remain below the nucleate boiling condition. While no nucleate boiling is indicated for the standard storage condition, an assumption of 50% cell blockage results in a possible highly localized two-phase condition near the top of the fuel. Fuel clad thermal stresses remain less than 7000 psi, which is considerably lower than the endurance limit of the clad material. Thus, there is no significant reduction in the margin of safety for thermal-hydraulic or SFP cooling.

Maintaining the spent fuel assemblies in a safe configuration during normal and abnormal loadings is the primary safety function of the SFP and racks. Abnormal loading associated with an earthquake, a spent fuel assembly drop, or the drop of any other heavy object were considered. The mechanical, material, and structural design of the new spent fuel racks complies with applicable portions of the NRC OT Position Paper. Rack materials are compatible with the spent fuel pool environment and the spent fuel assemblies. The structural assessment of the new racks concluded that tilting and deflection or movement will not result in impact in the active fuel region during postulated seismic events. In addition, the spent fuel assemblies remain intact with no criticality concerns. Thus, there is no significant reduction in the margin of safety for mechanical, material and structural considerations.

Therefore, it may be concluded that the proposed changes do not involve a significant reduction in a margin of safety.

IV. Conclusions

Based on the information presented above, PSE&G has concluded that the proposed Technical Specification changes satisfy the criteria for a no significant hazards consideration.