Public Service
Electric and Gas
Company

Stanley LaBruna

Public Service Electric and Gas Company P.O. Box 236, Hancocks Bridge, NJ 08038 609-339-1700

Vice President - Nuclear Engineering

NOV 17 1993

NLR-N93164
LCR 93-02

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

Gentlemen:

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
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 a letter dated September 13, 1993, the NRC Staff transmitted a request for additional information regarding Public Service Electric and Gas Company's (PSE&G) request to increase spent fuel pool capacity through the installation of new spent fuel pool storage racks. The specific questions involved control of heavy loads and thermal hydraulic considerations. PSE&G has provided responses to the identified questions in the enclosed attachment.

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

Sincerely,

290073

A001

C Mr. T. T. Martin, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. J. C. Stone, Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. C. Marschall (S09)
USNRC Senior Resident Inspector

Mr. K. Tosch, Manager, IV
NJ Department of Environmental Protection
Division of Environmental Quality
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

REF: NLR-N93

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 17th day of November, 1993

Kimberly Jo Brown
Notary Public of New Jersey

My Commission expires on KIMBERLY JO BROWN
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 21, 1998

ATTACHMENT

REQUEST FOR ADDITIONAL INFORMATION 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

A. HEAVY LOADS

1. NRC QUESTION

The method you have chosen for complying with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," is not clearly stated. Compliance with the guidelines of NUREG-0612 requires either evaluating potential heavy load drops to satisfy the criteria of Section 5.1 of NUREG-0612 or showing that the potential for a load drop is extremely small by employing a single failure proof heavy load handling system which complies with the guidelines of Section 5.1.6, "Single Failure-Proof Handling Systems." Describe how the Salem Nuclear Generating Station adequately complies with either set of guidelines.

1. PSE&G RESPONSE

PSE&G has evaluated the potential for a heavy load drop per NUREG-0612, Section 5.1, as discussed in sections A thru C below. Although we are not employing a single-failure proof crane, a review of the crane structural analysis and rerack procedures confirmed substantive compliance with the recommendations of NUREG-0612, Appendix C, as discussed in section D below.

A. NUREG-0612, Section 5.1 "Recommended Guidelines"

Criterion I and II - During the reracking, heavy loads (greater than 2200 pounds) will not be carried over the racks with stored spent fuel. Thus, these Criterion are met.

Criterion III - The consequences of a heavy load drop in the Spent Fuel Pool are limited to water loss resulting from potential concrete structure damage. The limiting case is a heavy load drop from above the water surface to the slab. Analysis results concluded that gross section failure of the slab would not occur, any slab damage would be strictly localized and that no uncontrolled water loss would occur. Note that the Salem Spent Fuel Pool slab is not elevated, so overall slab collapse is not possible during the drop scenario.

Criterion IV - All reracking activities will be carried out along the defined safe load path. Since safe-shutdown equipment is not located in the vicinity of the safe load path, this Criterion is met.

B. NUREG-0612, Section 5.1.1 "General"

Refer to our previously submitted Licensing Report Section 2.5 and Table 2.5.1 for a discussion of compliance to the guidelines indicated in Section 5.1.1.

C. NUREG-0612, Section 5.1.2 "Spent Fuel Pool Area - PWR"

PSE&G has reviewed the additional guidelines specified in (3) of this section. At no time during lifting or lowering is the rack carried over a rack containing spent fuel. As noted in our previously submitted Licensing Report, spent fuel in adjacent racks is located as far as practical from the designated lifting area. Due to the size of the Spent Fuel Pool, it is impractical to <u>always</u> maintain loads at a distance greater than 25 feet from "hot" fuel. Rerack operations will be carried out in a manner consistent with all previous Holtec reracks. All rack movements into and out of the designated lifting area will be carried out underwater, with the lowest point in the rack positioned just above the fuel pool slab. This confines the effects of an uncontrolled drop to the local area under the rack pedestals and minimizes the drop distance. Analysis of a rack drop from the top of the water surface to the slab floor demonstrated no uncontrollable consequences. Additionally, procedural guidance will direct the crane operator to exercise caution when lifting racks into and out of the Spent Fuel Pool.

D. To assure added conservatism, the crane structural analysis and rerack procedures were compared to NUREG-0612, Appendix C requirements for upgrading cranes to single-failure proof status. The results of that review are presented below.

During a Salem Spent Fuel Pool rerack in 1980, the semi-gantry crane structure was analyzed by the crane manufacturer and shown to meet CMMA-70 stress allowables with a 20 ton lifted load. The hoist was uprated to 20 ton capacity, but was subsequently replaced by the original, lower-rated hoist.

This proposed rerack involves no modification to the existing crane other than the installation of a new bridge trolley and upgraded hoist. Nevertheless, the previous CMMA crane analysis was reviewed and reconfirmed. A finite element analysis was performed to demonstrate that a 20 ton lifted load produces primary stresses less than or equal to both 1/5 of the ultimate strength, and 1/3 of the yield strength of the crane material. The concrete corbel that supports the rail on the east side of the crane was evaluated using ACI 318-71 guidelines, with the factored dead plus live load applied on the crane. The analysis concluded that the ACI 318-71 factored load (based on a 20 ton dead load) did not exceed the corbel design strength. These structural calculations were performed to provide additional assurance that the load drop potential due to overstress is extremely small.

Salem Licensing Report pages 2-8 and 2-9 list additional procedural requirements imposed to provide substantive compliance with the applicable recommendations of NUREG-0612, Appendix C (even though these recommendations deal specifically with uprating an existing crane to single-failure proof status). Thus, additional "defense-in-depth" protection against a heavy load drop is provided wherever feasible.

E. In summary, NUREG-0612 requirements are met both by evaluating the consequences of a heavy load drop, and by minimizing the potential for such a drop.

2. NRC QUESTION

By letter dated August 12, 1993, in the update to the spent fuel pool rerack submittal from Public Service Electric and Gas Company (PSE&G), it was stated that PSE&G completed calculations to support Fuel handling Crane uprating. Describe how these calculations comply with the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

2. PSE&G RESPONSE

As stated in the response to Question 1, NUREG-0612 requirements are met by demonstrating that the consequences of a heavy load drop during new rack installation and old rack removal do not result in radiological safety concerns. Additional conservatism is provided through stress and mechanical evaluation of the crane and hoist support structure.

Structural modifications to the existing semi-gantry crane are not contemplated. A new hoist with a rated load of 37.5 metric tons will be installed for the rerack. This hoist is designed per relevant hoist design standards to have a safety factor of 5. Since the heaviest lifted load consists of an existing fuel rack (33800 pounds) plus approximately 3000 pounds for the lift rig and rigging, the actual hoist safety factor during the rerack is 11.21 [37.5 x 2200 x 5 / 36800].

The hoist is attached to a single beam trolley with wheeled end supports designed to ride on the existing semi-gantry crane rails. Note that the Salem crane structure moves longitudinally along the pool length, while the hoist moves on rails located on top of the main crane cross beams across the pool width. We will only use the new trolley and hoist for lifting heavy loads during the rerack.

For conservatism, the new temporary beam trolley was analyzed as a special lifting device using the acceptance criteria of ANSI N-14.6. The trolley produces primary stresses that are less than or equal to both 1/10 of the ultimate strength and 1/6 of the yield strength of the material. CMAA-74 allowable stress requirements were invoked for local flange stresses in the trolley beam. A design load of 20 tons was used. We will test the new beam trolley structure to 125% of the rated load.

Since we are not modifying the main semi-gantry crane structure, and the crane has been previously qualified and used for a 20 ton lifted load, our calculations merely reconfirmed the crane capacity. The following results were obtained:

* The corbel that supports one of the main crane rails was analyzed per ACI 318-71, assuming the most limiting 20 ton lifted load location on the crane. Safety margins of 5 or more were computed.

* Five times the lifted load results in crane structure primary stresses below 90% of the ultimate strength. Three times the lifted load results in crane structure primary stresses below the yield strength.

* All CMAA-70 provisions for allowable stress are met with the 20 ton lift design load.

The supporting calculations demonstrate that the potential for heavy load drop due to crane structure component overstress is extremely small.

3. NRC QUESTION

A. How are dynamic loads included in the design?
B. What maximum load can the rig hold before the minimum yield stress is reached?
C. What is the maximum load the rig can hold before reaching the minimum ultimate stress?
D. Do you plan to use the lifting rig for some time after reracking Unit 1 & 2 SFPs? If so, discuss your plans for maintaining compliance with ANSI N14.6-1978 (see Section 5.3, "Testing to Verify Continuing Compliance," of ANSI 14.6).

3. PSE&G RESPONSE

A. Note that our lift rigs are a proven design most recently used in the successful reracking effort at the D.C. Cook Nuclear Power Plant. The existing racks at Salem are identical in design to those at D.C. Cook.

The lift rig design accounts for dynamic loads, by demonstrating that a load that just meets the ANSI N14-6 stress limits for special lifting devices, is greater than 1 + 0.2 times the required dead load. The 0.2 bounds all expected dynamic load increments. The maximum lifted load for existing racks is 33800 pounds. The maximum lifted load for new racks is 21100 pounds.

B. The maximum calculated lift rig load for existing racks, which results in a primary stress anywhere in the rig equal to the minimum yield stress, is

$$P_{EY} = 128109 \text{ lbs.}$$

The maximum calculated lift rig load for new racks, which results in a primary stress anywhere in the rig equal to the minimum yield stress, is

$$P_{NY} = 84824 \text{ lbs.}$$

The associated margins are (including the dynamic load factor in the required margin):

SM (existing) = 3.79 > 3.6 *
SM (new) = 4.02 > 3.6 *

* 3(1+.2) = 3.6

SM = calculated load to reach limit / dead load to be lifted

C. The calculated loads that result in primary stresses reaching the minimum ultimate strength at the respective locations, were obtained by amplifying loads discussed in part B by the ratio of ultimate stress to yield stress for the material at the limiting location in the lifting rig.

For an <u>existing</u> <u>rack</u> lift, the limiting load is

$$P_{EU} = P_{EY} \times 58000/36000 = 206398 \text{ lbs.}$$

For a <u>new</u> <u>rack</u> lift, the limiting load is

$$P_{NU} = P_{NY} \times 60000/35000 = 145413 \text{ lbs.}$$

The associated margins are:

SM (existing) = 6.106 > 6 *
SM (new) = 6.89 > 6 *

* 5(1+.2) = 6

Note that limiting stresses occur at different locations in the lifting rig. The yield and ultimate stresses used in the calculations above reflect the actual material properties at those locations.

The margins calculated in B and C are in excess of lift rig requirements where a single failure cannot lead to an uncontrolled drop. We have included the dynamic effects by increasing the required minimum margins rather than increasing the lifted loads.

D. At the present time, there is no plan to retain the Salem lift rigs for re-use.

4. NRC QUESTION

On page 2-13 of the Holtec Report HI-92950, it was stated that PSE&G's contractor, Holtec International, plans to develop over twenty operating procedures to cover the entire gamut of operations pertaining to the rerack effort, including but not limited to, mobilization, rack handling, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA compliance. Since the rerack efforts are planned to commence in 1994 for Unit 1 and 1995 for Unit 2, when will these procedures for handling both the old racks and new racks be developed?

4. PSE&G RESPONSE

The procedures for handling both old and new racks have been developed by Holtec and are undergoing review and comment by PSE&G. These procedures are presently in the second review cycle. All procedures will be approved prior to the reracking activities.

B. THERMAL HYDRAULIC CONCERNS

1. NRC QUESTION

In Section 9.1.3.1 under the design basis of the spent fuel pool cooling system in the updated FSAR for Salem Nuclear Generating Station, it is stated that the SFP water is normally limited to 120 F except in the unloading of a full core, in which case the temperature is limited to 150 F with one pump in operation. However, on page 5-2 of the Holtec Report HI-92950, it is stated that the SFP water is limited to 180 F with one pump in operation in the unloading of a full core. Justify the deviation from the design basis for the increase in the SFP water temperature. Provide assurance that these values are not inconsistent or make the necessary changes in the updated FSAR in accordance with 10 CFR 50.59.

1. PSE&G RESPONSE

Calculations presented in the Licensing Report are intended to determine the upper bound of bulk pool water temperature under the most conservative set of assumptions. These include:

* Reactor core power was assumed at 3600 MWT instead of the plant rated 3411 MWT.

* Refueling batch size was assumed at 68 assemblies instead of the typical 65 assemblies.

* All spent fuel assemblies were assumed at 4.5 years of effective full power operation.

* Component cooling water temperature was increased from 95 F to 99 F.

* The heat exchanger was assumed fouled to its design maximum with 5 % of the tubes assumed plugged.

* Maximum pump heat generation was assumed.

Also, the decay heat calculation considered 24 refueling cycles of spent fuel inventory (1592 assemblies) plus the final discharges to accumulate up to 1853 assemblies. The in-core decay time for the final discharge (refueling batch or full core offload) was assumed to be 168 hours. As a result, the maximum pool water temperature following a normal discharge is 149 F at 195 hours after shutdown, and the maximum pool water temperature following a full core offload is 180 F at 205 hours after shutdown.

The original discharge scenarios, as documented on page 9.1-10 of the Salem UFSAR, are restated below for comparison.

* At the 18th refueling, the maximum normal pool water temperature following the discharge of 65 assemblies with 150 hours of decay is 134 F with one pump running.

* For full core offload condition with 15 prior refuelings, the required decay time to keep the pool water temperature below 150 F with one pump running is approximately 570 hours (24 days) after reactor shutdown.

Note that the full core offload case described in the UFSAR envisaged a long in-core decay time.

The bounding calculations provide a measure of the cooling system capacity, rather than the actual values that occur during Salem plant operation. We plan 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.

As part of our normal design change process, PSE&G intended to revise the Salem UFSAR to reflect the new calculated bounding temperatures under the new bounding conditions.

2. NRC QUESTION

The installed rack capacity in each pool is 1632. On page 5-8 of the Holtec Report HI-92950, it is stated that the assumption of the gross count of 1853 assemblies implies that 221 locations (1853 minus 1632) hold consolidated (with 2:1 consolidation ratio) canisters. Describe the consolidation process. Has this process been found acceptable? What is the maximum decay heat to be generated during the process?

2. PSE&G RESPONSE

Conceptual approaches to fuel consolidation at Salem have not been considered or developed. Our license application does not seek authorization to consolidate fuel. The decay heat calculation was conservatively completed using a larger fuel quantity than the installed capacity would permit. To explain the difference between the assumed (higher) inventory of 1853 cells (1592 after cycle 24 + 68 normal discharge + 193 full core offload) and the actual capacity of 1632, the allusion to potential future increases in stored inventory through fuel consolidation was made. In the context of this submittal, the assumption of increased fuel inventory may be considered another element of analysis conservatism.

3. NRC QUESTION

The updated FSAR states that the Spent Fuel Cooling System provides the cooling capacity required for both the annual discharge of 65 fuel assemblies and for a full core discharge of 193 fuel assemblies into the spent fuel pool after 15 years accumulation of spent fuel. The Holtec report HI-92950 states that the normal discharge is 68 fuel assemblies. Provide assurance that these values are not inconsistent or make the necessary changes in the updated FSAR in accordance with 10 CFR 50.59.

3. PSE&G RESPONSE

Effective and efficient core management suggests a typical reload batch size of 64 fuel assemblies. However, the reload batch size in certain cycles may reach as high as 68 fuel assemblies. To bound the calculated results, 68 was used as the normal discharge batch size.

As part of our normal design change process, PSE&G intended to revise the Salem UFSAR to reflect these changes.

4. NRC QUESTION

Standard Review Plan 9.1.3 provides guidelines for cooling and sources of make-up water to the SFP. (1) The SFP cooling system should be designed to seismic Category I requirements or (2) the fuel pool make-up water source, its source, the fuel pool building and its ventilation and filtration system should be designed to seismic category I requirements. The Holtec Report HI-92950 does not specify the SFP cooling system as seismic Category I or specify a make-up water system and its source as seismic Category I. Discuss how the decay heat removal function for the SFP is assured during a seismic event. What effect does reracking have on the capability to remove decay heat from the SFP following a DBE?

4. PSE&G RESPONSE

Even though the Spent Fuel Pool Cooling System is not designed as seismic Category I, four redundant makeup water sources are available to the Spent Fuel Pool (SFP). These sources are: (1) Demineralized Water System, which is the normal source of makeup; (2) Primary Water Storage Tanks; (3) Chemical and Volume Control System holdup tanks; and (4) Refueling Water Storage Tanks (RWSTs).

The RWSTs are seismic Category I. They have spare nozzles where a portable pump can be connected to provide makeup water. The spare nozzles are normally isolated by valves that are locked closed, capped, and under administrative control. The portable pump, along with the associated hose and hose connections, are also under administrative control to ensure constant and timely availability.

This information is described in UFSAR Section 9.1.3.3 and in our Licensing Report Section 5.2.4.

The existing Spent Fuel Pool Cooling System design was reviewed and approved by the NRC in Salem SER Section 9.5 dated 10/11/74. The Spent Fuel Pool rerack does not alter the original system design or the associated makeup water sources. Note that Salem Units 1 and 2 Spent Fuel Pool Cooling Systems were not originally designed to comply with the NRC Standard Review Plan.