U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos. 50-272/82-35
50-311/82-32

Docket Nos. 50-272
50-311

License Nos. DPR-70
DPR-75 Priority -- Category C

Licensee: Public Service Electric and Gas Company

80 Park Plaza

Newark, New Jersey 07101

Facility Name: Salem Nuclear Generating Station - Units 1 and 2

Inspection at: Newark, New Jersey and Hancocks Bridge, New Jersey

Inspection Conducted: November 30 - December 10, 1982

NRC Personnel: for L. E. Tripp 1/13/83
W. J. Lazarus, Project Engineer date

P. K. Eapen 1/13/83
P. K. Eapen, Reactor Inspector date

Approved by: L. E. Tripp 1/13/83
L. E. Tripp, Chief, Reactor Projects Section No. 2A, Reactor Projects Branch No. 2 date

Inspection Summary: Inspection on November 30 - December 10, 1982 (Combined Report Numbers 50-272/82-35 and 50-311/82-32
Areas Inspected: Routine inspection by two Region-based inspectors (35 hours) of licensee actions taken to comply with the plant shielding design review required by NUREG 0737 Item II.B.2.2.

Results: No violations were identified.

DETAILS

1. Persons Contacted

R. Douglas, Manager - Licensing and Analysis
*L. Reiter, Manager - Nuclear Systems Engineering
R. Rippe, Assistant General Manager - Engineering
M. Rosenzweig, Assistant General Manager - Engineering
*T. Taylor, Manager - Nuclear Engineering Control
R. Yewdall, Principal Staff Radiation Analyst
R. Ziegler, Senior Engineer

*Denotes those present at the exit interview.

2. Plant Shielding Design Review

a. Background and Scope

As discussed in NUREG-0737, "Clarification of TMI Action Plan Requirements," each power reactor licensee was required to perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, each licensee was required to provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review was to determine which types of corrective actions were needed for vital areas throughout the facility.

These requirements were originally issued by NRC letters to all operating nuclear power plants, dated September 13 and October 30, 1979, and were incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in requirements or guidance were described in NUREG-0737, Item II.B.2. In the case of Salem Units 1 and 2, the shielding design review and corrective actions were discussed by the licensee in a letter to the NRC dated July 8, 1980.

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. The review included a sampling verification of the shielding design review, methodology, and representative calculations; a review of selected emergency procedures to determine if the vital areas where personnel must go are safely accessible; and a review of corrective actions taken or planned by the licensee, including plant modifications.

b. Shielding Design Review Verification

The licensee's shielding design review methods were reviewed, including source terms, calculation of dose rates, calculation of dose to personnel during post accident access to vital areas, and acceptance criteria. The licensee submitted a summary of the Shielding Design Review to the NRC on July 8, 1980.

The inspector discussed the details of the Shielding Design Review with the licensee's representatives. The licensee's representatives provided the assumptions and methodology used in shielding calculations and the results of such calculations. The assumptions were consistent with the guidelines of NUREG 0737. The methodology and the mathematical model employed state-of-the-art techniques for shielding design. The licensee's calculation models were independently verified by an outside consultant. The dose rates obtained from licensee's calculations were compared and found to be consistent with those developed by the NRC's consultants for similar configuration.

The inspector noted that the licensee's shielding analyses addressed the dose rates for all radioactive fluid carrying systems. The licensee completed an accessibility study to determine which areas in the plant need to be accessed during various accident scenarios. The conclusions of this study are presented in the licensee's document SGS/M-NM-077 dated August 12, 1980. The licensee also used Westinghouse Owner's Group Emergency Response Guidelines for determining access to systems and components carrying radioactive fluids.

The licensee has installed the required shielding to reduce the doses to acceptable levels in areas where access was deemed necessary (See Inspection Report 50-272/82-06, 50-311/82-05). The shielding for the post accident sampling system is being installed. The adequacy of the shielding for the post accident sampling system will be addressed in the review of NUREG-0737 item II.V.3. The adequacy of electrical components will be addressed in the review of licensee's IE Bulletin 79-01B submittal.

The inspector noted that an independent review of the shielding design calculations was not performed in a manner consistent with the licensee's commitments to meet ANSI N45.2.11 and Regulatory Guide 1.64. The licensee's representatives stated that the shielding design calculations would be reviewed in accordance with the requirements of Regulatory Guide 1.64, as modified by the NRC staff position stated in Section 17.1 of the Standard Review Plan (Rev. 2).

The inspector also noted that the licensee did not have specific procedures to address the shielding design calculations. The licensee attributed this to the recent reorganizations. The licensee's representatives stated that the required procedures would be developed with the assistance of corporate Quality Assurance personnel by March 31, 1983.

The inspector informed the licensee that the progress of the above action items would be followed in future NRC inspections (50-272/82-01).

Expect for the items noted above the inspector had no further questions in this regard.

c. Vital Area Accessibility

The inspector reviewed Emergency Instructions (EI) I-4.4, "Loss of Coolant," Rev. 11, I-40, "Safety Injection Initiation", Rev. 3, and II-15.3, "Hydrogen Recombiners-Normal Operation," which would be implemented by the licensee in the event of a loss of coolant accident. The review included a walkthrough of portions of the procedure where access to valves or components in potential radiation areas was required.

Based on this review, the inspector determined that the procedures were technically adequate and contained appropriate provisions to assure controlled access to vital areas and keep post-accident doses to plant personnel within the guidelines of NUREG-0737. The inspector noted the areas where shielding had been added and where modifications had been completed to allow remote operations of valves and components from the Control room.

The inspector had no further questions regarding vital area accessibility.

3. Exit Interview

A meeting was held with facility management on December 10, 1982, to discuss the inspection scope and findings as detailed in this report.