U.S. NUCLEAR REGULATORY COMMISSION
REGION 1

Report No. 50-322/83-14

Docket No. 50-322

License No. CPPR-95 Priority -- Category B

Licensee: Long Island Lighting Company

175 East Old Country Road

Hicksville, New York 18801

Facility Name: Shoreham Nuclear Power Station

Inspection Shoreham, New York and Boston, Massachusetts

Inspection Conducted: April 27-29, and May 6, 1983

Inspectors: E. Kelly 5/20/83
E. Kelly, Reactor Engineer date

Approved by: Robert M. Gallo 5/23/83
R. M. Gallo, Chief, Reactor Projects Section No. 1A, Branch No. 1. Division of Projects and Resident Programs date

Inspection Summary:
Inspection on April 27-29, and May 6, 1983 (Report No. 50-322/83-14)

Areas Inspected: Special safety inspection by a region-based inspector (21 hours) of licensee actions taken to comply with requirements discussed in NUREG-0737, Item II.B.2, Design Review of Plant Shielding. The inspection consisted of onsite review of emergency procedures and plant areas designated as vital, and a review of calculations and assumptions at the Boston office of Stone and Webster Engineering Corporation.

Results: No violations were identified.

DETAILS

1. Persons Contacted

The below-listed technical and supervisory level personnel were among those contacted:

Long Island Lighting Company (LILCO)

G. Gogates, Technical Support Compliance Engineer
I. Haas, Radiation Protection Engineer
*J. Smith, Manager, Special Projects
*R. Wittschen, Licensing Engineer

Stone and Webster Engineering Corporation (SWEC)

A. Auciello, Radiation Protection Engineer
*F. Baldwin, Assistant Manager, Quality Assurance
*F. Bamdud, Lead Nuclear Technical Engineer
*F. Berl, Licensing Engineer
*W. Eifert, Chief Engineer, Engineering Assurance
R. Gauthier, Principal Nuclear Engineer
*P. Holden, Project Engineer
S. Ingeneri, Supervisor, Radiation Protection
*R. Misiaszek, Lead Licensing Engineer
*R. Najuch, Principal Mechanical Engineer
S. Wakefield, Lead Power Engineer
*J. Webb, Engineering Assurance Engineer

USNRC

J. Higgins, Senior Resident Inspector
**C. Hinson, Radiological Assessment Branch, NRR

*Denotes those present at the exit interview on May 6, 1983 at SWEC, Boston, Massachusetts.

**Indicates telephonic contact

2. Plant Shielding Design Review

a. Background and Scope

As discussed in Item II.B.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," each power reactor licensee was required to perform a radiation and shielding design review of 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 present during post-accident operation 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.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area.

These requirements were discussed in Item 2.1.6.b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Requirements"; were issued by NRC letters dated September 13 and October 30, 1979 to all licensees; and were incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in requirements or guidance were described in an NRC letter to all licensees dated September 5, 1980, and were subsequently described in Item II.B.2 of NUREG-0737.

With respect to documentation specified by NUREG-0737 for vital area access, operating license applicants were to provide to the NRC a summary of the shielding design review, a description of the results of this review, and a description of the modifications made or to be made to implement the results of the review. The submittals were to include:

(1) Specification of source terms used in the evaluation, including time after shutdown that was assumed for source terms in systems;

(2) Specification of systems assumed in the analysis to contain high levels of radioactivity in a post-accident situation;

(3) Specification of areas where access is considered necessary for vital system operation after an accident; and,

(4) The projected doses to individuals for necessary occupancy times in vital areas and a dose rate map for potentially occupied areas.

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. This review included licensee submittals to the NRC, 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. Licensee Submittals to the NRC and Previous Staff Evaluation

In the case of Shoreham Nuclear Power Station, Unit 1, the shielding design review was first discussed by Long Island Lighting Company (LILCo or the licensee) in their August 29, 1980 (SNRC-503) letter to the NRC which responded to the Category "A" Lessons Learned for this item (NUREG-0578, Item 2.1.6.b).

Subsequent licensee letters dated May 15 (SNRC-563), June 11 (SNPS-586), and July 22 (SNRC-602), 1981 to NRC addressed NUREG-0737, Item II.B.2. The licensee's position, and a summary of the shield design review, were incorporated into the Shoreham FSAR, Volume 16, in Revision 22 dated July 1981. The NRC staff evaluated LILCo's review and found the licensee's response to Item II.B.2 acceptable for full power operation in Section 22.2, "TMI Action Plan Requirements for Applicants for Operating Licenses", of the Shoreham Safety Evaluation Report, NUREG-0420, Supplement 1, dated September 1981.

The above licensee and NRC documents were reviewed during this inspection to determine the licensee actions completed or to be taken, and the extent of previous staff evaluation, regarding the plant shielding design review for Shoreham. The licensee's statements and commitments contained in these letters provided the basis, in part, for the inspector's verification that plant modifications have been adequately identified and implemented to allow access to required vital areas, as discussed in this report.

c. Shielding Design Calculations and Dose Estimates

The inspector reviewed the details of the licensee's shielding design review calculations with licensee representatives (plant architect-engineer, Stone and Webster Engineering Corporation) in Boston, Massachusetts on May 6, 1983. These details included the mathematical models, assumptions, source terms, dose rates and doses to personnel during post accident access to vital areas. The licensee employed five computer codes to calculate doses: RADIOISOTOPE, DRAGON 4, COHORT II, ANISN"D", and QAD-MOD.

RADIOISOTOPE calculates isotopic activities as a function of time, given initial (time-zero) activities, by solving appropriate decay-purification equations. This code was used to obtain specific source term gamma energy activity for each of seven fixed energy groups.

DRAGON 4 evaluates activities, dose rates, and integrated doses at some vicinal site (such as in the reactor building or control room) following either an instantaneous or continuous release of radioactive halogens and noble gases from a control volume. The doses calculated are for airborne activity or cloud immersion.

COHORT II is a three-dimensional Monte Carlo general purpose shielding program which calculates unscattered and scattered fluence for primary and secondary photons. This code was used to calculate "skyshine" (air-scattered) or indirect radiation from atmospheric reflections in the vicinity of (and from) the reactor building.

ANISN "D" solves the one-dimensional Boltzmann transport equation for gamma rays in slab, sphere, or cylindrical geometry, using an advanced discrete ordinates numerical method and an 18 group P_{12} cross section set. This code was used to model sources requiring near-surface dose rates such as fluid-carrying pipes (suppression pool water) and direct shine from gaseous releases or the plume to control room and the TSC. The code calculates direct and scattered gamma "shine" from such sources.

QAD-MOD is a point-kernel, general purpose, three-dimensional gamma shielding code used to calculate dose rates at a series of detector locations for source points representative of volumetric sources. This code was used to model volume sources, including certain components (CRAC and RBSVS filters) and the reactor building, and to calculate direct gamma shine (with buildup) from such sources.

The licensee utilized the prescribed post-accident source terms described in Regulatory Guides 1.3 and 1.7 (as specified in NUREG-0737) to perform the radiation and shielding design review. The licensee considered two accident scenarios, a LOCA and a pipe break in the secondary containment, along with the above mentioned-source terms to calculate post-accident gamma radiation fields at Shoreham. The radiation sources considered in calculating these dose rates included: 1) direct radiation from airborne and liquidborne radioactivity in the primary (drywell) and secondary containment (reactor building); 2) direct radiation shine from the externally-released plume, and system piping and components in the primary and secondary containment; 3) immersion dose rates from airborne radioactivty due to primary containment and equipment leakage; and 4) skyshine from the upper position of the reactor building.

The inspector also reviewed licensee documents which demonstrated how doses to personnel during post-accident access to vital areas were maintained within the guidelines of GDC-19 and NUREG-0737, Item II.B.2. The licensee's shielding design review methodology and dose rate calculations were acceptable, and the inspector had no further questions in these areas.

d. Specification of Vital Areas - Location and Doses

The control room, the Technical Support Center (TSC), and the Post Accident Sampling and Analysis Facility (PASF) are all areas which will require continuous or frequent occupancy following an accident at Shoreham and are considered vital areas. The licensee evaluated other areas suggested as vital post-accident areas in NUREG-0737,

but concluded that these areas either do not apply to Shoreham or their functions can be controlled/monitored from the main control room at Shoreham. The control room and permanent TSC will have post-accident dose rates of less than 15 mRem/hr (30-day average). The integrated dose to these areas for the duration of the accident will be less than 5 Rems, whole-body, as specified in GCD 19.

The PASF can be accessed from any of the Emergency Response Facilities without passing through areas with radiation fields which could potentially add incremental whole body exposures of concern (i.e. appreciable contributions to the 5 Rem limit). The PASF is accessible from outside the Reactor Building (as well as from inside) and provides capability to sample and analyze reactor coolant and containment atmosphere. The inspector verified that no unshielded sample lines were present such that an individual taking a sample would be exposed to direct high-level radiation sources.

The newly constructed, permanent TSC is an annex to the main office building and is both appropriately shielded (18-inch concrete walls and ceiling) and easily accessible from either the PASF or control room. Stone and Webster calculation SNPS-1-UR-21-Q, "Permanent TSC Dose Rates and Doses" approved on August 6, 1981 was reviewed and discussed with its authors. Principal contributors to the dose inside were the overhead outside plume and the airborne cloud inside the room; both were on the order of tens-of-mR/hr (with a combined maximum value less than 50 mR/hr at approximately 6 hours after the accident). Levels dropped to less than 15 mR/hr after approximately 48 hours.

The control room dose calculations were reviewed to determine the dominant source contributors. Airborne radiation inside the room is, by far, the major contributor and even this valve is relatively low (SWEC calculation SNPS-1-UR-21-A approved September 10, 1981 indicates a maximum dose rate on the order of 10-20 mR/hr at between 4 and 10 hours post-accident). During the site inspection on April 27-29, 1983, the inspector raised the question as to the impact of the control room's emergency air conditioning and ventilation (CRAC) system standby filter train as a potential radiation source affecting the control room. In discussions with SWEC representatives on May 6, 1983, calculational assumptions and results were presented which, although not yet finalized (design review and approval), substantiated that the thirty-day 15 mR/hr average and 5 rem integrated whole body dose guidelines would not be affected. The presence of the filter adds a maximum dose rate of approximately 3 mR/hr; the additional increment in thirty-day integrated whole body exposure was conservatively estimated to be 1.5 to 2 Rads, increasing the total integrated whole-body dose to less than 3 Rem.

The inspector had no further questions regarding specification of vital area locations and doses.

e. Vital Area Accessibility - Procedure Review

The inspector reviewed the below-listed Shoreham emergency procedures that would be implemented by the licensee in the event of certain accident situations (reflected by plant parameters indicative of such).

The review included 1) a plant walkdown of certain procedures to determine the ability to perform the procedural steps and the accessibility of manual valves or breakers that may require local operation, and 2) an assessment of potential exposures to plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included the following:

SP Number	Revision	Effective Date	Title
29.001.01	1	5/14/82	Acts of Nature
29.002.03	1	5/14/82	Abnormal Radiation Release-Station Ventilation
29.010.01	3	5/14/82	Emergency Shutdown
29.013.02	2	9/7/82	Loss of Secondary Containment Integrity
29.015.01	3	9/7/82	Loss of Off-Site Power
29.015.02	2	9/24/82	Loss of All AC Power
29.017.01	3	9/24/82	Loss of Reactor Building Closed Loop Cooling Water
29.019.01	3	11/5/82	Loss of Service Water
29.022.01	3	11/26/82	Shutdown From Outside Control Room
29.023.01	2	3/30/83	Level Control
29.023.03	5	3/31/83	Containment Control
29.023.04	2	3/29/83	Level Restoration
29.023.05	2	4/14/83	Rapid RPV Depressurization
29.023.09	2	4/14/83	Reactor Pressure Vessel Flooding
29.024.01	1	9/7/82	Transient With Failure to Scram

Based on this review, the inspector determined that 1) the procedures could be performed from the vital areas identified by the licensee's shielding design review, 2) the procedures contained appropriate provisions to assure controlled access to vital areas for post-accident operations, and 3) post-accident doses to plant personnel would be within the guidelines of NUREG-0737.

The inspector had no further questions regarding the accessibility of vital areas associated with these procedures.

3. Exit Interview

The inspector met with licensee representatives on May 6, 1983 (denoted in Paragraph 1) to discuss the scope and findings of this inspection as detailed in this report.

Battelle

Report

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