U. S. NUCLEAR REGULATORY COMMISSION
Region I

Report No. 50-322/83-09

Docket No. 50-322

License No. CPPR-95 Priority -- Category B

Licensee: Long Island Lighting Company

175 East Old Country Road

Hicksville, New York 11801

Facility Name: Shoreham Nuclear Power Station

Inspection At: Shoreham, New York

Inspection Conducted: April 12-15, 1983

Inspectors: J. T. Wiggins, Project Engineer — 4/27/83 date signed

date signed

Approved By: R. M. Gallo, Chief, Reactor Project Section, 1A — 5/5/83 date signed

Inspection Summary:
Inspection on April 12-15, 1983 (Inspection Report No. 50-322/83-09)

Areas Inspected: Routine, announced inspection by a region-based engineer (31 inspection hours) covering NUREG-0737 items; followup on previous inspection findings; review of licensee controls over portable radio transmissions in critical plant areas; and, results of a plant tour.

Results: No violations were identified.

DETAILS

1. Persons Contacted

H. Carter, Assistant Operations Engineer
J. Notaro, Chief Operating Engineer
R. Reen, Security Supervisor
*J. Rivello, Plant Manager
K. Rottkamp, Training Supervisor
*J. Smith, Manager, Special Projects
*W. Steiger, Operations Manager
*R. Wittschen, LILCO Licensing
*J. Wyne, Lead Engineer, Compliance Staff
*E. Youngling, Startup Manager

*denotes those present at the exit meeting on April 15, 1983.

In addition to the above, the inspector interviewed various other personnel in the operations, startup, security and administrative organizations.

2. Followup on Previous Inspection Findings

(Open) Follow Item 83-02-01: Procedural Controls Over Locked Valves. The inspector reviewed SP 21.007.01, Revision 3, "Control of Operations Section Locks and Keys" and determined that this procedure did not adequately address administrative controls over those valves whose locking mechanisms were keylock switches on control room panels. This item is further discussed in Paragraph 4 of this report.

3. NUREG-0737 Items

The inspector reviewed the licensee's implementation of NUREG-0737 Items I.A.3.1, I.C.1, I.C.4, I.C.8, II.E.4.2, II.F.2, II.K.3.16, and II.K.3.25. In addition to the NUREG, references for this review included the Final Safety Analysis Report (FSAR), Supplemental Safety Evaluation Reports (SSER) 1 through 3, licensee correspondence to the NRC and those procedures identified in the Attachment to this report. Further, in those instances where design changes were required for implementation of the item requirements, the applicable Engineering and Design Change Reports (E&DCR) were reviewed and field-verified.

The items are grouped below into two areas: closed items and items remaining open. For the latter group, those actions necessary for closure are identified. No violations were identified.

3.1 Closed Items

3.1.1 I.C.1 Guidance for the Evaluation and Development of Procedures for Transients and Accidents and I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOLs

(Note: Because Items I.C.1 and I.C.8 have definite interfaces, they are discussed together in this report.)

For Item I.C.1, applicants and licensees were required to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures consistent with the guidelines prepared and conduct operator training. I.C.8 required NTOL applicants, such as Shoreham, to submit selected emergency procedures to NRC for review.

LILCO participated as a member of the BWR Owners' Group (BWROG) in the preparation of generic emergency procedure guidelines. The guidelines were developed to comply with NUREG-0660, Item I.C.1(3) as clarified by NUREG-0737 and included reanalysis of small-break locas and inadequate core cooling. As stated in SSER-3, NRC has judged these guidelines as being acceptable for trial implementation at Shoreham.

Pursuant to Item I.C.8, LILCO forwarded the following emergency operating procedures (EOPs) for review by NRC:

SP 29.023.01 thru 05;
SP 29.023.09, and
SP 29.024.01.

The NRC review activities included a visit to the Limerick simulator in October 1981, to evaluate the operators use of these revised EOPs in response to various transient and accident scenarios. As a result, in SSER-3, NRR found LILCO's actions regarding I.C.8 acceptable provided LILCO would provide readable, plant-specific graphs where appropriate in the EOPs.

The inspector checked the latest revision of the above identified EOPs and assured each had been formally implemented. Further, for each of the above procedures which contained graphs, the inspector verified each graph was readable and was specific to the Shoreham plant.

This item is closed.

3.1.2 I.C.4 Control Room Access

In Item I.C.4, licensees and applicants were required to make provisions for limiting access to the control room to those individuals responsible to the direct operation of the plant, to establish the authority and responsibility of the person in charge of the control room to limit access and to establish a clear line of authority and responsibility in the control room including the establishment of a line of succession. Shoreham's program to address I.C.4 is documented in FSAR Section I.C.4. NRC, in Section 13.5 of SSER-1, indicated this program was acceptable.

The inspector verified that SP 21.004.01 had formally implemented the licensee's commitments in this regard. Further, the inspector discussed this program with the Chief Operating Engineer and the Training Supervisor, finding its planned implementation acceptable.

This item is closed.

3.1.3 II.F.2 Instrumentation for Detection of Inadequate Core Cooling

Item II.F.2 required licensees and applicants to provide a description of the instrumentation used or proposed to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). Additionally, in NUREG-0737, the staff clarified the design requirements for this instrumentation system.

In FSAR Section II.F.2, and in letters to the NRC on September 16, 1982 (SNRC-770), October 28, 1982 (SNRC-781) and November 4, 1982 (SNRC-788), LILCO stated its position relative to ICC instrumentation. The licensee indicated the normally installed water level monitoring system (WLMS) was adequate for service in monitoring ICC. LILCO based this position on a generic BWROG report (SLI 8211, July 1982) and a plant-specific report (SLI 8221, September 1982). Further, LILCO stated that these reports demonstrated that no additional instrumentation, such as incore thermocouples, was warranted. In SSER-1, NRC documented the results of its evaluation of the licensee's position. The evaluation found that the WLMS was adequate to predict the approach to ICC and to support the emergency procedure guidelines for ICC. Further, as indicated in Section 3.1.1 of this inspection report, NRC found the emergency procedure guidelines along with the resultant emergency operating procedures acceptable. However, the

staff required the licensee to provide incore thermocouples in addition to the WLMS as a condition for closeout of this item.

During the ASLB proceeding for Shoreham, Suffolk County (SC) and the Shoreham Opponent's Coalition (SOC) filed contentions which bore upon the acceptability of the ICC instrumentation system. Accordingly, in an Agreement filed December 9, 1982, representatives of LILCO, NRC, SC and SOC agreed to two license conditions to effect closure of the ICC issue. One proposed condition covered hardware modifications to improve the performance of the WLMS. The second condition would require the addition of other ICC instrumentation which may result from the NRC staff's review of a BWROG report on this subject.

During this inspection, the inspector reviewed the monitored ranges of the WLMS. These ranges include: hot-calibrated narrow and wide range instrumentation covering from about 7" above the top of active fuel to high in the vessel; cold-calibrated fuel zone channels covering from below the core to high in the vessel; and a cold-calibrated shutdown channel used to detect a full vessel. Additionally, the inspector verified the installation of selected Rosemount 1153 DB-type transmitters and compared power supply information documented in plant drawings with that presented in Table 3.3 of the Shoreham plant-specific WLMS study (SLI 8221) for transmitters N037A, N037B, and N081D. Finally, the inspector verified the installation of a four-channel Rosemount analog trip system for WLMS trip signals to the reactor protection system and the emergency core cooling actuation systems. No discrepant conditions were noted.

This item is closed.

3.1.4 <u>II.K.3.16 Reduction of Challenges and Failures of Relief Valves-Feasibility Study and System Modifications</u>

This item required licensees and applicants to review methods for reduction to challenges to relief valves and to implement modifications necessary to reduce these challenges and to prevent failures. To address this issue, LILCO participated in a BWROG on this subject. The report of this group is contained in Section II.K.3.16 of the FSAR. Briefly, the BWROG evaluated about 23 design and operations alternatives finding about 10 of which being suitable for implementation.

LILCO has implemented three design features and plans to implement one additional feature after fuel load. Currently, LILCO has implemented: (1) a procedural requirement in SP 29.023.01 to open one safety relief valve (SRV) and reduce pressure to 800-960 psig if SRVs are cycling; (2) the installation of the two-stage target rock design; and (3) design changes to the nitrogen supply to the SRV operators to prevent overpressurizing these operators (ref. IEB 80-25 and GE SIL 196 Supplement 8). LILCO plans to change the MSIV isolation setpoint from L2 (low level) to L1 (low-low level) after fuel load.

NRC documented its acceptance of the licensee's approach in SSER-1.

For this inspection, the implementation of SP 29.023.01 was checked; a field verification of the installation of relief valves between each SRV accumulator and its respective valve operator (E&DCR P-3697) was performed; and a review of plant drawings was conducted to assure that relief valves in the nitrogen purge system (E&DCR P-3697A) were provided. No problems were noted.

This item is closed.

3.1.5 <u>II.K.3.25 Effect of Loss of Alternating-Current Power on Pump Seals</u>

The licensee was required to determine the effects of a loss of ac power to the cooling supply to the recirculation pump (RP) seals and to demonstrate these seals could withstand a 2-hour loss of ac power.

At Shoreham, cooling water to the RP seals is normally supplied by the control rod drive hydraulic system (CRDHS) and the reactor building closed loop cooling water (RBCLCW) system. Following a loss of offsite power, emergency diesel generators start and supply power to three 4160V buses. From these buses, safety-related 480V buses receive power. Some of the 480V loads remaining after this type of event are the RBCLCW pumps. Additionally, the safety-related service water pumps continue to operate. Therefore, in these events, cooling water supply to the RP seals remains. In SSER-1, NRC documented its acceptance of the licensee's approach.

The inspector reviewed plant drawings to verify that the RBCLCW and the reactor building SW pumps are powered from Class 1E, safety-related busses which receive backup power from emergency diesel generators. Further, the inspector

field-verified that the RBCLCW pump breakers were installed in the proper motor control centers.

This item is closed.

3.2 Open Items

3.2.1 I.A.3.1 Revise Scope and Criteria for Licensing Examinations

This item required the licensee to: (1) revise the licensed operator training program to ensure license candidates would authorize NRC to forward the results of licensing examinations to licensee management; (2) to revise the licensed operator requalification program to include instruction in heat transfer, thermodynamics, fluid flow and mitigation of accidents with a degraded core; (3) to revise the evaluation criteria for requalification annual examinations; and (4) to revise the list of control manipulations required of operators during the requalification program. LILCO responded to this item in FSAR Section I.A.3.1 indicating commitments to implement each of the above requirements. NRC, in SSER-1, documented acceptances of the licensee's actions.

The inspector attempted to verify the implementation of each licensee commitment in this area. Implementation was acceptable with the following exceptions:

(1) the licensee had not implemented the commitment requiring license candidates to authorize NRC to forward examination results to licensee management so that these results could be factored into the requalification training program. SP 12.014.06 does not incorporate this commitment.

(2) SP 12.014.07 did not specify periodic training in mitigating accidents with a degraded core, although the 1983 requalification training schedule indicated this training would occur.

(3) SP 12.014.07 did not require that personnel who failed the annual requalification examination be removed from licensed activities pending satisfactory completion of an accelerated requalification program.

The Training Supervisor indicated the above changes would be implemented into SP 12.014.06 and .07. Further, he indicated that a change to FSAR Section 13.2 was in progress to also address item (2) above.

This item remains open pending completion of the procedure revisions described in (1), (2), and (3) above.

3.2.2 II.E.4.2 Containment Isolation Dependability

This item consists of 7 requirements for review and action by the licensee. These requirements are:

(1) Diversity in the parameters chosen for automatic containment isolation (CI).

(2) Modification of the CI design as a result of classifying systems as either essential or nonessential.

(3) Automatic isolation of all nonessential systems.

(4) CI reset not causing repositioning of CI valves.

(5) Reduction of the CI pressure setpoint.

(6) Sealed closure of containment purge valves that do not conform to Branch Technical Position CSB 6-4.

(7) Containment purge and vent valves auto-isolation on high containment radiation.

In FSAR II.E.4.2, and in letters dated May 17, 1981 (SNRC-598), January 11, 1982 (SNRC-657) and August 31, 1982 (SNRC-762), the licensee documents proposed actions to address each of the seven requirements.

In SSER-1, NRC stated the licensee's actions were acceptable with two exceptions: (1) the containment purge lines did not have the auto-isolation feature on high containment radiation; and, purge valve operability had not been demonstrated. Subsequently, NRC Region I Inspection Report 50-322/82-23 and SSER-3 acceptably closed out the operability issue. As far as the first exception, the Staff determined that operations without this isolation signal could be permitted provided: (1) a control room operator shall be dedicated to the purge and vent system when it was in operation for online purging to ensure prompt manual closure of the applicable valves in the event of high containment radiation indications; (2) purge and vent operations shall be limited to

safety-related cases; and (3) the licensee shall commit to not use the purge and vent system after December 31, 1983 without specific NRC approval, if the high radiation auto-isolation feature for the purge and vent system is not operable.

The inspector reviewed the administrative controls implemented to address the above requirements for operations of the purge and vent systems. In addition, the inspector reviewed the administrative controls to maintain the 18" purge valves in a sealed closed condition and reviewed those controls on penetrations X-44, X-45, X-46, X-47, XS-7, XS-8, XS-20, and XS-21. The following problems were noted:

(1) Operation of purge and vent system valves is limited to safety-related cases by draft technical specification (TS) 3.6.1.8 and by notes in draft revisions to SP 23.418.01 and SP 23.425.01. However, only the 6" purge valves (IT 46*AOV078 A/B, 079 A/B) are discussed. The 4" nitrogen supply valves (IT 24*AOV004 A/B, 001 A/B) were not included, contrary to licensee commitments in letters SNRC-657 and SNRC-762.

(2) Draft revision to SP 23.418.01 discussed the need for the dedicated operator to provide the closure function to the purge and vent valves. However, as above, the 4" valves were excluded.

(3) The licensee has not committed, in writing, to obtain specific NRC approval prior to operation of the purge and vent systems after December 31, 1983 if the high radiation isolation is not operable. However, based on licensee phone records and on conversations with the NRR Licensing Project Manager, the licensee and the Staff have agreed to resolve this item via a license condition.

(4) In TS 3.6.1.8/4.6.1.8, the licensee has committed to seal closed the 18" purge valves (IT 46*AOV038 A-D and 039 A-D) and inspect them at 31 day intervals. However SP 23.405.01, Revision 1, lists these valves as closed instead of locked closed.

(5) Valve IT 48*MOV 032B was not listed in SP 23.402.01, Revision 1, with a required position of locked closed.

In addition to the above (5) findings, the Senior Resident Inspector has identified a problem with the TIP system isolation ball valves in that they may reopen following CI reset. The Senior Resident is following this matter in connection with the IE Bulletin on Engineered Safety Feature Reset (IEB 80-16).

Further inspection by NRC is required prior to closeout of this item. In addition, the inspector indicated at the April 15 exit meeting that a re-review of this item by the licensee's staff was required to assure all aspects of this item have been implemented.

This item is open.

4. Control of Locked Valves

As part of the review of NUREG-0737 Item II.E.4.2, the inspector reviewed the administrative controls implemented by the licensee to seal closed the containment isolation valves in the containment purge and vent and containment inerting systems, and to seal closed those valves isolating penetrations X-44, X-45, X-46, XX-7, XS-8, XS-20 and XS-21. These controls have been implemented through SP 21.007.01, Revision 3 which addresses sealing of three types of valves: containment isolation valves (CIVs), safety-related valves (S-RVs) and other valves.

Although Step 3.2 of the SP indicated that the SP-controls would cover any type of sealing mechanism, the inspector determined that the balance of the procedure was written in the context of using padlocks. With padlocks as a basis, the procedure directed that the locks for CIVs and S-RVs be uniquely color coded and further directed that the locks for the CIVs have unique cores (i.e. keys to CIV locks do not also operate other equipment). This system has not been effectively applied to all CIVs. As an example, the locking mechanisms for the 18" containment purge valves and for the 4" and 6" purge and inerting valves are key-lock switches on two control room panels. None of these switches have been color-coded. Further, the lock cores for the 6" valves are not unique; the key for these switches also operates the Standby Liquid Control System switches and the switches for the reactor building - turbine building service water isolation valves. Additionally, the inspector determined that precaution 4.1, dealing with the key to the mode switch, was unclear and further, that those keys on the equipment operator's (EO) special key ring have not either been designated or limited to identify which keys the EO is permitted to possess.

Inspection Item 83-02-21 will remain open pending NRC's review of the licensee's resolution of the matters discussed above.

At the exit meeting on April 15, the inspector stated, as an observation, that the method for listing those valves required to be locked closed was cumbersome in that many individual procedures are involved instead of

using a consolidated listing. This method is susceptible to error and may result in some valves not being checked in the locked position at the periodicities committed to by the licensee in Technical Specifications and in other commitments. The licensee representatives present at the meeting acknowledged this observation.

No violations were identified.

5. Plant Tour

In the course of reviewing the field-implementation of design changes that supported various NUREG-0737 items, the inspector noted construction work in progress in the reactor building including the drywell. On April 14, the inspector found a welder working on support C11-191, outboard of the west control rod drive hydraulic control unit bank near the floor level of reactor building elevation 78 without a fire watch present. The inspector determined that a properly-charged fire extinguisher was in the immediate area and that the area was free of readily-combustible material. However, also in the area was a cutout in the floor leading to exposed areas in lower reactor building elevations; the cutout was not covered prior to welding.

The inspector questioned the welder involved and discussed the matter with a member of plant staff who was accompanying the inspector. The inspector also observed the subsequent actions of the plant staff member and other LILCO management. The inspector determined the approach taken by LILCO management was aggressive and complete.

On April 15, a representative of the construction organization onsite met with the inspector. He informed the inspector that LILCO construction practices permitted a welder to be his own fire watch in instances where welding was being performed near floor levels. Also, he told the inspector that the operating Fire Protection Plan had not yet been implemented. However, he indicated that the floor cutout should have had a fire proof cover applied over it. He also stated that the LILCO Safety organization would review this matter.

At the exit meeting on April 15, the inspector stated, as an observation, that LILCO should consider phasing in operations-related programs such as Fire Protection, before the plant reaches fuel load and low power testing. The Plant Manager acknowledged this observation and stated that LILCO management has taken some actions in this area.

No violations were identified. However, NRC will continue to monitor the licensee to assure that LILCO safely and effectively makes the transition from the construction phase to the operation phase of onsite activities.

6. Controls Over Portable Radio Transmissions

Portable radio transmitters can have a detrimental effect on the performance of instrumentation and control equipment as a result of radio-frequency interference. To address this concern, the licensee has prohibited transmissions in certain plant areas. Procedure SP 23.319.01, Revision 3, "Plant Communications" identifies five applicable plant areas including the Control Room, the Relay Room, the Radwaste Control Room, diesel generator hallways and the vicinity of the emergency switchgear room on reactor building elevation 25'.

The inspector reviewed general employee training (GET), station security procedures, and security officer training (SOT) to determine if this prohibition was being discussed with site personnel. The inspection found that both the GET and SOT programs cover this matter; however neither program captures all the affected areas identified in the SP. Also, the inspector was informed that the security communications procedure has not yet been prepared.

The inspector discussed this finding at the April 15 exit, stating the acceptability of the licensee's controls in the area would be unresolved until a consistent list of areas in which transmissions are prohibited is developed and incorporated into the appropriate training programs and procedures (50-322/83-09-01).

7. Exit Meeting

An exit meeting was held on April 15, 1983, to discuss the findings of this inspection. Attendees at this meeting are denoted in Paragraph 1.

ATTACHMENT

Procedures Reviewed

SP 12.014.06		Licensed Operator Training
SP 12.014.07	Revision 0	Licensed Operator Requalification Program
SP 21.004.01	Revision 5	Main Control Room - Conduct of Personnel
SP 21.007.01	Revision 3	Control of Operations Section Locks and Keys
SP 23.319.01	Revision 3	Plant Communications
SP 23.402.01	Revision 1	Primary Containment Post-LOCA Hydrogen Recombination
SP 23.405.01	Revision	Reactor Building Standby Ventilation System
SP 23.418.01	Revision	Reactor Building
SP 23.425.01	Revision A	Primary Containment Inerting System
SP 29.023.01	Revision 2	Level Control Emergency Procedure
SP 29.023.02	Revision 2	Cooldown Emergency Procedure
SP 29.023.03	Revision 5	Containment Control Emergency Procedure
SP 29.023.04	Revision 2	Level Restoration Emergency Procedure
SP 29.023.05	Revision 1	Rapid RPV Depressurization Emergency Procedure
SP 29.023.09	Revision 1	Reactor Pressure Vessel Flooding Emergency Procedure
SP 29.024.01	Revision 1	Transient with Failure to Scram Emergency Procedure