U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-322/85-30

Docket No. 50-322

License No. NPF-36

Licensee: Long Island Lighting Company
P. O. Box 618
Shoreham Nuclear Power Station
Wading River, New York 11792

Inspection At: Shoreham, New York

Inspection Conducted:

Inspectors: E L Conner for 9/18/85
J. A. Berry, Senior Resident Inspector — date

E L Conner 9/18/85
E. L. Conner, Project Engineer — date
Section 1B

Roy L Fuhrmeister 9/18/85
R. L. Fuhrmeister, Reactor Engineer — date
Section 1B

Approved by: Jack Strosnider 9/20/85
J. R. Strosnider, Chief — date
Reactors Projects Section 1B, DRP

Inspection Summary:
During this inspection period, August 2, 1984 - August 31, 1984, the inspectors observed plant operations conducted under the Power Ascension Test Program. These included reactor startup, reactor shutdown, and system testing. In addition, the inspectors reviewed licensee action regarding an inadvertent power spike which occurred during testing, a problem with the operation of HPCI valves at high pressure, and the environmental qualification of electrical/instrumentation panels in the Reactor Building.

The inspectors also reviewed the status of previous opened items and closed 4 of these items. Two new items were opened as a result of this inspection. No violations were identified.

During this inspection period, the Senior Resident Inspector accompanied a representative of the NRC Office of Congressional Affairs on a tour of the Shoreham facility conducted by the licensee.

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DETAILS

1.0 Persons Contacted

H. Carter, Operating Engineer (L)
J. Leonard, Vice President, Nuclear (L)
J. Scalice, Operations Division Manager (C)
D. Terry, Maintenance Division Manager (C)
J. Notaro, QA Division Manager (L)
B. McCafferty, Assistant to the VP-Nuclear (L)
R. Rheen, Security Supervisor (L)
G. Rhoodes, Compliance (I)
R. Kubinak, Director, QA Safety; Compliance (L)

L - Long Island Lighting Company
I - Impell Corporation

The inspectors also held discussions with other licensee and contractor personnel during the course of the inspection.

2.0 Plant Status

Shoreham continued operations and testing under the 5% license issued on July 3, 1985. The completion of testing using the 'B' rod sequence occurred on August 23, 1985, and the plant was shutdown for a preplanned outage period for maintenance and modifications. On August 30, 1985 the plant was restarted using the 'A' rod sequence for the continuation of Power Ascension Test Program testing. The plant was manually scrammed at approximately 3:40 AM on August 31, 1985 due to a loss of instrument air which was caused by the incorrect addition of air drying desiccant to an air dryer tower. The plant remained shutdown at the end of the inspection period.

3.0 Status of Previous Inspection Items

3.1 (Closed) NUREG-0737 Item I.G.1: Special Low Power Testing and Training.

The status of this item was addressed in Supplement 7 to the Shoreham Safety Evaluation Report (SSER-7) where the staff confirmed, "that the BWR owners group recommendations will constitute compliance with Item I.G.1". LILCO's letter SNRC-1014, dated February 16, 1984, provided an argument to demonstrate the adverse impact the station blackout test will have on plant equipment and excepting the augmented BWR owners group (BWROG) program as presented in the BWROG (Walters) to NRC (Eisenhut) letter of February 4, 1981. In the BWROG letter, the following additional training and testing were recommended to be added to the test program; the scope was specified.

- Startup of the RCIC System after loss of AC power to the system.

- Operation of the RCIC System with a sustained loss of AC power to the system.

- RCIC operation to prove DC separation.

- Integrated Reactor Pressure Vessel Level Functional Test.

- Integrated Containment Pressure Instrumentation Test (Test to be performed in conjunction with Containment Integrated Leak Rate Testing).

The first three test requirements were folded into TP 23.119.02 (Special I.G.1 RCIC System Test) and performed in March of 1984. In November 1982, PT 136.001-2 (Nuclear Boiler Process Instrumentation) and PT 654.003-1 (ILRT) were performed meeting the last two requirements, respectively.

In reviewing the RCIC system test, the inspector requested a copy of the completed and signed test procedure. The licensee has been unable to locate this documentation in their files. The inspector confirmed that the test was performed by reviewing training records. The completed test results for the other special test (PT 136.001-2 and PT 654-003-1) were located in the central records. The inspector confirmed that the record copy of nine (9) Preoperational Test Procedures and five (5) Station Procedures were properly stored in central files. It is concluded that the missing signed copy of RCIC System Test (TP 23. 119.02) is an isolated case. The licensee plans to rerun this test to generate records of test completion.

3.2 <u>(Closed) Outstanding Item 50-322/84-29-04 Emergency Preparedness</u>

The reference inspection report created this item based on the Emergency Planning Licensing Board memorandum and order which left the following two items open:

- Whether the "final" version of the brochure lists the radio stations that are participating in LILCO's emergency broadcast system; and,

- Whether LILCO has completed installation of "path finder signs" at "every major road".

The inspector's review found the present versions of the emergency brochure contains the listing of the radio stations that would participate in the emergency broadcast system. The license sees no reason why this listing would be removed in subsequent versions of the brochure. Revision 5 of the Emergency Plan makes no mention of pathfinder signs at every major road. The licensee finds this requirement impractical. This outstanding item is administratively closed pending resolution of emergency planning issues at Shoreham.

3.3 (Closed) Outstanding Item 50-322/84-02-07: Reduced Power Output of EDGS

In February 1984, while Emergency Diesel Generator (EDG) 103 was being run at high power levels for testing, the licensee observed that to reach the overload rating (3900 KW) the fuel racks had to be full open. In report 50-322/84-37, the inspector acknowledged LILCO's request to down grade the KW capabilities of the TDI diesel generators to 3300 KW (from 3500 KW) and to change the FSAR to reflect this. This inspection was to verify corrective actions regarding the EDG capacity including Licensing Board ruling, emergency loads evaluation, and License/Technical Specification (TS) changes.

The June 14, 1985 Partial Initial Decision (PID) addressed the issue of EDG "qualified load" reduction from 3500 KW to 3300 KW and found "reasonable assurance that for the first fuel cycle the TDI EDGs can perform their required safety function, if necessary, at a qualified load level up to 3300 KW, and that operation at such a level will not lead to failure of the crankshaft." They also stated that "the current programs of the staff and LILCO will result in acceptable procedures and training exercises that will minimize the likelihood of operator errors that could result in EDG overload."

FSAR Revision 34, dated November 1984, provides the revised EDG emergency loads evaluation based on measured/actual loads (for certain equipment) or nameplate loads. The new Table 8.3.1-1A, entitled Emergency Diesel Generator System - Maximum Emergency Service Loads, shows the total loads under loss of offsite power (LOOP) and loss of coolant accident (LOCA) - 3253, 3209, and 3226, for TDI EDGs 1, 2, and 3, respectively. The Surveillance Requirements (SR) for TSs 4.8.1.1.2 (EDG Operability) require operational testing with the unit loaded to 3300 ± 100 KW.

The inspector reviewed SP 29.015.01, loss of Off-Site Power, containing subsequent action step 4.3," If 3300 KW is or will be exceeded on any EDG by the addition of other loads, reduce the load to less than 3300 KW by removal of loads in the order of priority shown in Table 1." To address the question of training, the inspector reviewed the requalification training lesson plan on "TDI - EDG 3300 KW load" and the training records and found the training given acceptable. In addition, an on-duty operator was questioned regarding EDG loading; he was knowledgeable about the procedure to be used. Finally, the inspector noted the three annunciators that have been installed to warn the control room staff when 3300 KW loading is occurring on any EDG.

The procedure and training issue was addressed in a detailed response to NRR questions of February 5, 1985, by LILCO letter dated April 4, 1985. The inspector briefly reviewed this submittal; NRR is making a detailed review of the April 4, 1985 submittal. From an inspection standpoint, this issue is closed.

3.4 (Closed) Resolution of NUREG-0737 Item II, K.3.18: Modification of Automatic Depressurization System (ADS)

The original ADS actuation logic at Shoreham contained a high drywell pressure permissive to activate the 105 second timer. In response to NUREG-0737 Item II.K.3.18, the BWR Owners Group Study - Option 2 was to remove the high drywell pressure permissive and install a manual switch that may be used to inhibit ADS actuation if necessary. The Shoreham Safety Evaluation Report, Supplement 7 (SSER-7) reported the physical modifications completed and satisfactorily tested prior to May 1984. SSER-7 found the modifications acceptable for resolutions of Item II.K.3.18 and states. "an NRC regional inspector will verify that instruction regarding the use of the inhibit switch has been addressed in plant emergency procedures." This inspection was to perform that function.

The reason for installing the inhibit switch is to provide a convenient way for the operator to prevent ADS when it would be adverse to safety; primarily, following a transient with failure to scram. The emergency procedure for this transient, SP 29.024.01 was reviewed by the inspector. Immediate Operator Action Step 3.6.1.2 states, "Place the ADS inhibit switch to 'INHIBIT' and terminate all injection into the RPV with the exception of CRD and RCIC or HPCI to maintain RPV water level above the top of active fuel (TAF)." In addition, the inspector reviewed the following calibration and functional test procedures.

SP 44.201.01 - ADS-Times Calibration and functional and actuation functional Test,

SP 44.621.01 - ADS-Water Level Calibration and functional Test, and

SP 44.654.01 - Drywell Pressure Calibration and functional Test.

The inspector confirmed that the inhibit switch is addressed and the references to the high drywell pressure permissive in regards to ADS have all been removed with the exception of one. Prerequisite 5.6 stating, "Drywell pressure below 1.65 psig" remains. In discussions with the licensee, the inspector acknowledged that this prerequisite had no real adverse effect on the procedure. This closes II.K.3.18 since the SSER-7 open issues are resolved.

4.0 Inadvertant Power Spike Greater Than 5% Power

At approximately 8:40 PM on August 15, 1985 an unplanned power excursion of about two percent power occurred for a period of 3-5 seconds during testing. The event occurred while the plant was increasing power in preparation for operational testing of the High Pressure Coolant Injection System (HPCI). An analysis concluded that this power excursion caused core thermal power to peak at between 5.8 and 6.3 percent power.

The licensee reduced power to 1.5 percent and began an investigation into the cause of the power excursion. The inspector was notified on August 16, 1985 by the Operations Division Manager.

An investigation into the cause of this event determined that the feedback linkage on one of the two startup level control valves had become disconnected, thereby allowing excessive feed water flow to enter the reactor vessel for a short period of time. This extra feedwater, which is at a lower temperature than the water in the vessel, had the effect of adding extra reactivity to the core, thereby causing the power spike.

The purpose of the feedback linkage is to signal back to the valve controller what the valve position is, so that the valve does not "overshoot" when the controller attempts to reposition it. When this feedback linkage disconnected, the valve controller did not have the signal telling it the valve had reached its desired position, so the valve fully opened. Once fully open, a separate full open signal was sent to the controller which caused it to then fully close the valve. During this full open to full close stroke of the valve, the excess feed water entered the vessel.

Upon determination of the cause of the event, the licensee began a design analysis to change the method by which the feedback linkage arm is mechanically connected to the valve controller so this event would not occur in the future. Additionally, the licensee inspected all other valves with this type of linkage to see if a similar problem could occur with other valves. The results of the licensee's actions in this matter will be the result of future inspections, and this item is designated as Open Item 85-30-1 pending the outcome of these future inspections.

The Shoreham Operating License (NPF-36) states that the licensee shall report any violations of the requirements contained in Section 2.C of the license, which limits the maximum power level to 5%, in the following manner: initial notification shall be made within twenty four (24) hours to the NRC Operations Center via the Emergency Notification System with written follow-up within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e). The NRC Senior Resident Inspector at Shoreham was notified approximately twelve hours after the event. However, the licensee, initially, did not believe that a notification to the NRC Operations Center was required and strong encouragement by the NRC Senior Resident Inspector and Licensing Project Manager was required before such notification was made. The notification to the NRC Operation Center was made within the required 24 hours.

The licensee's basis for their initial belief that notification was not required was a letter sent on July 24, 1985 to H. Denton, Director of the Office of Nuclear Reactor Regulation from J. Leonard, Jr., Vice President - Nuclear Operations, LILCO (REF: SNRC-1194, Limits of Operation of Shoreham Nuclear Power Station Unit 1 under Condition of Facility Operating License NPF-36). In this letter, the licensee stated that it's interpretation of the limitations of the 5% power license issued July 3, 1985 was that "the

plant can be operated up to 5% rated thermal power on a steady state basis. LILCO assumes that this interpretation includes recognition of the potential for momentary power spikes above the 5% steady - state limit which could be the result of a transient caused by equipment malfunction." The licensee further stated that it's understanding was "... that power spikes from unplanned transients do not constitute a violation of our operating license...". Based on this stated interpretation, the licensee felt that no notification was required.

Although Region I commends the licensee for their foresight in bringing this matter up for discussion with the NRC early in the Power Ascension Test Program, the licensee's position that their interpretation was valid, prior to a formal response from the Office of Nuclear Reactor Regulation, was premature. Until the licensee receives formal notification that the NRC concurs in it's interpretation, in this matter or any other which may affect potential violations of license conditions, it would be prudent to consult with NRC representatives to ensure that all parties concur.

5.0 HPCI Valve Operators

During the high pressure (960 psi) testing of the HPCI system, problems with operation of MOV-36 (4 inch minimum flow bypass) and MOV-37 (10 inch CST test return throttle) valves occurred. Both valve operators failed to stroke under dynamic conditions and required torque switch setting increases. In addition, MOV-36 required stem cleaning, 3 rings of packing, and adjustments to make it operable.

The inspector reviewed the maintenance records for these two valves and found they had been tested at low line pressure a number of times. The data recorded was valve operating time (seconds) and motor running current (amperes) for open and close operation, motor torquing in current (amperes), torque switch settings, and voltage at the motor. Questions regarding the appropriate Limitorque Operator torque settings required to assure valve operability at the highest delta-pressure (normally accident conditions) across the valves were discussed with the licensee. Although the inspector's concerns remain, this issue will not become an open inspection item because it is the subject of a comprehensive IE Bulletin soon to be issued.

6.0 Electrical/Instrumentation Panels

While performing a general inspection of equipment in the reactor building, the inspector noted an open safety-related instrumentation panel; one of the two clips that hold the cover closed was loose. Checking similar panels in the reactor building resulted in a list of eleven (11) electrical/instrumentation panels where one or more clips were loose or missing or a condition existed where environmental qualification was questionable. The inspector provided the information to the watch engineer on duty and later discussed the findings with LILCO management. In confirming the inspector's findings, the licensee found four (4) more panels with similar problems.

The licensee's corrective measures have been as follows:

- Perform an extensive inspection of safety system panels in the reactor building. In this inspection an additional 70 problems were found with the clips and other devices necessary to keep panel doors closed. Of this number, 66 were repaired at the time of discovery and the other four repaired a few days later. In addition, the original 15 (11 NRC plus 4 LILCO) have been corrected.

- LILCO plans an inspection of the junction boxes in the Drywell during the upcoming "source outage".

- The Quality Control Division will perform an independent walkdown to assure that all panels are securred and will add an inspection item to their surveillance reports to assure that electrical panels are correctly restored after maintenance is performed.

- Maintenance, Instrumentation and Control, and Maintenance Service Division personnel were cautioned about the need to properly restore electrical panels after work.

The licensee performed an analysis of the equipment that could be effected by loss of environmental qualifications of the original 15 panels. Included were valve position indications, operating controls, pressure and flow alarms, initiation logic, leak detection, and level and pressure indications for reactor containment isolation, primary containment fans, main steam, cooling water pumps, recirculation pumps, RCIC leak detection, RHR, core spray, reactor vessel instrumentation, and RCIC pumps. No argument was presented to show that equipment inside any panel would meet EQ requirements with the cover loose. Shoreham License Conditions No. 8 states that, "Prior to November 30, 1985, the licensee shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49." Therefore, this issue is not a violation at this time. It will be left an open inspection item (50-322/85-30-2) to be reviewed at a later time.

Note: During a subsequent inspection on September 4, 1985, the inspector checked a few panels on the 63 foot level in the Reactor Building. He found loose clips on panels 1JB*574A (MSIV Leakage-Containment) and E32 N 053F (Steam Line Leakage-Flow).

7.0 Licensee Event Reports

7.1 LER 85-006, 85-017, 85-020, ESF Actuations on False RPV Low Water Level Signals.

These apparently related LER's were reviewed as a group. The inspector reviewed five surveillance procedures selected at random for the caution notes, revised steps for returning transmitters to service, and lesson plan for special training of I&C technicians who work on instruments affecting reactor pressure vessel level reference legs. Surveyed instrument racks 1H21*PNL 005 and 004 were inspected to assure that special

caution tags had been placed on the instrument valves. No discrepancies were noted.

7.2 LER 85-010, Inadvertent RHR Loop 'B' Trip in Shutdown Cooling Mode.

The inspector reviewed RHR Operating procedures for changes and discussion of shutdown cooling suction pressure, reactor vessel pressure, and reactor vessel water level relationships. No deficiencies were noted.

7.3 LER 85-015 Ultimate Heat Sink, Accumulation of Sediment.

The inspector discussed the soundings and dredging with licensee personnel and reviewed post-dredging soundings and calculation sheets for average depths. No deficiencies were identified.

8.0 Action Item Tracking

In the process of validating the Region I Open Item Report for Shoreham, the inspector reviewed the licensee's action item tracking system. This system, under SP No. 12.006.03 (Master Punch List Procedure), tracks LILCO's status on NRC bulletins, circulars, information notices, and inspection open items in addition to numerous administrative and plant process system items. The computerized list may be added to by any section, division or department, is prioritized by the originator, reviewed by management, modified under careful control, and closed only by the planning and scheduling section after a completion form is processed.

The inspector found the Master Punch List system to be user friendly, well controlled, and very comprehensive. With the exception of a few minor errors, no problems were identified.

9.0 Nuclear Review Board (NRB)

The inspector observed the conduct of the August 1985 Nuclear Review Board Meeting conducted the mornings August 28 and 29, 1985.

The inspector reviewed the membership of the Nuclear Review Board against Technical Specification requirements. The inspector also reviewed the NRB agenda and observed NRB discussions of agenda items. The inspector found the NRB members to be knowledgeable and well prepared for discussion of agenda items. The inspector noted the NRB meetings to be open, candid and well run.

No unacceptable conditions were identified.

10.0 Site Tours

Site Tours

The inspectors conducted periodic tours of accessible areas in the plant, in the Colt Diesel Generator Building, and around the site. During these tours the following specific items were evaluated.

- Fire Equipment - Operability and evidence of periodic inspection of fire suppression equipment;

- Housekeeping - Maintenance of required cleanliness levels;

- Equipment Preservation - Maintenance of special precautionary measures for installed equipment, as applicable;

- QA/QC Surveillance - Pertinent maintenance activities were being surveilled on a sampling basis by qualified QA/QC personnel; and

- Component Tagging - Implementation of appropriate equipment tagging for safety, equipment protection, and jurisdiction.

Except as noted in Section 6.0, all items observed on these tours were satisfactory.

In addition the Senior Resident Inspector accompanied a representative of the NRC Office of Congressional Affairs on a site familiarization tour on August 17, 1985. Areas of the plant toured were the refueling floor, 15 ft. elevation of the Reactor Building, Rad Waste, Diesel Generator rooms, the new Colt Diesel Building, and the Control Room.

11.0 Unresolved Items

Areas for which more information is required to determine acceptability are considered unresolved. Unresolved items are contained in paragraphs 4.0 and 6.0.

12.0 Management Meetings

At periodic intervals during the course of this inspection, meetings were held with licensee management to discuss the scope and findings of this inspection. With the exception of the list of improperly closed panels (secton 6), written material was not provided to the licensee by the inspectors.

Based on NRC Region I review of this report and discussions held with licensee representatives it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

The inspectors also attended entrance and exit interviews for inspections conducted by region-based inspectors during the period.