U.S. NUCLEAR REGULATORY COMMISSION

REGION I

REPORT NO. 50-322/84-45

DOCKET NO. 50-322

LICENSE NO. CPPR-95 / NPF-19

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: November 1 - December 14, 1984

INSPECTOR: P W Eselgroth 12/19/84
P. W. Eselgroth, Senior Resident Inspector Date Signed

APPROVED BY: Jack Strosnider 1/3/85
J. Strosnider, Reactor Projects Sect. 1C Date Signed

SUMMARY: The inspector reviewed and closed seven previous inspection items and opened one new unresolved item pertaining to control of plant evolutions. This latter item stemmed from problems encountered during the lowering of the reactor vessel water level. Plant system description controls, emergency equipment lockers, reactor building flooding, plant modification administrative controls, equipment failure history analysis, emergency diesel testing, service water system corrosion, fuel support piece alignment and Bahnson Co. HVAC units were also reviewed. No violations were identified.

This report involved 70 hours of inspection by the resident inspector.

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DETAILS

1.0 Persons Contacted

W. Burnett, Compliance Engineer (I)
C. Cole, Colt Diesel Building Construction Manager (L)
R. Gutmann, Maintenance Engineer (L)
J. Kelly, Field QA Manager (L)
G. Montgomery, Nuclear Engineer (L)
A. Muller, QC Division Manager (L)
J. Notaro, Modification/Outage Division Manager (L)
J. Leonard, Vice President - Nuclear (L)
R. Purcell, Startup Manager (L)
R. Rheen, Security Supervisor (L)
G Rhoades, Compliance Engineer (I)
J. Scalice, Operating Division Manager (L)
J. Smith, Manager Nuclear Operations Support Division (L)
D. Steiger, Plant Manager (L)
D. Terry, Maintenance Division Manager (L)
J. Wynne, Lead Compliance Engineer (L)

I - Impell, Inc.
L - Long Island Lighting Company

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

2.0 Status of Previous Inspection Items

2.1 (Closed) Unresolved Item (82-14-06): Plant System Descriptions.

This item concerned the use of Plant System Descriptions which do not reflect the current plant configuration. These descriptions serve as one of several reference sources for plant technical support staff and control room personnel. The inspector's concern was that control room personnel might inadvertantly consider the plant descriptive information in these documents to reflect the as-built condition of the plant. Accordingly, the licensee has marked the control room copies of the Plant System Descriptions "For Information Only".

The inspector had no further questions; this item is closed.

2.2 (Closed) Unresolved Item (84-23-01): Emergency Equipment Lockers.

This item concerned the potential for damage of safety-related equipment during a seismic event due to missile impact by portable emergency equipment lockers. During routine tours of the plant, the inspector noted that the licensee had placed emergency equipment lockers in various areas of the plant, including one on each level of the reactor building.

These lockers are made of steel, contain fire fighting equipment, weigh several hundred pounds, and are mounted on wheels. They have no permanent location and in some cases are located near safety related equipment. The inspector requested that the licensee determine if these lockers could damage safety related equipment during a seismic event. The licensee evaluated this concern and determined that providing a permanent foundationing for these lockers was the proper course of action. The inspector observed these installations which utilize substantial angle plates, approximately one-half inch in thickness, to anchor these lockers.

The inspector had no further questions; this item is closed.

2.3 (Closed) Unresolved Item 84-29-02: Reactor Building Internal Flooding.

Review of the Shoreham Probabilistic Risk Assessment by NRC Consultants (Brookhaven National Laboratory) verified that adequate provisions existed to protect essential equipment in the Reactor Building from pipe break flooding or minor leakage after a LOCA. This is documented in Appendix A to the Shoreham Safety Evaluation Report, Supplement No. 7. However, this report did identify that some potential deficiencies existed in the Shoreham Alarm Response Procedures (ARP's) for mitigating a flood. These ARP's contained general guidelines for monitoring system parameters for determining leakage location and initiating the leakage isolation, but did not contain specific requirements for operators to systematically check the operating parameters of ECC and RCIC systems, and did not contain a checklist with specific steps which should be followed during a flood in the reactor building.

Subsequently, the licensee revised ARP 5670 and 5671, Reactor Building Flood Level High, to include a detailed checklist of system parameters to monitor for determining leakage location and initiating the leakage isolation. In addition, the licensee revised twenty-five other ARP's to require the operator to take the immediate and subsequent actions specified in the ARP 5670 and 5671 checklists. These twenty-five ARP's include such monitored alarms as: Reactor Building Closed Cooling Water (RBCLCW) System Header Pressure Low; RCIC Flow Low; Condensate Storage Tank Level Low; HPCI Pump Suction Pressure Low; and Reactor Building Service Water Header A Pressure Low. All of these alarms could indicate a leak which might cause flooding of the Reactor Building.

The inspector reviewed these procedures and verified that these latest revisions appear to address the concerns identified by BNL during their review of the PRA.

The inspector had no further questions; this item is closed.

2.4 (Closed) Unresolved Item 84-32-02: Plant Modification Administrative Control.

During the September 1984 inspection period the administrative controls implemented by the licensee for plant modifications were reviewed. The purpose of this inspection was to determine whether the plant modification administrative controls contained in the station operating manual provide clear definition of the administrative steps necessary for plant modifications including all required approvals. The following Station Procedures were reviewed:

SP12.010.01 - Interim Station Modification Program
SP12.010.02 - Station Modification Activities
SP12.013.01 - Maintenance Work Requests

This review focused, in particular, on those steps required for returning a modified system to service. The inspector found, as previously documented, that the above procedures do not clearly define what approval signatures the Watch Engineer should check for prior to returning a modified system to service. Plant management agreed with the NRC inspector's observation and informed the inspector that steps were being initiated to correct the lack of procedural clarity in this area.

The licensee revised the Station Modification Procedure (SP12.010.02) to correct the procedural inadequacies effective December 4, 1984. This revision was reviewed and approved by the Review of Operations Committee and the Plant Manager. Also, Station Procedure Change Notice No. 84-1713 dated December 14, 1984 modified SP12.013.01 (Maintenance Work Requests). This change, which was approved by the Review of Operations Committee and the Plant Manager, clarifies for the Watch Engineer what approval signatures are necessary prior to returning modified systems to service for differing repair circumstances.

The inspector had no further questions; this item is closed.

2.5 (Closed) Unresolved Item 84-32-03: Equipment Failure History Analysis.

As stated in Inspection Report 84-32, a review was conducted by the inspector in September 1984 of the equipment history area to ascertain how the licensee tracks equipment failures. The purpose of this inspection was to determine to what extent the licensee maintains an equipment failure trend analysis program for assessing and highlighting significant failure rate trends related to such causes as poor design and/or materials or poor maintenance practices.

The inspector found that no system or method, manual or computerized, was in use at that time for performing equipment failure rate trend analysis for Shoreham plant equipment. Several large file cabinets of Maintenance Work Requests (MWR) are on file by component number; however, this information was not being analyzed for significant failure rate trends.

During the inspection period for report 84-39, the inspector was provided with a copy of a memorandum by the Maintenance Division Manager entitled "Implementation of Equipment History Program, Including Plant Trending Data and Analysis". This program established implementation dates for various milestones of an equipment history trending and analysis system and, in particular, identified a December 1, 1984 implementation date for the trend analysis capability for Safety-Related equipment.

During the week of December 10, 1984 the inspector reviewed this area again. The inspector found that the licensee has established a computerized data base of all previous Maintenance Work Requests, filed by component, for all Safety-Related equipment. The licensee has also established a documented practice whereby the maintenance history of each such piece of equipment requiring repair is retrieved from the computer and reviewed for applicability to the planned repair actions by Maintenance Coordinators. The inspector reviewed the capability of the system by requesting the equipment history for a randomly selected piece of Safety-Related equipment (core spray pump).

The inspector had no further questions; this item is considered closed.

2.6 (Closed) Unresolved Item 84-39-01: Emergency Diesel Generator Lube Oil Piping Failure.

As discussed in the previous resident inspector report, a leak was observed in the 1¼ inch lube oil supply line to the turbocharger of EDG-103 during full load testing of the engine. The leak was observed at a circumferential crack one-inch in length adjacent to a weld in the area where this 1¼ inch line joins the main lube oil supply header. The piping crack was observed to be leaking on the order of a tablespoon a minute and necessitated shut-down of the engine to avoid development of increased leakage and consequent turbocharger drainage and fire hazard problems.

The licensee determined from an examination of the lube oil piping installation that either an improperly installed pipe hanger or pipe run installation stresses had overstressed the pipe and caused the failure. This overstressed conditioned was evidenced by piping deflection that occurred when a flanged joint was unbolted during examination of EDG-103. This pipe hanger had been installed by the licensee for vibration reduction purposes along with several others. The licensee concluded following the EDG-103 piping failure that the proper course of action for all three EDG's was to remove this particular hanger since it did not appear to be necessary and might be causing a stress problem. A deterministic analysis of vibrational stress was not feasible in this location. In the process of returning the EDG-103 lube oil line to service to complete the crank-shaft fatigue endurance testing, the "stab-in" weld connection of the 1¼ inch line to the supply header was replaced with a "socket" weld connection.

This modification was not considered necessary for the other two engines since the cause of failure was the pipe hanger. The EDG-101 and 102 turbocharger lube oil piping installations have been checked to ensure that they do not have locked-in (cold sprung) stresses. The piping will also be non-destructively examined.

The inspector had no further questions; this item is closed.

2.7 (Closed) Construction Deficiency Report 84-00-02: Service Water System Strainer Corrosion.

The previous inspection report discussed a problem the licensee had experienced with leaks in the Service Water System Pump Strainers due to salt water corrosion. Two of the four strainer shells were removed for examination and repair. The leaks developed due to corrosion in areas where the internal epoxy protective coating had failed which allowed salt water to contact the carbon steel strainer casing. At the end of the previous report period the review of this problem was still in progress. In the last report, the inspector also discussed a previous service water system corrosion problem. Corrosion of the P-41 Service Water System Pump internals had been previously identified as a problem (CDR-82-00-07) caused by galvanic corrosion due to dissimilar metals.

In light of these problems, the licensee was requested to consider the generic problem of the corrosive sea water environment and determine if any other components in the Service Water System are subject to accelerated corrosion caused by failure of epoxy coatings, dissimilar metals, or other mechanisms.

Licensee response to strainer problem: The corrosion problem was identified by leakage through the shell of 001A and 001D strainers. Strainer 001D is currently operating with a temporary repair. Two of the strainers, 001A and 001C were disassembled and inspected. The inspection revealed interaction between the strainer assembly and the protective coating on the carbon steel shell, which caused the protective coating to wear down and expose the base material. Once exposed, selective corrosion was initiated. Disassembly of both strainers, 001A and 001C, revealed partial failure of the epoxy lining at and above the seal weld of the monel tube sheet support ring to the strainer body. For the 001A strainer a hole entirely through the shell was apparent as well as a number of locally corroded areas. For the 001C strainer, the corrosion did not penetrate completely through the shell. Ultrasonic thickness measurements were made of the shell thickness above,behind, and below the ring. The corrosion of the tube sheet is considered responsible for the failure since enlarged clearances between the holes in the sheet and hold down bolts were observed.

After a review of measured data, drawings and stress calculations, the licensee determined that the shell thickness was 0.370 inches and the minimum thickness required was 0.236 inches. The licensee then determined that weld repairs would be carried out in any area where the shell thickness was less than 0.320 inches, which provides for approximately 0.1 inch of corrosion allowance.

The 001A and 001C strainers have now been weld repaired under the ASME Section XI, Article IWD-4000 program and hydrostatic pressure tested using the manufacturer's standard.

The weld repaired areas including the repaired flange face have been recoated with the epoxy material under the supervision of representatives from the material manufacturer. The coating was inspected for thickness and soundness with thickness and spark testing devices. New tube sheets of the same type material (Noehanite Iron alloyed with Nickel) are being installed. Replacement tube sheets of other materials such as Monel are being evaluated by LILCO.

To ensure that corrective actions are taken to reduce the possibliity for future failures, a plan is being developed to examine the strainers after 6 months operating time and the results of this inspection will be utilized to develop a long-term inspection program. Upon the reinstallation of strainers 001A and 001C,strainers 001B and 001D will be removed from service and repaired as necessary.

Licensee reponse to generic corrosion problem: The licensee has recently performed a review of the material for the piping and associated components in the Service Water System (P41) down to 1" pipe size. Based on this review, the licensee has reconfirmed that the P41 system materials had been suitably selected for their intended saltwater service. Wetted internal surfaces are either manufactured of materials generally recognized as suitable for this service, such as monel, inconel, aluminum-bronze, etc., or protected from the corrosive media by a lining material (such as rubber or epoxy).

The inspector had no further questions; this item is closed.

2.8 (Open) Unresolved Item 84-29-01: Bahnson Co. HVAC Units.

Inspection Report 84-29 discussed a February 9, 1984 Board Notification (84-006) relative to possible quality control problems with HVAC units manufactured by the Bahnson Co., Winston Salem, North Carolina. Board Notification 84-006 indicated that Bahnson had provided HVAC units for safety-related applications at a number of nuclear reactor facilities including Shoreham and Inspection Report 84-29 requested the licensee to identify these applications. This problem was also the subject of Information Notice 84-30.

There are four Bahnson HVAC units in safety-related applications and the licensee has inspected each of these units to determine if the units conform to specification and drawing requirements. Two of these units are used in the control room air conditioning system and the other two are used in the relay room air conditioning system.

These Bahnson units have been inspected by the licensee and the following deficiency reports have been written on the four units:

LILCO DEFICIENCY REPORT #	SYSTEM/COMPONENT NUMBER(s)	SERVICE
2167	1X61*ACU-007A&B	Control Room
2168	1X41*ACU-014A&B	Relay Room
2470	1X41*ACU-014B	Relay Room
2471	1X41*ACU-014A	Relay Room
2472	1X61*ACU-007A	Control Room
2473	1X61*ACU-007B	Control Room
2537	1X61*ACU-007A	Control Room

These LDR reports covered principally welding and fastener deficiencies. The licensee expects to have all of the above LDR's dispositioned by the end of January 1985. The resident inspector is following this issue and will report on its disposition in future inspection reports.

3.0 Control of Plant Evolutions

On November 2, 1984, preparations were made by the licensee to drain water from the reactor vessel to support the inspection of reactor vessel internals by the licensee's Reactor Engineer. In the process of draining the reactor vessel, several thousand gallons of this water were inadvertently spilled onto the drywell floor. Since the reactor had not been operated to date, this water was uncontaminated.

In order to lower reactor vessel water level, a drain path was lined up to drain the vessel via the Reactor Water Clean Up (RWCU) System to the Hotwell and the vessel was drained without incident until the level in the reactor vessel annulus fell to the recirculation pump suction line, at which point no further water could be removed via this path. Up to this point the lowering of the vessel level was performed in accordance with procedure TP 22.500.01. To further lower the level, a flow path through the reactor vessel bottom head drain to the Drywell Equipment Drain Tank was lined up. This portion of the evolution was done without a procedure. It appears that the combination of the rate of water draining into the Equipment Drain System and the fact that the Drywell Equipment Drain Tank Vent Valve (2G11-02V-3533) was shut, caused the water in the Drywell Equipment Drain System to back up and overflow at system scuppers, and the recirculation pump seals, onto the drywell floor. Water accumulated on the drywell floor and then passed through the drywell floor drains to the Floor Drain System Tank. Following the inspection of the reactor vessel internals, the licensee refilled the vessel with water. However, on

November 5, 1984 it was necessary to again lower the reactor vessel water level for an inspection and the flooding occurrence described above was repeated.

The licensee found during its investigation of these occurrences that an unapproved valve lineup sheet, showing vent valve 1G11-02V-3533 as shut, was in the official valve lineup file. This was contrary to the open position shown for this valve in Station Procedure 23.702.02. Also the Drywell Equipment Drain Tank Level recorder in the control room was found to be out of service.

The inspector's review of this occurrence to date has resulted in the following concerns relative to the control of plant evolutions and the timely response to abnormal occurrences:

1. In the event of abnormal occurrences, what steps will be taken to preclude a repetition of the problem prior to completion of the review/corrective action process.

2. Some steps appear warranted to ensure that the control associated with performing evolutions in a safe and orderly manner are actually implemented such as maintaining accurate approved plant status files, using procedures and checking the availability of system instrumentation including backups.

This is unresolved item 84-45-01.

4.0 Fuel Support Piece Misalignment

During this inspection period the licensee determined, from slower than normal control rod movement times associated with rods 1015 and 1407, that two fuel support pieces were misaligned relative to the support piece tabs and adjacent core plate guide pin positioning. Specifically, the support piece tabs were observed by the licensee to be located to one side of the pin, rather than on both sides of the guide pin for both rod locations. The licensee subsequently removed the fuel support pieces and associated control rods for inspection. The licensee was provided an inspection criteria for the control rods by the Nuclear Steam Supply System vendor, General Electric, which called for verification that the boron absorber rods are free to move horizontally and vertically for thermal expansion; that there are no raised edges on the rod blade surfaces; that adequate clearance (0.11 inches) exists between the rod blade sheaths and the boron absorber rods; and that the control rod blade rollers roll freely. The control rods were inspected to this criteria and found to be satisfactory. The fuel support pieces were also visually examined and found to be in satisfactory condition. The control rods and fuel support pieces were then reinstalled in the reactor vessel. The licensee then conducted an inspection of the alignment and location of all but three of the other fuel support pieces in the reactor vessel and found them to be installed as required. The remaining three fuel support pieces must be examined during fuel load operations.

The inspector had no further questions.

5.0 Emergency Diesel Generator Testing and Inspection

The fatigue cycle (740 hour) testing of the TransAmerica DeLaval, Inc. replacement design crankshaft in EDG-103, at a power level of 3300 ± 100 KW, was completed on November 1, 1984 at 8:30p.m. The crankshaft had accumulated approximately 747 hours of testing at this load by the time the engine was shutdown the morning of November 2, 1984.

Following the engine shutdown from testing on November 2, 1984, disassembly of EDG-103 for inspection was begun. By November 6th, the engine gear inspection had been completed with no significant findings. This included the crankshaft gear, idler, pump and governor drive gears. Other areas inspected by this date were the cam lobes, cam gallery and wrist bushings. These areas were also free of significant indications.

During a licensee visual examination prior to a liquid penetrant inspection of cylinder head (S/N H-60), an indication of weld repair was noted in the fuel injector port area of the cylinder head fire deck. As per disposition of LILCO Deficiency Report (LDR) 2541 pertaining to cylinder head (H-60), the indication of weld repair noted on cylinder head (H-60) was identified as unacceptable. This is consistent with the disposition of spare cylinder head (G-70) documented in LDR-2510. This finding by the licensee pertains to a manufacturing process, not a conditioned caused by engine testing. The licensee stated that this will result in head (H-60) being replaced, based on an agreement between Suffolk County, the NRC and the licensee. Inspection of the crankshaft following the 740 hour, 10^7 cycle, fatigue endurance test revealed several apparent indications by liquid penetrant testing in the web to pin regions. However, these indications were determined by Eddy Current inspection to be non-relevant.

Other parts of the disassembled EDG-103 engine that were inspected at this time include the engine block;piston heads, skirts and liners; connecting rods and wrist pins, bearings and the cylinder heads. The inspections conducted by the licensee were overviewed by NRC TDI Diesel Task Force consultants and monitored periodically by the NRC Resident Inspectors. At the conclusion of the EDG-103 engine part inspection there were no significant findings other than that pertaining to the H-60 cylinder head observed to have been factory weld repaired.

The inspector had no additional questions.

6.0 Reactor Fueling Activities

The reactor neutron sources were received on site by the licensee on December 9, 1984 in support of planned fuel load activities. On December 12, 1984 the ten neutron sources were removed from their shipping cask in the fuel storage pool. Placement of the sources in their holders was completed on December 14, 1984. The resident and region-based inspectors overviewed segments of these activities in the refueling area for proper adherence to procedures, radiological controls and security requirements. The inspectors found these areas to be performed satisfactorily. The licensee estimates that fuel loading into the reactor will occur on or about December 19, 1984.

7.0 Colt Diesel Generator Building

As of December 12, 1984, construction of the Colt Diesel Generator Building is proceeding on schedule and is approximately 89% complete. All electrical systems necessary to support engine runs have been released for use and all mechanical systems necessary to support engine runs, with the exception of combustion air and exhaust, have also been released for use. Test runs of the three Colt Diesel Generators are scheduled to begin early in January 1985.

8.0 Site Tours

The resident inspector conducted periodic tours of accessible areas in the plant, in the new Colt Diesel Generator Building and around the site in general. 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 construction activities were being surveilled on a sampling basis by qualified QA/QC personnel;

- Security - Adequate construction security;

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

During a routine tour of the refueling floor area in the reactor building, the inspector performed a visual check of the synthetic fiber slings stored in that area. The slings were inspected for abnormal wear, broken or cut fibers and variations in strand size. The inspector observed one synthetic fiber sling with a section of strands where the transverse fibers holding the strands together were broken. The inspector pointed this out to the Reactor Engineer who was also in the refueling area at the time. The Reactor Engineer had the sling removed from the refueling floor.

All other items observed during general site/plant tours were found to be satisfactory.

9.0 Unresolved Items

Areas for which more information is required to determine acceptability are considered unresolved. Unresolved items are contained in paragraphs 2.8 and 3.0.

10.0 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. Based on the NRC Region I review of this report and discussions held with licensee representatives on December 19, 1984, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.

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

11.0 NRR Operational Readiness Review

On November 15 and 16, 1984, the Director of the NRC Office of Nuclear Reactor Regulation and members of his staff conducted an operational readiness review at Shoreham. On the afternoon of November 15, the review team conducted inspection tours of the plant which included the reactor building and drywell area, radwaste and turbine buildings and general site areas. Although the TDI diesels are not required for fuel load and low power operation, a tour was made of the TDI emergency diesel generator rooms and of the disassembled EDG-103 parts laydown area. Also, a tour of the Colt emergency diesel generator building construction area was made. On the morning of November 16, licensee management made an operation readiness presentation to the NRR staff, the Region I Administrator and the Shoreham NRC Resident Inspectors which covered the following areas:

- Management Organization/Experience
- Plant Description/Background
- Special Plant Features
- Construction Completion
- Shift Advisors
- Shift Operations and Training
- TMI Items
- Emergency Electric Power Supplies
- Schedule for Fuel Load

At the conclusion of the presentation, the Director, NRR, and Regional Administrator commented on the organizational and material readiness improvements evidenced during the most recent Systematic Assessment of Licensee Performance (SALP) period and on the high level of readiness observed during plant tours on this visit.

12.0 Upper Management Involvement At Plant

The Vice President, Nuclear Operations, held several meetings during this report period in order to discuss the importance of high quality performances from all plant personnel in the months ahead. The Vice President illustrated, by examples from past experiences, how dependent the plant team performance is on each individual's performance. He pointed out how crucial the upcoming months are to the long term success of the Shoreham plant. The Vice President also stated that he intends for the fuel load, initial criticality and follow-on activities to be conducted in an unhurried and methodical manner so as to minimize errors. He called upon each individual to know their assigned tasks, follow procedures (without turning their heads off) and to not hesitate to call problems to the attention of their supervisors. This important safety message was clearly presented and positively received by plant employees.