U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-322/84-29

Docket No. 50-322

License No. CPPR-95

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: July 2 - August 19, 1984

Inspectors: C Petrone 8/17/84
C. Petrone, Resident Inspector — Date Signed

P. Eselgroth 8/20/84
P. Eselgroth, Sr. Resident Inspector — Date Signed

Approved By: Edward G. Greenman for 8/27/84
J. Strosnider, Reactor Projects Sect. 1C — Date Signed

Summary: The inspector reviewed and closed out three previous inspection findings and one TMI Action Plan Item. Five new unresolved items and one Inspector Follow Item were identified. Final Preoperational testing of EDG-103 and Integrated Electrical Testing using all three Delaval Emergency Diesel Generators were witnessed. No violations were identified; the licensee's actions were found acceptable

DETAILS

1.0 Persons Contacted

R. Gutman, Maintenance Engineer (L)
J. Kelly, Field QA Manager (L)
A. Muller, OQA Engineer (L)
J. Notaro, Chief Modification/Outage Engineer (L)
J. Leonard, Vice President - Nuclear (L)
R. Purcell, Startup Manager (L)
J. Riley, Operational Manager (GE)
J. Smith, Manager Nuclear Operations Support Division (L)
W. Steiger, Plant Manager (L)
D. Terry, Chief Maintenance Engineer (L)
J. Wynne, Lead Compliance Engineer (L)
E. Youngling, Nuclear Engineering Manager (L)

GE - General Electric
L - Long Island Lighting Company
S&W - Stone and Webster

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

2.0 Previous Inspection Item Update

2.1 (closed) Violation 83-02-14: Inadequate QA Inspection of Turbocharger Supports. During a previous inspection the inspector noted that the final documented OQA inspection had failed to identify the following nonconforming conditions in the completed turbocharger support installation:

(1) A tubular support weld varied from its design drawing detail because it failed to wrap around the entire fit-up joint, as required, and the welding technique specified for the rework was not qualified for the as-built angular orientation (acute angle less than 30°); and,

(2) The high strength bolting installations did not conform to the applicable AISC Code requirements in that several bolted joints had missing washers and that ASTM A-490 bolts in these installations had been retorqued (reused).

The licensee performed an engineering evaluation of the welding on the tubular support and determined that the as-built configuration was acceptable. Engineering and Design Coordination (E&DCR) F-37646J, was revised on February 16, 1983 to indicate that the weld was not required in the area where the support formed the acute angle; therefore the weld procedure was not to be used in this area. The licensee performed a review of similar welds on the remaining two TDI Diesels and determined that these welds had been performed correctly. In both cases the angle was greater than 30° and both joints had been welded all around.

The licensee performed retraining of inspection personnel to emphasize the importance of accurate verification. The inspector examined the welds and confirmed that the final as-built condition had been evaluated and approved by engineering and that it conformed to the final as-built drawings.

In response to concerns of inadequate A-490 high strength installations, the licensee replaced the suspect bolts with new bolts and established a policy of always using new bolts whenever a high strength installation is made. To determine the number of other A-490 installations, the licensee performed a survey of 250 pipe support installations. Three additional A-490 installations were identified; these were inspected and verified satisfactory. Subsequently, these bolts were removed during the disassembly of the Emergency Diesel Generators for crankshaft replacement, and new bolts were installed for each reassembly. During the present inspection, the inspector reviewed the Repair/Rework Requests R43-1903, R43-1904, and R43-1109 for the most recent installation of these supports and verified that new A-490 bolts were specified and that a Quality Control inspector had witnessed the final torquing of these bolts. The inspector examined these bolts and verified that they appeared to be tight (at least hand tight) and the required washers were present. The inspector also selected a sample of two other Repair/Rework Requests (R48-521 and R43-647) and verified that all bolts appeared to be tight, all required washers were present, and the final configuration of all welds was correct. No discrepancies were identified.

2.2 (Closed) 85-21-02, Station Blackout Procedures. Station Blackout refers to the complete loss of all AC electrical power to the plant. This is considered very unlikely and beyond the plant's design basis, due to the number and diversity of AC power sources available. Nevertheless, due to the significant consequences of a station blackout, the licensee was required to establish procedures and training for this event. During a previous inspection, a confirmatory review of these procedures and training was performed. The inspector identified specific shortcomings which were subsequently addressed by the licensee. During the present inspection, the inspector verified that the following corrective actions had been taken:

- A lesson plan to cover training for a Station Blackout was issued as part of the Requalification Plan;
- Station Procedure SP29-015.01, "Loss of Offsite Power" was revised to include Scram, Turbine Trip, and NSSS Isolation in the Automatic Actions section of the procedure; and,
- SP29.015.02, "Loss of All AC Power" was revised extensively to correct editorial errors and incorporate the recommended procedure improvements and clarifications.

Based on this review, the inspector determined that the concerns addressed in this unresolved item had been adequately addressed. The inspector had no further questions.

2.3 (Closed) Unresolved Item 84-18-02. This item identified a problem of significant numbers of personnel in the Office Building Annex (and Technical Support Center) ignoring fire alarms. A history of spurious fire alarms in this area had contributed to the problem, and no attempt was being made to announce to all personnel in the area whether or not the alarm was spurious. The licensee has taken action to inform all personnel of the importance of the importance of treating such alarms as real, unless or until an announcement is made to the contrary. The licensee has taken action to further reduce the number of spurious alarms. Subsequent to these actions, the inspector has observed proper fire alarm response. This resolves the concerns identified in this item.

3.0 Three Mile Island (TMI) Modifications

As a result of the accident at Three Mile Island in 1979, the NRC issued a number of new requirements, detailed in NUREG-0737, "Clarification of TMI Action Plan Requirements". The implementation of one of these items was reviewed on site.

3.1 (closed) TMI Item II.K.3.21 Restart of Core Spray and Low Pressure Coolant Injection Systems on Low Level: This item identified that the Core Spray (CS) and Low Pressure Coolant Injection Systems (LPCI) system flow would not automatically restart (on loss of reactor water level) once the operator had manually stopped the flow. This TMI Action Plan item recommended that the LPCI and CS system logic be modified so that these systems will restart automatically, if required, to assure adequate core cooling.

The BWR Owners Group reviewed the existing system design and determined it was adequate based on the:

- Comprehensive nature of BWR operator training;
- Emphasis on reactor water level control during training;
- Emergency procedure guidelines;
- Relatively long time available for operator action; and,
- Extent to which low reactor water level conditions are displayed and alarmed in the control room.

The owners group concluded that further automation would unnecessarily increase system complexity, reduce system reliability, and restrict operator flexibility. The licensee endorsed the BWR Owners Group position in a letter from J. P. Navarro (LILCO) to H. Denton (NRC) dated April 15, 1981. Subsequently, NRR issued Supplemental Safety Evaluation Report No. 1 in which they agreed that no modification to provide for automatic restart was necessary. Based on this review, the inspector determined that the concerns addressed in this TMI item had been adequately addressed and no further action was required.

4.0 Falsification of Quality Control Documents by Bahnson Co.

On February 9, 1984 Board Notification 84-006 was issued by NRR Division of Licensing to the Commission to advise them that an OI investigation has been made of allegations that quality control documents were being falsified and that signatures of QC inspectors were being forged at the Bahnson Co., Winston Salem, North Carolina. The Bahnson Company has provided safety-related HVAC units for a number of nuclear reactor facilities including Shoreham. The HVAC units supplied by Bahnson Co. may provide, among other things, cooling for the reactor containment building during normal and accident conditions.

The staff is currently considering the safety implications of this information and until more specific facts are obtained that will confirm a technical decision regarding the continued acceptability of the defective Bahnson equipment, the staff is allowing the affected plants to operate. In the interim the licensee is requested to identify any HVAC units supplied by Bahnson Co. that are in use at Shoreham. This will be tracked as Unresolved Item 84-29-01 which should be addressed prior to the end of the first refueling outage.

5.0 Reactor Building Internal Flooding

On December 29, 1983, Senior NRR Management reviewed the results of an on-site assessment of internal flood protection at Shoreham. As a result of the December 6, 1983 site visit, all participants agreed that adequate provisions exist to protect essential equipment from pipe break flooding and from minor leakage after a LOCA. However, further evaluation remained to be done to resolve the flooding concern as it related to flooding from maintenance procedure errors.

Brookhaven National Laboratory (BNL), the contractor which is reviewing the Shoreham PRA, was tasked with an advance review of the probabilistic risk assessment of maintenance induced flooding which had been done by LILCO (see LILCO's December 2, 1982 submittal). The BNL evaluation notes that some potential deficiencies exist in the Shoreham alarm response procedures for mitigating a flood. Otherwise, we have determined that the report confirms our previous conclusion that maintenance flooding sequences do not contribute significantly to risk. The BNL report will be published in the next SSER to document the closure of this item. The modifications to the procedures will be listed as a confirmatory item, whose completion will be verified by a Region I inspector prior to exceeding 5-percent power. The Region I inspector will verify that the revised procedures are consistent with the assumptions made in the BNL PRA for flooding alarm response by the operators. This is unresolved item 84-29-02.

6.0 Core Spray Full Flow Test Isolation Signal

Susquehanna Licensee Event Report LER84-026, dated June 14, 1984, described a potentially generic problem with the Core Spray Full Flow Test Isolation Signal. Their Technical Specifications require that Core Spray Full Flow Test Isolation Valves isolate on Reactor Vessel Low Level or Primary Containment High Drywell Pressure. The as-built condition at Susquehanna provides an isolation on Reactor Vessel Low Level, but not on Primary Containment High Drywell Pressure.

At Shoreham, the licensee performed a review and determined that the as-built Core Spray Full Test Isolation Valves did isolate on either the Reactor Vessel Low Level or Primary Containment High Drywell Pressure, which is in agreement with Shoreham Technical Specifications. The inspector reviewed Elementary Drawings for the Core Spray System and the Analog Trip System and noted that both isolation signals were incorporated in the Shoreham design. The inspector also reviewed the completed Preoperational Test Procedure and noted that these isolation signals had been tested and verified operational. The licensee has issued surveillance procedures to perform ECCS Monthly Trip Unit Calibration and Functional Testing which include the appropriate steps to verify proper functioning of both isolation signals.

The inspector identified no discrepancies and had no further questions. This item is designated unresolved item 84-29-03 and is considered opened and closed during this inspection period.

7.0 Emergency Preparedness

The Shoreham EP Licensing Board Memorandum and Order Ruling on LILCO's motion for Summary Disposition of Contentions 11.E, J, K, L and M granted partial summary disposition to LILCO, but left two items open for "post-hearing" confirmation by the staff. These items include confirming:

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

2. Whether LILCO has completed installation of "pathfinder signs" at "every major road."

These items must be complete prior to exceeding 5% power and are collectively designated unresolved item 84-29-04.

8.0 Reactor Building Cable Tray and Conduit Support Allegation (RI-83A-95)

An allegation concerning reactor building cable tray and conduit supports was reviewed by a region based inspector (K. Manoly). To resolve the concerns identified during this review, the following areas must be addressed in more detail:

- The prying effect on angle connectors needs to be addressed in the analysis or justification provided for not considering this effect.

- The percent of allowable bolt-load needs to be determined for the ten cases analyzed by finite element analysis.

- The need to verify that the ten cases analyzed represented the "worst-case" loadings.

This is considered unresolved item 84-29-05.

9.0 Salem ATWS Events

The NRC has issued inspection requirements to followup on the licensee's response to Generic Letter 83-28, Salem ATWS Events. The required inspections are expected to be completed by October 30, 1984. This will be tracked as Inspector Follow Item 84-29-06.

10.0 Emergency Diesel Generator Preoperational Testing

The licensee began a seven day endurance test of Emergency Diesel Generator EDG-103 at 0100 on July 27, 1984. The engine was run at a power level of approximately 2750Kw. On August 2 a water leak developed in the jacket cooling water manifold that cools the diesel exhaust manifold. This leak was estimated to be one-half gallon per hour when it began; and inc eased to approximately 3 gallons per hour by the end of the seven day run on August 3. The total capacity of the jacket water system is approximately one-thousand gallons. The test engineer on shift reported that makeup water was manually added every six hours. Following completion of the test the crack was weld repaired.

On August 6, 1984 at 2047 the licensee began the 24 hour load test of EDG-103. The engine was run at the (proposed) overload rating of 3500Kw for two hours and at or above 3475Kw for the remaining 22 hours. The inspector observed portions of this test and verified that the leak had been repaired satisfactorily. The inspector noted that the leak occurred at a welded joint on the exhaust manifold water jacket supplied with the original engine. The crack did not occur on the new cylinder block. If this leak had occurred during normal operation its presence would have been annunciated by a low jacket water level alarm in the diesel generator room and by a diesel trouble alarm in the main control room. The leak would not have caused an engine shutdown since the gradual loss of jacket water could be corrected by periodic addition of makeup water.

The inspector observed the last two hours of the 24 hour test and noted that all engine parameters were in the normal range. Following completion of the 24 hour run a hot restart was performed to demonstrate the ability of EDG-103 to automatically start on loss of AC power and pickup its required emergency loads. EDG-103 started and stabilized at a load of 3225Kw which included two service water pumps. No discrepancies were noted.

11.0 Integrated Electrical Testing

During the period August 13 - 18, 1984 the licensee performed the Integrated Electrical Test (IET) in accordance with PT.307.002-3, dated July 26, 1984. This IET was performed using all three Delaval Emergency Diesel Generators. The previous IET, performed in May 1984, was run using only EDG-101 and EDG-102. EDG-103 was out of service for replacement of the cracked cylinder block.

The objective of the Integrated Electrical Test was to verify the existence of independence among redundant onsite power sources and their load groups. Various equipment configurations were utilized to demonstrate independence during Plant response to the Loss of Coolant Accident (LOCA) with, and without, a Loss of Offsite Power (LOOSP). Plant equipment was aligned for each test to demonstrate maximum response to each event while attempting to adhere to normal system lineups. Equipment response was verified using control room instrumentation.

Test 1

Simulated loss of coolant accident (LOCA) with offsite power available. This test demonstrated that the diesels started without load shedding and with the automatic initiation of Emergency Core Cooling Systems.

Test 2

Loss of Offsite Power (LOOSP) with simultaneous simulated Loss of Coolant Accident (LOCA). This demonstrated load shedding, diesel automatic start, bus re-energization, and sequencing of the bus loading, initiated by concurrent bus undervoltage and a LOW LOW LOW reactor water level accident signal.

Test 3

Loss of offsite power and simulated LOCA with orange emergency DC battery system and associated 103 emergency AC diesel generator out of service. This demonstrated separation of the orange DC and 103 emergency bus from the red and blue DC buses and the 101 and 102 emergency AC buses.

Test 4

Loss of offsite power and simulated LOCA with blue emergency DC battery system and associated 102 emergency AC diesel generator out of service. This demonstrated separation of the blue DC and 102 emergency bus from the red and orange DC buses and the 101 and 103 emergency AC buses.

Test 5

Loss of offsite power and simulated LOCA with the red emergency DC battery system and associated 101 emergency AC diesel generator out of service. When completed, this test will demonstrate separation of the red DC and 101 emergency bus from the blue and orange DC buses and the 102 and 103 emergency DC buses.

During the inspection, the inspector witnessed a dry run of test 2 and the official record run of tests 2, 3 and 4. The following observations were noted:

- Senior plant management was present and actively participated in the tests;
- Test prerequisites were met;
- A designated test engineer was in charge and directed the performance of the test;

- The minimum test personnel requirements were met;

- Proper plant supporting systems were in service;

- Special test equipment required by the test procedure was calibrated and in service;

- Test personnel actions appeared to be correct and timely during performance of the test;

- QA personnel were present and completed required verification signoffs; and,

- Data was collected for final analysis by qualified personnel.

Findings

Through observations, records, review, and performance evaluation of licensee personnel involved in test runs of the integrated electrical tests, the inspector verified that testing was conducted in accordance with approved procedures which incorporated the licensee's test commitments and regulatory requirements. The inspector verified the preliminary acceptability of test results for the test runs witnessed.

During Test 2, Loss of Offsite Power with Simulated LOCA, all three Delaval diesel generators started, came up to speed, and pickup the load in 7.2 seconds. The acceptance criteria is less than 10 seconds. All three output breakers closed virtually simultaneously and all required loads were picked up. Approximately 10 minutes into the run the total load on each of the diesel generators, with both core spray pumps and all four RHR pumps running at rated load, was:

EDG-101	EDG-102	EDG-103
2700Kw	2650Kw	2800Kw

During tests 3 and 4 similar results were obtained. The appropriate emergency diesel generators came up to speed and picked up their load in the specified time limits. Preliminary review indicated that all required automatic switching and valve cycling had occurred as planned.

The results of Test 5 and a detailed review of these test results will be performed during a future inspection after all the data has been retrieved (from the GETARS transient analysis recorder) and analyzed.

12.0 LILCO Strike

On August 14, 1984 striking members of the IBEW reached a contract agreement with LILCO and returned to work. They were out on strike for thirty-two days during which time plant management took over the duties of the striking union members. The major ongoing activities, completion of EDG-103 testing and construction of the new Colt diesel generator building, were not affected. In fact, there was some reduction in the backlog of maintenance work requests. Routine inspection by the resident inspectors did not identify any problems caused by the absence of maintenance personnel. The strikers return to work was uneventful with both union and management apparently relieved to have the strike resolved. The new contract will be in effect for 18 months.

13.0 Plant Tours

The inspector conducted periodic tours of accessible areas in the plant during normal, backshift, and weekend hours. 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 of systems under or following testing;

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

- QA/QC Surveillance - Pertinent construction and startup activities were being surveyed on a sample basis by qualified QA/QC personnel; and

- Security - Adequate security for site construction and new fuel storage activities.

During tours of the reactor building the inspector noted a steady improvement in general housekeeping. However, elevation 8' continues to be a problem area due to rusting of some equipment including service water pipes and reactor building closed loop cooling water heat exchangers. Some rust is bleeding through recently painted areas. The inspector will continue to monitor the licensee's housekeeping efforts. Unresolved Item 82-04-13, "Monitor Housekeeping Until Fuel Load", will remain open until the corrosion problems in Elevation 8' are corrected.

14.0 Unresolved Items

Areas for which more information is required to determine acceptability are considered unresolved. Unresolved items are contained in paragraphs 4 through 9.

15.0 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.

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