U.S. NUCLEAR REGULATORY COMMISSION

REGION I

REPORT NO. 50-322/84-23

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: May 28, 1984 - July 1, 1984

Inspectors: C Petrone — 7/3/84
C. Petrone, Resident Inspector — Date Signed

P.K. Eselgroth — 7/9/84
P. Eselgroth, Sr. Resident Inspector — Date Signed

C Petrone for — 7/3/84
J. Shedlosky, Sr. Resident Inspector, Millstone — Date Signed

Approved By: E. McCabe — 7/19/84
E. McCabe, Chief, Reactor Projects Sect. 1C — Date Signed

Summary: May 28 - July 1, 1984 (213 hours)

The inspectors reviewed and closed out one Bulletin, five previous inspection findings, and one TMI Action Plan Item. The accidental discharge of a security guard's weapon was investigated: the status of the Colt Diesel Generator project was reviewed, and plant inspection tours were performed. An unresolved item concerning the storage of emergency equipment lockers near safety related equipment was identified. An allegation regarding the installation of BISCO seals at Shoreham was reviewed: the installation was found adequate. No violations were identified: the licensees actions were found acceptable.

## DETAILS

### 1.0 Persons Contacted

R. Gutman, Maintenance Engineer (L)
J. Kammeyer, Assistant Head SEO (S&W)
J. Kelly, Field QA Manager (L)
R. Lawrence, Startup Engineer (S&W)
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) NRC Bulletin 80-06 and Deviation 83-08-04: Engineered Safety Feature (ESF) Reset. This Bulletin was issued because several instances had been reported to the NRC when automatic closure of primary containment isolation valves would not have occurred because the safety actuation signals were either manually overridden or bypassed during normal plant operations. In addition, a related design deficiency was discovered at several operating reactors where, upon resetting the ESF actuation logic, certain safety related equipment was found to return to its non-safety operating mode. It is the NRC's position that resetting of an ESF actuation signal should not cause any equipment to change position.

Although the licensee responded to this Bulletin in FSAR question 223.88 by Rev. 21, License Application Ammendment 39, dated May 19, 1981, during subsequent review Region I discovered several additional reset conditions and recorded the details in Report 50-322/81-06, paragraph 4. These involved Control Room Air Conditioning System Valves, which change state on ESF reset following a manual actuation, and Battery Room and Emergency Diesel Generator Room Ventilation systems, which automatically restart following the reset of the fire suppression system.

The licensee responded to additional requests for information in FSAR questions 223.99 and 223.100. These responses were contained in License Application Ammendment 44, dated May 10, 1982 and 46, dated December 30, 1982 which transmitted FSAR Rev. 26 and 28 respectively. These questions restated the NRC's original request for information stating that the Bulletin was intended to cover both automatic and manual initiations.

During a review, conducted in March 1983, Region I again identified additional components which change position following an ESF reset. These involved Reactor Building Standby Ventilation System (RBSVS) components and the Traversing Incore Probe (TIP) System nitrogen purge valve. Findings were recorded in Report 50-322/83-08, paragraph 8 and a Deviation was issued (322/83-08-04).

The licensee's response, dated May 17, 1983, contained additional data following engineering reviews expanded to include ESF systems actuated by non-ESF actuation signals and affected by the subsequent reset of these signals and non-ESF systems affected by resets of ESF actuation signals. The operating characteristics of components of the Steam Condensing Mode Subsystem of RHR, and of the RBSVS were investigated. Also, the operation of the TIP System nitrogen purge containment isolation and Automatic Depressurization System Safety Relief valves were addressed. The licensee also committed to revise responses to FSAR questions and include preoperational test verification of component change of position after an actuation signal clears and again after a system reset.

The licensee informed the NRC Office of Nuclear Reactor Regulation of this issue by letter dated June 8, 1983. Additional testing was included in the Integrated Electrical Test, and in special additional tests conducted on Core Spray System full flow test valve logic and in RBSVS initiation on high radiation.

License Application Ammendment 48, dated June 30, 1983 and Ammendment 50, dated December 1, 1983 which transmitted FSAR Rev. 30 and 32, respectively, provided update information to FSAR questions 223.88, 223.99 and 223.100.

The Integrated Electrical Test (PT.307.007-Integrated Electrical Test with Diesels 101 and 102 Available, Diesel 103 Not Available) incorporated demonstrations of the ESF reset operation. At that time it was observed that the air operated actuators of the testable Feedwater Check Valves 1B21*AOV-036A and B cycled and automatically returned to its normal operating state when the ESF logic was reset. These feedwater check valves have an actuator which provides spring force to provide a positive closure differential pressure on the seated disk. The actuator provides this force when an isolation signal is present or during remote testing. Isolations associated with the feedwater check valves are high drywell pressure and reactor vessel low water level No. 1 (The Core Spray and LPCI Systems initiation point) and remote manual isolation from the control room. These features do not interfere with the reverse flow closure of the valves, but during normal feedwater flow conditions, an isolation signal to the actuator will cause a slight reduction in feedwater flow to verify mechanical operability of the valve disc. The interaction of the Feedwater Testable check valve actuator with the ESF reset logic has been reviewed by the NSSS. The reset of the Remote Test/Isolation was found to be appropriate and the preferred method of operation.

Based on this review the inspector determined that the concerns addressed in IEB-80-06 and Deviation 83-08-04 havebeen satisfactorily resolved.

2.2 (closed) Unresolved Item 83-02-01: Valves in Reactor Building Elevation 8' and 40'. In order to address concerns regarding the valve packing gland studs/nuts and valve stem corrosion in the 8' and 40' elevation of the reactor building, the licensee issued Repair/Rework Request R/RR-071-001 to inspect all valves in these areas. This inspection identified 47 valves which needed repair, cleaning or lubrication. Corrective actions were performed in accordance with various maintenance work requests.

To verify that corrective action was complete, the inspector performed a walkdown inspection of selected HPCI, RCIC, RHR and Core Spray system valves in the reactor building elevation 8' and 40'. A sample inspection of ten of the forty-seven identified valves found them to be clean and in good repair. The general condition of all valves in these areas has improved significantly during the past year and is now considered acceptable. The inspector will continue to monitor the condition of these valves during routine inspections of the plant.

2.3 (closed) Unresolved Item 84-07-03: Hydrogen Recombiners. IEEE-323-1974EQ testing of the Rockwell Hydrogen Recombiners Thermal Magnetic Circuit Breakers resulted in inadvertent opening at case temperatures exceeding 120$^{o}$F. These breakers are CB-10 and CB-13 in power cabinets IT48*PNL-048A and 048B. To correct this problem the installed breakers were replaced in accordance with Engineering and Design Coordination Reports (E&DCR) F-39958A, F-46339, F-46337A, and Repair/Rework Request T48-149. Post installation inspection by the Operational Quality Assurance Department (OQA), documented in Verification Report T48-189 dated April 25, 1984, revealed that the replacement breakers had been installed satisfactorily. The replacement breakers were retested in accordance with Checkout and Initial Operation procedure C&IO T48-56A, and 47A; and documented on Startup Form 8.7 IT48-12, dated April 19, 1984. No discrepancies were identified by the inspector during this review.

To resolve a concern about a lack of sufficient spare parts on hand for the Hydrogen Recombiners, the licensee issued Purchase Order 363247-1-SSP on March 30, 1984 to obtain additional spare parts to supplement those already on site. Included in the order were thirty different items including spare pressure transmitter circuit cards, gaskets, seals, thermocouples, fuses, seals, flow controllers, power supplies, and relays. This resolves the concern about the lack of spare parts.

Based on the above reviews, this item is resolved.

2.4 (closed) Unresolved Item 83-23-01: Failure of Emergency Diesel Generator Rotor Lifting Rig. This unresolved item reported that a lifting rig component failed during removal of the generator rotor from Emergency Diesel Generator EDG-101. The rotor dropped approximately one-inch onto wooden blocks, but suffered no damage. The lifting rig consisted of two wire rope slings arranged in a two point lift configuration with a spreader bar to hold the lower end of the wire rope slings apart. Each end of the spreader bar was clamped at the lower end of one of the wire ropes by two bolts. During the lift, one of these bolts failed allowing the spreader to slip, and the load to drop. The inspector determined that the wire ropes had been load tested prior to the lift, but the spreader bar had not been load tested since it was not considered a load bearing part.

Subsequent review by the inspector identified that the licensee had redesigned the spreader bar and successfully load tested it in the configuration in which it would be used. The inspector also reviewed the licensee's program for assuring the adequacy of lifting and handling devices used during plant maintenance and overhaul. In response to NUREG-0612, "Control of Heavy Loads at Nuclear Plants" (by letter dated February 21, 1984) the licensee committed to the requirements of:

- ANSI B 30.2-1976, Overhead and Gantry Cranes;
- ANSI B 30.9-1971, Slings: and
- ANSI N 14.6-1978, Standard for Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500Kg) or more.

The lifting devices which the licensee has procured to date were built to specific engineering design requirements and proof tested by the vendor following fabrication. Station maintenance procedure SP 31.023.01, Sling and Hoist Inspection and Identification, is in place and implements the requirements for identification and inspection of lifting devices.

No unacceptable conditions were identified: this item is resolved. The inspector will continue to monitor the adequacy of lifting and handling equipment during routine inspections of maintenance activities.

2.5 (Closed) Unresolved Item 83-08-07: BISCO FLEXIBLE BOOT SEALS This item was opened as part of an allegation received on March 23, 1983, by the NRC (RI-83-A-20) which questioned the adequacy of boot material overlap allowed for during installation of penetration seals at both Susquehanna and Shoreham. Asserted was that "a minimum of 3 inches overlap was required . . . and, in some cases, less than 2 inches was used." Also alleged was a problem with maintaining the proper 150 pcf density of BISCO type SF-150 NH elastomer aggregate material, used as shielding for radiation streaming in certain penetrations - this aspect is addressed in Detail 7 of this report. Neither the boot overlap nor the elastomer density question raised by the allegation has been found to be substantiated as an improper installation which would result in an unqualified seal.

Flexible boot seal overlap was initially investigated as part of Inspection 83-08 at Shoreham, including review of installation procedures and a sampling inspection of approximately 20 penetrations (many in the steam tunnel area) which employ the boot seal. While the allegation suggests that it's the axial seam and not the clamped connection at either the pipe or sleeve end which was in question, all such locations where the boot material would be prepared for application of the Dow 732 silicone adhesive were investigated. BISCO installation procedures have never referred to the word "overlap" when addressing the pipe and sleeve-end seal connections; "overlap" would only be associated with the axial overlapping seam which is prepared by roughening with sandpaper and wiped clean with solvent prior to lamination with adhesive. Those procedures used to specify an overlapping seam of "approximately" 2 inches - the procedures were later revised in May 1983 to be more precise in specifying a minimum seam width. Currently, the minimum overlapping axial seam is specified as a function of pipe diameter (1 inch for piping less than 2 inches in diameter, 2 inches for 2 through 20 inch diameter pipe, and 3 inches for piping larger than 20 inches). There are approximately 30 penetrations at Shoreham which employ the flexible boot seal, and inspection of over 80 percent of these verified that an axial overlapping seam of width of 2 inches or more was maintained.

The pipe and sleeve-end connections, while apparently not in question by the allegation, were also inspected. These locations were addressed by BISCO procedures as follows: prepare (rough and clean) about 3 inches of boot material; apply adhesive for approximately a 2 inch sealant surface; and, install a ½ inch compression ring on that surface to complete the caulked and clamped connection. As with the axial seam, procedures were revised in May 1983 to precisely specify a minimum sealant surface of 1 inch. Sampling inspection of pipe and sleeve-end connections, so made, showed proper installation. The half-inch clamp creates a thin uniform thickness of adhesive, bonding the boot fabric to the pipe or sleeve, necessarily ensuring proper resistance against shear forces resulting from pressurizing the boot seal.

All of the flexible penetration boots used as pressure seals are significantly overqualified, by one to two orders of magnitude. For example, penetration seals for the Reactor Building secondary containment are expected to experience a maximum service condition which is less than 1 psi differential pressure while, depending upon relative pipe and sleeve size, typical seals are qualified for and rated at pressures in excess of 10 psi. As a result of NRC Region I contact of BISCO's project office at Park Ridge, Illinois, in March 1984, a pressure test was developed, performed and documented in BISCO Test Report 748-141 dated April 11, 1984. The test was to determine the effectiveness of a boot constructed with a 2-inch laminated axial seam, a 1-inch pipe sealing surface, and a ½-inch sleeve sealing surface, and installed on a 6-inch pipe penetrating a 16-inch sleeve. This particular configuration is rated for 13.2 psi by vendor calculation, and the test confirmed BISCO's predicted qualification pressure. The test seal was inflated and pressurized in 1 psi increments (each held for five minutes) until, at 14 psi, the boot fabric was cut by the clamp at the sleeve-end and began to leak, finally bursting at 15 psi. The test conservatively used a ½-inch sleeve sealant surface (less than the minimum 1 inch) and a 2-inch axial overlapping seam (the minimum specified width). By comparison, such an installation on a Shoreham Reactor Building penetration would see a maximum predicted differential pressure of 7.3 inches of water or 0.3 psi during a post-accident transient as depicted on FSAR Figure 6.2.3-2. Normal operational vacuum in the Reactor Building will be maintained at minus 1 to 2 inches of water (less than 0.1 psi), and the Standby Ventilation System will be required to maintain building pressure at at least ¼-inch of water lower than outside pressure (or less than 0.01 psi) during the course of a design basis accident.

The as-installed BISCO boots used as pressure seals on penetrations at Shoreham were therefore evaluated as suitable for service.

## 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.B.4: "Training for Mitigating Core Damage" summarizes the NRC requirements for a training program to instruct all operating personnel in the use of installed instrumentation to recognize a degraded core condition and to take action to mitigate core damage using the available plant systems.

The licensee committed to develop a training program in compliance with the "Guidelines for Training to Recognize and Mitigate the Consequences of Core Damage from the Institute of Nuclear Power Operations (INPO) guidelines STG-01, Rev. 1, dated January 15, 1981. The course outline was reviewed and found acceptable by NRR in the Safety Evaluation Report - Supplement #2. (SSER#2). The inspector reviewed the licensees training program, "Degraded Core Training" and noted that it included the subject areas specified in the INPO guidelines: the Safety Evaluation Report, SSER #2; and the FSAR Rev. 27. The inspector also reviewed the attendance sheets and exam results for the course and noted that the required licensed personnel, including managers, had completed the course satisfactorily. No discrepancies were identified.

## 4.0 Security

On June 3, 1984 at 5:48a.m. a security guard accidentally discharged his 9mm semi-automatic pistol into the handgun bullet trap (clearing barrel) when he was checking his weapon prior to his shift. A loaded clip had inadvertently been inserted into the gun in lieu of the empty clip which should have been inserted during the gun check. The bullet was safety contained in the bullet trap. The Security Supervisor notified the Chief

Modification Engineer at 6:00a.m. who investigated the occurrence and notified the Vice President - Nuclear. The Resident Inspector was informed at 8:30a.m. on June 4, 1984. Immediate corrective actions were taken which included retraining of Security Personnel and the posting of the clearing procedure near the bullet trap.

Investigation revealed that the Security Shift Lieutenant who was serving as the "clearing barrel monitor" and the officer failed to follow written security procedures. The security officer and the lieutenant were suspended for three days and were retrained upon their return to duty.

This event was reviewed by a region-based security inspector and a Preliminary Notification of a Safeguards Event PNS-I-84-10 was issued on June 4, 1984.

The licensee's corrective actions were prompt and appeared to be adequate to prevent recurrence. The inspector had no further questions regarding this event. The adequacy of the clearing procedure (gun check) will be reviewed again during the final security inspection prior to implementation of the licensees security plan.

## 5.0 Colt Diesel Generator Project

On June 3, 1984 the inspector was informed that the licensee had decided to send Colt Diesel Generator #902 back to the manufacturer (Fairbanks-Morse) for inspection, after some minor damage was noted during transit. The tarp covering the engine was torn and some paint was scraped off the jacket water heat exchanger. The damage apparently occurredin transit from the Wisconsin factory and was noticed during a stop in an Ohio switchyard. Inspection at the factory revealed some minor damage to the jacket water heat exchanger. The licensee had already planned to replace the carbon steel heat exchangers on all three engines with copper-nickel heat exchangers when they arrive on site. (The original Hope Creek Unit II purchase specification called for carbon steel, the Shoreham purchase specification calls for copper-nickel.) This engine was returned to the site on June 25, 1984. The third Colt Diesel Generator (#903) is scheduled to arrive on site on July 5, 1984. Installation of the three Colt Diesel Generators in the new diesel generator building is scheduled to begin in the last week of July.

During this inspection period the new Colt Diesel Generator Building construction activities were reviewed by a Region-based NRC inspector who reviewed the quality assurance program and observed rebar installation and concrete placement. The findings will be documented in inspection report 322/84-24.

## 6.0 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 activitees.

During tours of the radwaste building, the inspector noted standing water in several areas of elevation 15' and apparent poor condition of some equipment. Water was dripping from various valves and pumps, floor drains were clogged, and many pieces of equipment were tagged out for maintenance. The licensee is taking corrective actions which include cleaning out the plugged drains and repair of valve packing leaks. Work continues on these items.

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 licensees housekeeping efforts. Unresolved Item 82-04-13, "Monitor Housekeeping Until Fuel Load", will remain open until the corrosion problems in Elevation 8' are corrected.

## 7.0 BISCO Seals

Many thru wall pipe penetrations at Shoreham are sealed with various materials supplied by Brand Industrial Services, Inc. (BISCO). These sealant materials can be used to provide air, moisture, fire break, or radiation protection dams in pipe penetrations.

The inspector performed a review of the program for the installation of BISCO Seals at Shoreham. The inspector reviewed the quality assurance program, implemented by the vendor Keasbey/Bisco, and noted that it included a Quality Assurance Manual which implements the requirements of 10CFR50 Appendix B, and ANSI N45.2. The implementing procedures for the installation and inspection of BISCO Seals were also reviewed. These procedures included instructions for:

- Control of Site Nonconformances;
- Stop Work Orders;
- Calibration of Test Equipment;
- Qualification Test for Silicon Foam Materials;
- Sample Evaluation Testing;
- Standardization for Density Measurements;
- Damming Depth and Penetration Inspection; and,
- Damming Installation.

No deficiencies were identified during this procedure review.

The inspector also reviewed the results of the audits performed by the LILCO Field Quality Assurance Department (FQA) and noted that fifteen audits had been performed between December 3, 1981 and January 19, 1984. These audits found that the installation and the documentation for the seals reviewed during each audit were found to be satisfactory. The scope and depth of the audits were approrpiate and satisfactory corrective actions were taken for those exceptions noted.

The inspector toured the plant and performed a sample re-inspection of all the BSICO Seals installed in two rooms of the radwaste building elevation (31 seals); four flexible boot seals located on the service water penetration to the diesel generator rooms; one flexible boot seal located in the turbine building steam tunnel, one in the reactor building elevation 40', and one in the radwaste building elevation 15'. In all cases the installation was satisfactory and in agreement with the licensees QC installation records. The inspector also performed a walkdown of other areas of the radwaste building and turbine building and did not identify any seals which had been installed incorrectly.

To resolve a concern regarding possible settlement of the metal filler material used in SF-150NH High Density Elastomer, the vendor performed a test to determine if this metal filler material, used to improve radiation shielding, would settle to the bottom of a penetration during curing. On June 4, 1984 the vendor performed Elastomer Settling Test 748-143. The SF-150NH formulation was prepared in accordance with the Standard Quality Assurance Procedures and poured into a one foot, two foot, and three foot section of 3" diameter pipe and allowed to cure for twenty-four hours. Two-inch thick cross-sections were cut from the top, middle, and bottom of each pipe section, then measured and weighed to determine the density of each of the three samples. In the worst case, the density of the bottom sample of the three foot section was only 6% higher than the top sample; indicating no significant settlement of the metal filler material. A review of the QC records revealed that a sample density measurement had been made for each seal installation and the measured density of each sample was greater than the minimum required 150 lbs./ft.$^3$.

The inspector also noted that radiation surveys during operation will monitor any operational streaming problems and that pre-work surveys required under post-accident conditions will identify hot spots and radiation streaming problems, and that corrective action (such as additional shielding) will be required for problem situations.

Based on this sample review, the inspector concluded that the BISCO Seals had been installed satisfactorily.

8.0 Shoreham Shift Advisor Examinations

On June 21, 1984 the licensee administered written exams to the five potential shift advisors. These advisors are all experienced SRO's from other BWR's who will be used to provide additional on-shift operating experience at Shoreham. The inspector observed that the exam was administered with the applicants seated at separate tables and with a training staff instructor acting as a proctor. The inspector also reviewed the written exam and noted it emphasized Shoreham specific material which was appropriate for these experienced SRO's. The exam will also be reviewed by region based license examiners.

No unacceptable conditions were identified.

9.0 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. This is unresolved item 84-23-01.

10.0 SALP Meeting

On June 6, 1984 a Public Meeting was held at Shoreham, New York to discuss the results of the Systematic Assessment of Licensees Performance (SALP) with the licensee. The assessment covered the period February 1, 1983 - February 29, 1984. The attendees included:

NRC Attendees:

T. E. Murley, Regional Administrator
R. W. Starostecki, Director, DPRP
E. C. McCabe, Chief, Reactor Projects Section 1C
C. D. Petrone, Resident Inspector
A. Schwencer, Chief, Licensing Branch 2, NRR

Licensee Attendees:

J. Dye, Executive Vice President
W. Wilm, Assistant to Executive Vice President
J. Leonard, Jr., Vice President - Nuclear
D. Binder, Assistant to Vice President - Nuclear
W. Steiger, Plant Manager
J. Smith, Manager, Nuclear Operations Support Divisions
E. Youngling, Nuclear Engineering Manager

Other Attendees:

A. Elberfeld, New York Public Serivce Commission

## 11.0 Unresolved Items

Areas for which more information is required to determine acceptability are considered unresolved. An unresolved item is contained in paragraph 9.

## 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.

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

LILCO LONG ISLAND LIGHTING

LONG ISLAND LIGHTING COMPANY

175 EAST OLD COUNTRY ROAD · HICKSVILLE, NEW YORK 11301

MILLARD S. POLLOCK
VICE PRESIDENT-NUCLEAR

May 1, 1984 SNRC-1043

Mr. Richard Starostecki, Director
Division of Project and Resident Programs
U.S. Nuclear Regulatory Commission - Region I
King of Prussia, PA 19406

NRC Inspection of January 2 - 13, 1984
Shoreham Nuclear Power Station - Unit 1
Report No. 50-322/84-01

Reference 1: SNRC-1022 dated March 16, 1984

Dear Mr. Starostecki:

This letter responds to your letter dated April 12, 1984 requesting a description of the "methods to be used to ensure only justified exceptions are taken to test or other procedures."

Attached to this letter are two internal memoranda from M. W. Herlihy (Lead Startup Engineer for the Emergency Diesel Generators) to E. J. Youngling (Startup Manager). These provide a summary of the topics discussed at the meetings LILCO referred to in SNRC-1022. Item B of the January 11, 1984 memorandum addresses your additional request. The essence of "personal confirmation of the basis of the test exception" is that test engineers were instructed to seek first hand information upon which to base their judgements and analyses regarding the taking of test exceptions. First hand information includes personally checking a particular parameter, personally verifying the completion of specified work, or receiving direct confirmation from the individual having the necessary first hand information. In addition to these specific instructions LILCO'S Independent Safety Engineering Group, as described in reference 1, provided continuous on-shift coverage of the preoperational testing for the emergency diesel generators beginning January 9, 1984.

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ATTACHMENT I

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Attachment I
Page Two

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January 11, 1984

E. J. Youngling

REVIEW WITH TEST PERSONNEL OF CIRCUMSTANCES OF 101
ENGINE #3 SUBCOVER PROBLEM
Shoreham Nuclear Power Station -- Unit 1
W. O. No. 44430/48923

On the evening of January 10, 1984, I conducted the subject review with all Test Engineers on the Diesel Team (Sign-on sheet attached). Mr. A. Dobrzeniecki, of the administrative group was not present, but I will cover the points pertinent to his area of work with him separately.

Major points of that discussion were:

A. The need for timely identification of exceptions rather than last minute procedure cleanup efforts. Copies of Startup Instruction #10 were handed out and discussed.

B. The need for personal confirmation of the basis of the exception.

C. The unimportance of schedule pressure when taking exceptions.

I am satisfied that all personnel understood the points being made, and will maintain adequate vigilance in this area.

In addition, I covered the role and importance of the ISEG involvement in the Diesel Efforts.

M. W. Herlihy
LSE/DG's

MWH:com
Attachment

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|---|---|
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| Tom Brown | T. Brown |
| Bart Tarlton | B. Tarlton |
| W J Cook | W. Cook |
| Bill Lark | W. Lark |
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| John M Walsh | J. Walsh |
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B ok

January 12, 1984

E. J. Youngling

DISCUSSION WITH TEST PERSONNEL OF CIRCUMSTANCES OF
101 ENGINE #3 SUBCOVER PROBLEM
Shoreham Nuclear Power Station - Unit 1
W. O. No. 44430/48923

REFERENCE: My memo to you, same subject, dated 1/11/84

In the referenced memo, I noted that a discussion with Mr. A. Dobrzeniecki on the subject would be held as he was unable to attend the general meeting.

That discussion was held on January 11, 1984, noting the same points. He understood the points being made.

Therefore, all Test Engineers of the Diesel Team have been briefed on the problem, and informed about the ISEG role in the Diesel Generator operations.

M. W. Herlihy

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LSE/DG's

MWH:com

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