October 6, 1982

Lawrence Brenner, Esq.
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. James L. Carpenter
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Peter A. Morris
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
Long Island Lighting Company
(Shoreham Nuclear Power Station, Unit 1)
Docket No. 50-322 (OL)

Dear Administrative Judges:

The following is an update of the status of the NRC Staff review for those SER open items impacting the unscheduled contentions in this proceeding:

Remote Shutdown Panel
(SC 1; Open Item #62)

Attached is the Staff's final SER on the Shoreham Remote Shutdown Panel. (Attachment 1). The SER is in the form of an input from T. Speis to T. Novak, dated September 29, 1982. Open Item #62 is now considered resolved. Routine verifications will be performed by Region I prior to start-up and at the first refueling outage. The Staff is prepared to write testimony on contention SC 1, and believes that the litigation schedule should begin to run upon receipt of this letter.

Environmental Qualification
(SC 8/SOC 19(h); SC 32/SOC 19(f); Open Item #9)

On September 20, 1982, the Staff sent LILCO a formal Request for Additional Information, A. Schwencer to M.S. Pollock, which includes a request on environmental qualification. (Attachment 2). Enclosure B to that request is in the form of an SER input. The SER input summarizes the environmental

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qualification program and review to date, and notes the open areas of the review. The SER, therefore, requests information rather than closes the issue.

This SER input had been originally made available to LILCO informally during August 1982. The Staff's September 3, 1982 update on this item essentially described the status of the review as noted in the SER input. LILCO responded to the deficiencies identified in the SER input with SNRC-767, J.L. Smith to H.R. Denton, September 9, 1982. The Staff is currently reviewing that large submittal to determine whether it closes out any of the open areas. We will keep the Board and parties informed as to any developments.

Seismic Qualification
(SOC 19(i); Open Item #8)

As noted in the September 3, 1982 update on the status of this item, the second SQRT audit was conducted at Shoreham from August 31 to September 3, 1982. Since that audit the Staff has been assessing the results, and LILCO has supplied some of the additional information requested by the reviewers during the audit. A trip report is in preparation which will contain the Staff's generic comments on LILCO's seismic qualification program and note any specific deficiencies. The report will provide a good indication of how much work remains to be done on this open item. The report is expected within the next two weeks. We will promptly provide copies to the Board and parties when it is issued.

Containment Isolation
(SC 23; Open Items #36, #II.E.4.2, #61)

The first open issue under containment isolation, #36, involved the operability of the 6-inch valves in the vent lines. This issue has been considered resolved since the Staff's June 29, 1982 Status Report on SER Open Items. The SER input provided at that time will be the final SER on the issue. However, we noted in out September 3, 1982 update that the results of an in-situ valve test would have to be verified. For the information of the Board and parties, the test has now been verified by Region I and the results written-up in Inspection Report 50-322/82-23, page 6, at item 5, September 16, 1982. (Attachment 3).

The second open issue, #II.E.4.2, has been narrowed to LILCO's justification for late installation of the high radiation signal. The Staff is still in the process of preparing a response to LILCO's proposed justification, submitted in SNRC-767.

On Item #61, concerning NUREG-0803, the status also is unchanged from that indicated in the September 3, 1982 update. The Staff review of LILCO's submittal on this subject, however, is progressing. Some questions have been transmitted to LILCO informally. A complete formal request for additional information will be compiled when all the relevant reviewers have completed their piece of the review. The complete request is expected to be transmitted to LILCO in mid-October.

In the Staff's September 3, 1982 update on containment isolation it was noted that a fourth open issue had been discovered, involving changed arrangements for the containment isolation instrument lines. A formal Request for Additional Information has been transmitted to LILCO as the second part of the Staff's September 20, 1982 omnibus request. (Attachment 2. The third part of that request involved environmental qualification).

As always, the Staff will endeavor to keep the Board and parties informed as to developments on this and all the other issues impacting the unscheduled contentions.

Sincerely,

David A. Repka
Counsel for NRC Staff

Enclosures: As Stated

cc: (w/enclosures)

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Howard L. Blau, Esq.	W. Taylor Reveley, III, Esq.
Cherif Sedkey, Esq.	Stephen B. Latham, Esq.
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MHB Technical Associates	Hon. Peter Cohalan

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DATE	:10/5/82	:10/15/82	:	:	:

ATTACHMENT 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 29 1982

MEMORANDUM FOR: Thomas Novak, Assistant Director for Licensing
Division of Licensing

FROM: Themis P. Speis, Assistant Director for Reactor Safety
Division of Systems Integration

SUBJECT: SHOREHAM NUCLEAR POWER STATION
SUPPLEMENTAL SAFETY EVALUATION REPORT

Plant Name: Shoreham Nuclear Power Station
Docket No.: 50-322
Licensing Stage: OL
Responsible Branch: LB #2
Project Manager: R. Gilbert
Review Branch: ICSB
Review Status: Complete

The purpose of this memorandum is to provide additional SER input based on information submitted by the applicant in letters dated November 23, 1981; April 20, 1982; and August 24, 1982 from J.L. Smith to Harold R. Denton.

Our evaluation of this information with regard to the Remote Shutdown Panel is enclosed as input for the next SER supplemental. This evaluation resolves this ICSB open item for the Shoreham Nuclear Power Station. It should be noted that SSER input will not be provided for confirmatory items unless a problem is found when final documentation is submitted.

If you have any questions contact J. L. Mauck in the ICSB.

Brian W Sheron for

Themis P. Speis, Assistant Director
for Reactor Safety
Division of Systems Integration

Enclosure:
Shoreham Supplemental SER

cc: See attached list

Contact:
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NOS 10/27/82
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7.4.3 Remote Shutdown System

GDC 19 requires in part that the ability be provided for the safe shutdown of the plant in case the main control room becomes uninhabitable. Plant designs should provide for control stations in locations removed from the main control room. These stations are to be used for manual control and alignment operations needed to achieve and maintain a hot shutdown and subsequently to be able to achieve a cold shutdown. The applicant has provided a remote shutdown panel located within an enclosure in the reactor building. Except for reactor scram, which can be initiated from other remote locations, this panel allows the operator to bring the reactor to the cold shutdown condition in an orderly fashion and includes all instrumentation and controls required for operating the needed systems.

The following systems can be operated and monitored from this remote shutdown panel:

(1) RCIC--turbine and valves

(2) SRV's--three solenoids

(3) RHR--pump, valves, and flow indication

(4) RBSW--pump, valves, and discharge pressure

(5) RBCLCW--pump, and inlet valves

(6) Fuel Pool Cooling System B--pump

(7) miscellaneous recorders for reactor pressure vessel (RPV) level, RPV pressure, drywell pressure, drywell temperature, suppression pool level, and suppression pool temperature.

The remote shutdown capability is designed to control the required shutdown systems (one division of equipment) from outside the control room irrespective of shorts, opens, or grounds in the control circuits in the control room that may have resulted from an event causing an evacuation. The functions needed for remote shutdown control (one division of equipment) are provided with manual transfer switches that override controls from the control room and transfer the controls to the remote shutdown panel.

When transferring control to the remote shutdown panel, controls for some functions are transferred to maintained contact switches. Assuming an orderly transfer of control to the RSP, the operator would have time to mimic the system's alignment of the main control panel before transferring control of the RSP. In a situation where realignment of the RSP prior to the transfer of control is not possible, station procedure SP 23.133.01 states the proper position for each switch during normal plant operation. These switch positions will be such that, when control is transferred to the RSP, system equipment will go to an alignment which will ensure no damage to either the system or the equipment. The system will then be at a predetermined alignment, from which the operator can continue an orderly shutdown. The specific methods employed in these operations are described in station procedure 29.022.01, Shutdown From Outside Control Room Emergency Procedure, and in related station procedures.

The Shoreham design provides only for the transfer of one train of equipment to the remote shutdown panel. It appeared to the staff that, given a failure in this train of equipment, sufficient instrumentation and controls would not be available to attain a shutdown condition from outside the control room. It is the staff's position that to meet GDC 19, the Remote Shutdown System (RSS) design should provide redundant safety grade capability to achieve and maintain hot shutdown and subsequently cold shutdown from a location or locations remote from the control room, assuming no fire damage to any required systems and equipment and assuming no accident has occurred. Credit may be taken for manual actuation (exclusive of continuous control) of systems from locations that are reasonably accessible from the Remote Shutdown Panel. Credit may not be taken for manual actions involving jumpering, rewiring or disconnecting circuits.

As a result of the staff's inquiries, the applicant has committed to provide and/or identify additional instrumentation and controls to meet the single failure criterion prior to fuel load (except where noted below). The proposed additional Instrumentation and Controls are as follows:

(1) RCIC -- Nothing additional; assume automatic operation of HPCS to maintain RPV water level

(2) Safety Relef Valves (SRV's) -- Provide controls for the division II SRV's on a local panel in the division II relay room.

(3) RHR -- Provide controls for the equivalent RHRA pump and valves from the division I emergency switchgear room and the Reactor Building Secondary Containment (RBSC). An RHR A flow indicator will be provided on a local panel. This flow indication will not be added until the first refueling.

(4) Reactor Building Service Water (RBSW) -- Provide controls for the equivalent train A pump and valves from the emergency switchgear room and the screenwell pump house. RBSW train A flow indication will be provided in a local panel by first refueling.

(5) Reactor Building Closed Loop Cooling Water (RBCLCW) System -- Nothing additional; A single failure of this system would prevent the use of the normal RHR flow path whenever fluid temperatures exceed 212°F. However, a circulation path using suppression pool water could be established through the RHR heat exchanger using the "B" RHR pump on the RSP in the LPCI mode. Flow would return to the suppression pool from the reactor pressure vessel (RPV) via the RSP controlled SRV's. The RHR pump can operate in this mode without RBCLCW cooling.

(6) Spent Fuel Pool Cooling -- Provide pump controls for the "A" Spent Fuel Pool Cooling pump on a local panel by first refueling.

(7) Miscellaneous -- Provide a division 2 indicator for RPV level, a division 2 indicator for RPV pressure (by first refueling), division 1 indicator for suppression pool level, division 1 & 2 indicators for suppression pool temperature (by first refueling).

The staff has concluded that the modifications to the RSS are an acceptable method for implementing the staff's position for redundant safety-grade capability. As noted above, several items will not be implemented until the first refueling outage. This is acceptable to the staff since there is an extremely low probability that an event will cause an evacuation of the control room to occur, concurrent with a single failure in the primary shutdown path at the RSP during the first cycle of plant operation. In addition, the redundant systems will still be operable from remote locations since only the indication for certain parameters will not be available until the first refueling outage.

The applicant has indicated that several of the readouts and associated sensors and power supplies on the remote shutdown panel are not safety grade. The applicant has reviewed the design to determine whether these non-safety grade readouts are required to achieve shutdown and has committed to upgrade them accordingly. The following primary path readouts are not presently classified as safety-grade:

(1) RHR "B" Flow

(2) RPV level

(3) RPV pressure

(4) Service Water "B" Header Pressure

(5) Suppression pool temperatures

(6) Suppression pool level

(7) RCIC flow

(8) RCIC turbine Speed

(9) SRV N_2 Pressure

The applicant has stated that the above readouts are to be upgraded to Quality Assurance Category I prior to the conclusion of the first refueling outage. This delay in implementation is due to the long leadtime associated with the procurement for the above instrumentation. At this time the RSP and all of the equipment located on the panel will be seismically qualified and environmentally qualified for its normal operating environment. This is acceptable to the staff due to the low probability of a seismic event occurring simultaneously with an event causing evacuation of the control room during the first cycle of plant operation.

In summary, the staff has reviewed the applicant's latest Remote Shutdown Panel design and has concluded that it will meet the regulatory requirements specified in GDC 19 and the guidance as detailed in the Standard Review Plan (NUREG-0800), Section 7.4.II and III. However, as a confirmatory item, the applicant must provide acceptable final operating procedures and technical specifications for the Remote Shutdown Panel, and perform an acceptable procedure verification test before plant start-up. In addition, as a license condition, the applicant must implement all of the required design changes by the end of the first refueling, provide acceptable final operating procedures and technical specifications for the new RSP design, and perform an acceptable procedure verification test for the new RSP design This verification test should include simulated system operability from remote stations away from the Remote Shutdown Panel with the assumption of the most limiting single failure in the equipment train controlled from the Remote Shutdown Panel. The verification should include a test of all communications required to accomplish the shutdown.

ATTACHMENT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 2 0 1982

Docket No.: 50-322

Mr. M. S. Pollock
Vice President - Nuclear
Long Island Lighting Company
175 East Old Country Road
Hicksville, New York 11801

Dear Mr. Pollock:

Subject: Request for Additional Information - Shoreham Nuclear Power Station

As a result of our review of your application for an operating license for the Shoreham Nuclear Power Station, we find that we need additional information in the areas of instrumentation and control, containment systems and the environmental qualification of safety-related electrical equipment.

The specific request for information in the area of instrumentation and control is stated in Enclosure A as staff question 223.100.

As part of the staff's review of the containment isolation arrangement for the Shoreham Station, the instrument lines were reviewed against the provisions of Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment." Since the publication of the staff's review of the containment isolation arrangement in the first supplement to the SER, you submitted Amendment 44 to your license application, which contained changes in the containment isolation arrangement design. Specifically, Amendment 44 stated that non-safety related instrument lines penetrating primary containment existed that relied on an orifice and a manual valve in each line to comply with the isolation provisions of Regulatory Guide 1.11. The considerations in the Supplement to Regulatory Guide 1.11 that apply to the Shoreham Station include the following:

a. Each instrument line connected to the reactor coolant pressure boundary and penetrating containment should be sized, or should include an orifice, such that if a postulated failure of the piping or of any component (including the postulated rupture of any valve body) in the line outside primary reactor containment occurs during normal reactor operation:

   (1) The leakage is reduced to the maximum extent practical consistent with other safety requirements;

   (2) The rate and extent of coolant loss are within the capability of the reactor coolant makeup system;

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(3) The integrity and functional performance of secondary containment, if provided, and associated safety systems (e.g., filters, standby gas treatment system) will be maintained; and

(4) The potential offsite exposure will be substantially below the guidelines of 10 CFR Part 100.

b. For each instrument line penetrating containment, including those connected to the containment atmosphere, some method of verifying during operation the status (open or closed) of each isolation valve should be provided.

After reviewing the information provided in the Shoreham FSAR on this issue, the staff concludes that the above considerations have not been adequately addressed for the Shoreham Station.

Enclosure B contains a Safety Evaluation Report on the environmental qualifications of safety-related electrical equipment. Requests for the submission of outstanding information by you are included in Sections 3 and 4 of Enclosure B.

It is requested that you inform my staff within seven (7) days of receipt of this letter of your schedule for submitting the required information. If you have any questions, please contact Ed Weinkam, Project Manager at (301) 492-8430.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
As stated

cc: See next page

Shoreham

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ENCLOSURE A

The following question is based on the Applicant's response dated April, 1982, to the staff question (223.99) that was transmitted to the applicant on February 11, 1982 (NRC letter from A. Schwencer to M. S. Pollock).

223.100 In the Applicant's response to question 223.99, it was indicated that valves in the Control Room Air Conditioning (CRAC) System would revert to a normal position if switch 1A2 was reset. It was also indicated that this reset would occur only if the CRAC System were manually initiated and not automatically initiated. This indicates to the staff that the Applicant apparently has not included a review of Engineered Safety Features (ESF) resets when the ESF Systems are manually initiated even though the review for IE Bulletin 80-06 (ESF Resets) is intended to cover both automatic and manual initiations. Therefore, the Staff has concluded that the Applicant should include a review of manual initiation for the ESF Systems and provide an updated submittal concerning IE Bulletin 80-06.

SAFETY EVALUATION REPORT BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
EQUIPMENT QUALIFICATION BRANCH
FOR LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION UNIT NO. 1
DOCKET NO. 50-322

SAFETY EVALUAION REPORT
OFFICE OF NUCLEAR REACTOR REGULATION
EQUIPMENT QUALIFICATION BRANCH
SHOREHAM NUCLEAR POWER STATION UNIT NO. 1
DOCKET NO. 50-322

ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

1 INTRODUCTION

Equipment which is used to perform a necessary safety function must be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1, 2, and 4 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, is applicable to equipment located inside as well as outside containment. More detailed guidance relating to the methods and procedures for demonstrating this capability has been set forth in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE Standard 323, and NRC Regulatory Guides which endorse ancillary daughter standards (e.g., IEEE Stds. 317, 334, 382, and 383).

2 BACKGROUND

NUREG-0588 was issued in December, 1979 to promote a more orderly and systematic implementation of electrical equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews. The positions contained in this report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods which are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation.

Commission Memorandum and Order CLI-80-21 issued on May 23, 1980 states that NUREG-0588 forms the requirements that license applicants must meet in order to satisfy those aspects of 10 CFR Part 50, Appendix A, GDC 4 which relate to environmental qualification of safety-related electrical equipment. IE Bulletin 79-01B "Environmental Qualification of Class 1E Equipment," issued January 14, 1980, and its supplements dated February 29, September 30, and October 24, 1980 established environmental qualification requirements for operating reactors. This bulletin and its supplements were provided to OL applicants for consideration in their review.

In response to the above, the applicant provided equipment qualification information by letters dated May 27, 1981, January 25, 1982, May 17, 1982, and July 8, 1982 to supplement the information contained in Section 3.11 of the FSAR.

2.1 Purpose

The purpose of this SER is to evaluate the adequacy of the Shoreham Nuclear Power Station environmental qualification program for safety-related electrical equipment. The staff position relating to any open items is provided in this report, as well as identification of any unresolved issues.

2.2 Scope

The scope of this report includes an evaluation of the list of systems and electrical equipment to be qualified, the criteria which they must meet, the environments in which they must function, and an assessment of the qualification documentation for equipment. It is limited to safety-related electrical equipment which must function in order to prevent or mitigate the consequences of a loss-of-coolant accident, or high or moderate energy line breaks, inside or outside of containment, while subjected to the harsh environments associated with these accidents. Equipment required to mitigate SDV breaks as described in NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," will be evaluated separately.

3 STAFF EVALUATION

The staff evaluation of the applicant's response included an onsite examination of Class 1E equipment, audits of qualification documentation, and a review of the applicant's submittals for completeness and acceptability of systems and components, qualification methods, and accident environments. The criteria described in NUREG-0588 Category II form the basis for the staff evaluation of the adequacy of the applicant's qualification program. Revision 1 of NUREG-0588 was utilized to clarify staff positions as required.

The staff performed audits of the applicant's qualification documentation and installed equipment on April 26-30 and June 2-3, 1982. The audits consisted of a review of approximately 20% of the applicant's equipment.

3.1 Completeness of Safety-Related Equipment

The applicant was directed to (1) establish a list of systems and components that are required to prevent or mitigate a LOCA and an HELB and (2) identify components needed to perform the function of safety-related display instrumentation, post-accident sampling and monitoring, and radiation monitoring.

The applicant's systems list for the environmental qualification program was compared to Table 3.2.1-1 of the FSAR. The subset of systems from Table 3.2.1-1 containing electrical equipment which are required for emergency shutdown or accident mitigation was reviewed by the staff and found to be acceptable, except the Suppression Pool Pumpback System, which is required for flood prevention in the secondary containment, should be included in the applicant's program. The staff also reviewed the applicant's operability times, Class 1E safety-functions, and required accidents for selected systems and components. With the exception of the RCIC system, the classification of the above was satisfactory. For the RCIC system, the applicant has stated that its equipment is not required to function for accident mitigation. Equipment in the RCIC system should either be qualified to demonstrate operability during accidents or the basis for its exemption from the program provided.

In addition to the above, the staff reviewed the list of equipment in a harsh environment and determined which systems in the master list (harsh and mild) had been omitted. With the exception of the following, the omissions were adequately explained as systems located only in a mild environment:

AC Uninterruptible Power (R36)

Condensate Transfer and Storage (P11)

Some equipment required by Regulatory Guide 1.97 Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," was included in the environmental qualification program. Full implementation of the requirements must be achieved by June 1983. The items identified were reviewed in the same manner as other safety-related equipment.

nentation in the emergency operating procedures which provides the reactor operators to aid them in the safe handling of the specifically identified by the applicant. All essential instru- equipment should be identified and added to the environmental program.

equipment required by NUREG-0737 "Clarification of TMI Action nts" was also evaluated by the staff. The Action Plan items Chapter 16 of the FSAR were compared to the information presented mental qualification submittal. Because several items do not program, the applicant should provide a summary of the status of all equipment required to be qualified by the TMI

identified 131 types of equipment which were assessed by the ese, 49 are conditionally qualified, and for the remaining 82, er being retested or replaced.

Conditions

fines the methods to be utilized for determining the environmental sociated with loss-of-coolant accidents or high energy line e or outside of containment. The review and evaluation of the hese environmental conditions are described below. The staff has qualification documentation to ensure that the qualification velop the conditions established by the applicant.

ature, Pressure, and Humidity Conditions Inside the Primary nment

provided the LOCA/MSLB profiles used for equipment qualification ttal. The peak values in the drywell resulting from these as follows:

Maximum Temperature, °F	Maximum Pressure, psig	Humidity, %
340°F	48	100

s reviewed these profiles and finds them acceptable for use in alification; i.e., there is reasonable assurance that the actual d temperatures will not exceed these profiles anywhere within the vironmental zone (except in the break zone).

ll, the staff calculated a temperature profile for steam bypass temperature of 270°F. Equipment in the wetwell was evaluated by it using this profile.

3.2.2 Temperature, Pressure, and Humidity Conditions Outside the Primary Containment

The applicant has provided the temperature, pressure, and humidity conditions associated with high and moderate energy line breaks in the secondary containment. The staff has used a screening criterion of saturation temperature at the calculated pressure to verify that the parameters identified by the applicant are acceptable.

3.2.3 Submergence

The maximum submergence levels have been established by the applicant in the environmental qualification program. Unless otherwise noted, the staff assumed for this review that the methodology employed by the applicant is in accordance with the appropriate criteria in NUREG-0588.

The applicant's values for maximum submergence levels are 63'4" in the drywell and 47'0" (including froth) in the wetwell during suppression pool swell. Essential equipment below these levels has been qualified.

The effects of flooding on safety-related electrical equipment in the reactor building is discussed in Appendix 3C of the FSAR and has been reviewed and approved by the staff, except as noted below. The maximum submergence level is twenty-two inches above the 8' elevation. The applicant, however, should address potential wetting of equipment due to drainage through stair openings, etc., to the 8' elevation.

3.2.4 Demineralized Spray

Demineralized water is available for primary containment heat removal following an accident. The applicant included this environmental parameter in the evaluation of equipment qualification.

3.2.5 Aging

NUREG-0588 Category II delineates two aging program requirements. Valve operators committed to IEEE Standard 382-1972 and motors committed to IEEE Standard 334-1971 must meet the Category I requirements of the NUREG. This requires the establishment of a qualified life, with maintenance and replacement schedules based on the findings. All other equipment must be subjected to an aging program which identifies aging-susceptible materials within the equipment. Additionally, the staff requires the applicant to:

a) Establish an ongoing program to review surveillance and maintenance records to identify potential age-related degradation.

b) Establish component maintenance and replacement schedules which include considerations of aging characteristics of installed components.

The applicant has established a qualified life for each qualified equipment type through test or analysis. In addition, the applicant has developed a plan for surveillance and maintenance to ensure that equipment will not degrade sooner than predicted. The plan will incorporate failure history data from

LERs, plant operating experience, manufacturers' recommendations, and other pertinent information in the central files. The staff has reviewed the outline of this plan and finds it acceptable. Surveillance and maintenance procedures are to be implemented prior to full power operation. Until these procedures are implemented, aging will remain an open item. The applicant is requested to notify the staff when the procedures are implemented.

The applicant has described the policy for replacement equipment and components in the May 17, 1982 revision to the program. The replacement program, however, differs somewhat from the description previously presented to the staff and does not conform to staff clarifications of requirements for replacement parts. The applicant should revise the program to conform to the appropriate requirements.

3.2.6 Radiation (Inside and Outside Primary Containment)

The applicant has provided values for the radiation levels postulated to exist following a LOCA. The application and methodology employed to determine these values were presented to the applicant in NUREG-0588 and NUREG-0737 "Clarification of TMI Action Plan Requirements." The staff review determined that the values to which equipment was qualified enveloped the requirements identified by the applicant.

The values specified by the applicant in the drywell are integrated doses of 1×10^8 to 1.7×10^8 rads gamma and 1×10^9 rads beta. In the secondary containment, required values of 3.06×10^6 to 1.27×10^8 rads gamma were used in the evaluation of equipment in areas exposed to recirculating fluid lines.

The values used for qualification of equipment are identical to those in the applicant's response to TMI Action Plan Item II.B.2 in the FSAR and are acceptable.

3.3 Outstanding Equipment

For most items not having complete qualification documentation, the applicant provided commitments for corrective action and schedules for completion. Where complete qualification documentation will not be available by fuel load, the applicant must provide for staff review at least three months prior to fuel load justifications for interim operation to provide assurance that required safety functions can be accomplished during accidents. The staff will review this information to determine if interim operation with this equipment will not degrade safety functions or inhibit accident mitigation systems or equipment in the unlikely event of exposure to an adverse environment.

4 QUALIFICATION OF EQUIPMENT

The following subsections present the staff assessment, based on the applicant's submittal, audits of documentation at the plant site, information in the NRC Equipment Qualification Data Bank, and previous staff evaluations of equipment in other plants.

The staff has separated the safety-related equipment into three categories: (1) equipment requiring replacement prior to plant start-up, (2) equipment requiring additional qualification information or corrective action, and (3) equipment considered acceptable pending implementation of the maintenance and surveillance program. An appendix listing equipment in each of these categories is provided.

4.1 Equipment Requiring Replacement Prior to Plant Start-up

Appendix A identifies equipment which the staff review has determined requires replacement prior to plant start-up. There is no equipment in this category for Shoreham Unit 1.

4.2 Equipment Requiring Additional Information and/or Corrective Action

Appendix B identifies equipment in this category. Corrective action or deficiencies are noted by a letter relating to the legend identified below.

Legend

A - material-aging evaluation; replacement schedule; ongoing equipment surveillance

CS - chemical spray

EXN - exempted equipment justification inadequate

H - humidity

I - HELB evaluation outside containment not completed

M - margin

P - pressure

QI - qualification information being developed

QM - qualification method

QT - qualification time

R - radiation

RPN - equipment relocation or replacement; adequate schedule not provided

RPS - equipment relocation or replacement schedule provided

RT - required time

S - submergence

SEN - separate effects qualification justification inadequate

T - temperature

As noted in Section 4, these deficiencies do not necessarily mean that the equipment is unqualified. However, the deficiencies are cause for concern and require further case-by-case evaluation.

For each equipment item identified in Appendix B, the applicant must perform an analysis to ensure that the plant can be operated safely pending completion of environmental qualification. This analysis must include, as appropriate, consideration of:

(a) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(b) The validity of partial test data in support of the original qualification.

(c) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(d) Completion of the safety function prior to exposure to the ensuing accident environment and the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(e) No significant degradation of any safety function or misleading of the operator as a result of failure of equipment under the accident environment.

4.3 Equipment Considered Acceptable or Conditionally Acceptable

Based on the staff review, the items identified in Appendix C have been determined to be acceptable, pending implementation of the maintenance/surveillance program. Before exceeding low-power operation, the applicant is required to inform the staff of the implementation of the maintenance/surveillance program.

5 CONCLUSIONS

The staff has determined that the applicant's listing of safety-related systems and associated electrical equipment required to function in an accident environment is complete and acceptable, except as noted in Section 3 of this report. The staff has also determined that the environmental conditions under which equipment must function are appropriate, except as noted in Section 3 of this report. The applicant's response to the outstanding items in Section 3 will be addressed in a supplement to this report.

The staff has reviewed the qualification of safety-related electrical equipment to the extent defined by this SER and finds the qualification status or corrective action plan for achieving full qualification to be acceptable. The applicant has established a test and replacement program with schedules for obtaining full qualification of equipment.

Subsection 4.2 identified items that must be resolved to establish the qualification of equipment; the staff requires the applicant to update the component worksheets when the noted deficiencies are resolved. The applicant must provide justifications for interim operation with equipment not fully qualified prior to fuel load.

Subsection 4.3 identified equipment which is conditionally acceptable pending implementation of the maintenance/surveillance program. The applicant should implement this program prior to exceeding low-power operation and notify the staff when implementation has occurred.

Based on these considerations, the staff concludes that satisfactory completion of the corrective actions and resolution of the outstanding items identified herein will ensure compliance with the Commission Memorandum and Order of May 23, 1980, and relevant parts of General Design Criteria 1 and 4 of Appendix A and Sections III and XI of Appendix B, 10 CFR Part 50 for equipment exposed to accident conditions.

APPENDIX A

EQUIPMENT REQUIRING
REPLACEMENT PRIOR TO PLANT STARTUP
(Category 4.1)

No equipment in this category

APPENDIX B

EQUIPMENT REQUIRING ADDITIONAL INFORMATION
OR CORRECTIVE ACTION

	Equipment	Manufacturer	Model	Deficiency/ corrective action
1.	Motor control centers	General Electric	DC MCC	Retest by 1984
2.	Motor control center	Square D	Model 4	Retest by 1983
3.	Breaker	GE	M26	Modify by 9/82
4.	Solenoid operated valve	ASCO	MV200-926-1F-EP	Replace by 9/82
5.	Temperature control valve	Beck	14-101-023645(ES)	Retest by 2/83
6.	Motor operated valve[1]	Limitorque	SMB Series	QI (84 items)
7.	Motor operated damper	ITT	NH91	Retest by 12/82
8.	Motor operated damper	Raymond	MASR-49	Retest by 12/82
9.	Motor operated damper	Raymond	MASR-9	Replace by first refueling outage
10.	Flow transmitter	Air Monitor Corp.	Veltron 800	Retest by 12/82
11.	Transmitters	Rosemount	1152	Circuit boards to be replaced by 9/82 (18 items)
12.	Pressure switch	ASCO	SB11AKR/TF10A32B	Retest by 9/82
13.	Pressure switch	ASCO	SB11AKR/TG10A32B	Retest by 9/82
14.	DP switch	Dwyer	1627	Retest by 12/82
15.	Level switch	Magnetrol	291-MPG-X-M14DC	Retest by 12/82
16.	Level element	GEMS	XM-54854	Retest by 10/82
17.	Radiation element	Kamen	---	Retest by 8/82.
18.	Position switch	Namco	EA740	Replace by 9/82
19.	Position switch	Namco	EA750	Replace by 9/82
20.	Transfer switch	ASCO	307A66C	Retest by 6/82
21.	Selector switch	GE	CR2940	Retest by 8/82

Appendix B (continued)

Equipment	Manufacturer	Model	Deficiencies/ corrective action
22. Panel	Atomics Int.	---	Retest and replace subcomponents by 9/82
23. Panel	Gould	5600 Series	QI
24. Panel	Kamen	---	Retest by 8/82
25. Pressure Indicator	Marsh Gage	H0212	QI
26. Panel	Square D	Bkr. Dist.	Retest by 1983
27. Panel	Square D	480V	Retest by 1983
28. Flex conduit	Electro Flex	CEA Sealtite	Retest by 8/82
29. Conduit couplings	Service-Air, Amer. Boa.	---	Retest by 8/82
30. Penetration	Conax	Low Volt. Power	QI (2 items)
31. Penetration	GE	Series 100 Med Volt	I, A, R
32. Penetration	GE	Series 200 Med Volt	I, A, QT, R
33. Tape	Okonite	T35, T95	Retest by 8/82
34. Insulating material	Raychem	WCSF-N	Retest by 8/82
35. Chico compounds	Crouse-Hinds	Chico (X), (A)	QI
36. Terminal blocks	GE	EB25A04W, -12W	Retest by 8/82
37. Terminal blocks	GE	EBI	Retest by 8/82
38. Terminal blocks	GE	CR151	Retest by 8/82
39. Recombiner	Atomics Int.	---	Retest and replace subcomponents by 9/82
40. Hydrogen analyzer	Comsip	B	Retest sample pump by 6/82
41. Oxygen analyzer	Comsip	J	Retest sample pump by 6/82
42. Filter train	Farr	N-240	Retest by 9/82
43. Solenoid operated valve	ASCO	HT-X-8320A20	Replace, first refueling outage
44. Solenoid operated valve	Target Rock	1/2 SMS-A-1	QI

Appendix B (continued)

Equipment	Manufacturer	Model	Deficiencies/ corrective action
45. Explosive valve	Conax	1832-159-01	QI
46. Motor operated valve	Limitorque	SMB-2	QI (2 items)
47. Motor operated valve	Limitorque	SMB-3	Replace actuator by 9/82 (2 items)
48. Pump motor	GE	CD259A7	Retest by 1/83
49. Pump motor	GE	1.5HP, 120VDC	QI
50. Pump motor	GE	3HP, 1900RPM	Retest by 1/83
51. Pump motor	GE	3HP, 3500RPM	Retest by 1/83
52. Pump motor	GE	5K324AK2084	QI
53. E/P converter	Fisher Governor	546	QI
54. Flow transmitter	Ametek	078-5004	Retest by 3/83
55. Pressure transmitter	Bailey	KG556	Replace, first refueling outage
56. Level transmitter	Barton	368	Replace, first refueling outage
57. Level indicating transmitter	Barton	760	Replace, first refueling outage
58. Transmitters	Rosemount	1151	QI
59. Pressure switch	Barksdale	B1T	Retest (5)
60. Pressure switch	Barksdale	D2H	QI
61. DP switch	Barton	288A/289A	QI (16 items)
62. Position switch	Namco	D1200	Retest by 1/83
63. Position switch	Namco	EA740	Replace by 9/82
64. Level switch	GE	---	QI
65. Position switch	Not determined	---	QI
66. Pressure switch	Square D	9012,ACW-12	Retest by 1/83
67. Level switch	Square D	9036	Retest by 1/83
68. Level switch	Square D	9038-AG154	Retest by 1/83
69. Pressure switch	Static-O-Ring	5N,6N	QI
70. Radiation element	GE	237X731G001	QI
71. Flow element	Schutte & Koerting	---	Retest by 3/83

Equipment	Manufacturer	Model	Deficiencies/ corrective action
72. Blower motor	GE	2CH6 041-U	QI
73. Turbine	Terry	GS-1	Retest by 1/83
74. Temperature element	Pyco/Calif. Alloy/NECI	145C3224/ 145C3234 Series	QI
75. Panel	GE	---	Retest by 2/83
76. Panel	GE	M26	Retest by 2/83
77. Level switch	Magnetrol	5.0-751-2X-MPG-M14HY	QI
78. Temperature element	Pyco	102-9039-08	QI
79. Switch	NMC	PMC-8000	QI
80. Radiation element	NMC	SC-2-15, SC-2B	QI
81. Temperature element (inside drywell)	Rosemount	89-86-4/88-14-1	QI
82. Motor starter	Square D	Size 5, 460V	QI

[1]Limitorque valve operators were purchased to various specifications, contain various motors, and reference several test reports. Qualification evaluation is based on the above and location in the plant.

APPENDIX C

EQUIPMENT CONSIDERED ACCEPTABLE
PENDING IMPLEMENTATION OF AGING PROGRAM
(Category 4.3)

Equipment	Manufacturer	Model no.
1. Solenoid operated valve	ASCO	NP8316E36E
2. Pump motor	GE	5K6339XC157A
3. Pump motor	GE	5K6339XC94A
4. Temperature element	Pyco	102-3171
5. Pressure switch	Barksdale	P1H
6. Pressure switch (6 items)	Barksdale	B1T
7. Pressure switch (8 items)	Barton	288A/289A
8. Level switch	Magnetrol	3.5-751-1X-MPG-M14HY
9. Solenoid operated valve	ASCO	WJHKX-8320-A89E
10. Solenoid operated valve	ASCO	WJK-206-380-6F
11. Motor operated valve (130 items)[1]	Limitorque	SMB Series
12. Air operated damper	Centerline	32046-6
13. Air operated damper	Powers	331-2792
14. Pump motor	Reliance	100-HP-444T
15. Pump motor	Reliance	30-HP-326T
16. Instrument cable	Brand Rex	Low Capacitance Cable
17. Cable	Kerite	5KV Power Cable
18. Cable	Okonite	600V Power Cable
19. Instrument cable	Raychem	---
20. Instrument cable	Rockbestos	Coax/Triax
21. TC wire	Rockbestos	---
22. Control and instrument cable	Rockbestos	300/600V
23. Switchboard wire	GE	Vulkene Supreme
24. Transmitter	Rosemount	1152 Series
25. Transmitter	Rosemount	1153GB Series
26. Temperature element	Rosemount	88-149-1

Appendix C (continued)

Equipment	Manufacturer	Model no.
27. Temperature element	Rosemount	88-149-2
28. Temperature element	Rosemount	89-138-2/88-14-3
29. Temperature element (outside drywell)	Rosemount	89-86-4/88-14-1
30. Temperature element	Rosemount	89-86-4/88-14-3
31. Position switch	Namco	EA180
32. Fan motor	Westinghouse	143TCZ
33. Fan motor	Westinghouse	286T
34. Fan motor	Westinghouse	326T
35. Fan motor	Westinghouse	364T
36. Fan motor	Westinghouse	405TCZ
37. Fan motor	Westinghouse	7.5HP/245T
38. Blower motor	Reliance	324T
39. Heater	GE	47D518673
40. Panel	Comsip	K-IV
41. Panel	Systems Control	120VAC Distr.
42. Electrical penetration (2 items)	Conax	Low Voltage Power
43. Lugs and splices	Amp	52900-53900
44. Tape	Kerite	S-5MT-NUC
45. Motor generator	Louis Allis	COGSF
46. Transformer	Magnetics	L-12514
47. Solenoid valve	Valcor	V105-205, 305; V526-5295-61, 62, 63; V526-5683-26, 27, 28, 29, 30, 32
48. Fan motor	Westinghouse	143T
49. Fan motor	Westinghouse	213T

[1]Limitorque valve actuators are purchased to various specifications, contain various motors, and reference several test reports. Qualification evaluation considered the above and the location in the plant.

APPENDIX D

SAFETY-RELATED SYSTEMS AND FUNCTIONS

Function	System
1. Emergency Reactor Shutdown	Nuclear Boiler System (1B21) Reactor Water Recirculation System (1B31) Control Rod Drive System (1C11) Reactor Protection System (1C71) Process Radiation Monitoring System (1D11) Residual Heat Removal System (1E11) Reactor Primary Containment (1T23) Auxiliary Power Systems[1] Battery and Distribution Systems[1]
2. Containment Isolation	Nuclear Boiler System (1B21) Reactor Water Recirculation System (1B31) Control Rod Drive System (1C11) Neutron Monitoring System (1C51) Reactor Protection System (1C71) Process Radiation Monitoring System (1D11) Residual Heat Removal System (1E11) Core Spray System (1E21) High Pressure Coolant Injection System (1E41) Reactor Core Isolation Cooling System (1E51) Radwaste System (1G11) Reactor Water Cleanup System (1G33) Fuel Pool Cooling and Cleanup System (1G41) RBSVS & Control Room Chilled Water System (1M50) Reactor Bldg. Closed Loop Cooling Water System (1P42) Compressed Air System (1P50) Diesel Emergency Power System (1R43) Reactor Primary Containment (1T23) Primary Containment Inerting System (1T24) Reactor Bldg. Standby Ventilation System (1T46) Primary Containment Atmospheric Control System (1T48) Miscellaneous HVAC (1X41) Diesel Generator Ventilation System (1X60) Control Room HVAC (1X61) Auxiliary Power Systems[1] Battery and Distribution Systems[1]
3. Reactor Core Cooling	Nuclear Boiler System (1B21) Reactor Water Recirculation System (1B31) Residual Heat Removal System (1E11)

Function	System
	Core Spray System (1E21) High Pressure Coolant Injection System (1E41) RBSVS & Control Room Chilled Water System (1M50) Demineralized and Make-Up Water System (1P21) Service Water System (1P41) Reactor Bldg. Closed Loop Cooling Water System (1P42) Compressed Air System (1P50) Diesel Emergency Power System (1R43) Reactor Primary Containment (1T23) Reactor Bldg. Standby Ventilation System (1T46) Diesel Generator Ventilation System (1X60) Miscellaneous HVAC (1X41) Control Room HVAC (1X61) Auxiliary Power Systems[1] Battery and Distribution Systems[1]
4. Containment Heat Removal	Nuclear Boiler System (1B21) Residual Heat Removal System (1E11) RBSVS & Control Room Cooling Water System (1M50) Service Water System (1P41) Diesel Emergency Power System (1R43) Reactor Primary Containment (1T23) Reactor Bldg. Standby Ventilation System (1T46) Miscellaneous HVAC (1X41) Control Room HVAC (1X61) Diesel Generator Ventilation System (1X60) Post Accident Monitoring System (1Z93) Auxiliary Power Systems[1] Battery and Distribution Systems[1]
5. Reactor Heat Removal	Nuclear Boiler System (1B21) Reactor Water Recirculation System (1B31) Residual Heat Removal System (1E11) Core Spray System (1E21) RBSVS & Control Room Cooling Water System (1M50) Service Water System (1P41) Reactor Bldg. Closed Loop Cooling Water System (1P42) Compressed Air System (1P50) Diesel Emergency Power System (1R43)

Function	System
	Reactor Primary Containment (1T23) Reactor Bldg. Standby Ventilation System (1T46) Miscellaneous HVAC (1X41) Diesel Generator Ventilation System (1X60) Control Room HVAC (1X61) Post Accident Monitoring System (1Z93) Auxiliary Power Systems[1] Battery and Distribution Systems[1]
6. Prevention of Release of Radioactive Material to the Environment	Nuclear Boiler System (1B21) Process Radiation Monitoring System (1D11) Area Radiation Monitoring System (1D21) Residual Heat Removal System (1E11) MSIV Leakage Control System (1E32) High Pressure Coolant Injection System (1E41) Reactor Core Isolation Cooling System (1E51) Radwaste System (1G11) Reactor Water Cleanup System (1G33) Fuel Pool Cooling and Cleanup System (1G41) RBSVS & Control Room Chilled Water System (1M50) Main Steam System (1N11) Miscellaneous Drains (1N23) Sample System (1P33) Service Water System (1P41) Reactor Bldg. Closed Loop Cooling Water System (1P42) Diesel Emergency Power System (1R43) Reactor Primary Containment (1T23) Primary Containment Inerting System (1T24) Reactor Bldg. Standby Ventilation System (1T46) Primary Containment Atmospheric Control System (1T48) Miscellaneous HVAC (1X41) Diesel Generator Ventilation System (1X60) Control Room HVAC (1X61) Auxiliary Power Systems[1] Battery Power and Distribution Systems[1]

Function	System
The following systems were not included in the Class 1E safety function listing because they do not perform any safety function during accident mitigation. Some of the systems are not safety related, but may have components which are connected to safety power.	
	Standby Liquid Control System (1C41) Remote Shutdown System (1C61) Condensate and Feedwater Systems (1N21) Condensate Transfer and Storage System (1P11) AC Uninterruptible Power (1R36) Reactor Bldg. Normal Ventilation System (1T41) Primary Containment Cooling System (1T47) Post Accident Sampling System (1Z96) Excess Flow Check Valves (1Z92)

[1]The Auxiliary Power and Battery Power and Distribution systems are comprised of the following. Not all of these systems are needed for each safety function.

Metal-Clad Switchgear (1R22)
Unit Substations (1R23)
Motor Control Centers (1R24)
Power Cable & Wire (1R31)
Control Cable & Wire (1R32)
AC Control and Instrument Power (1R35)
DC Instrument Power (1R41)
Battery Power (125V-DC) (1R42)

ATTACHMENT 3

SEP 16 1982

Docket No. 50-322

Long Island Lighting Company
ATTN: Mr. M. S. Pollock
Vice President - Nuclear
175 East Old Country Road
Hicksville, New York 11801

Gentlemen:

Subject: Inspection 50-322/82-23

This transmits the August 1 - September 8, 1982 routine resident safety inspection findings by Mr. J. C. Higgins and Mr. P. Hannes at the Shoreham Nuclear Power Station, Shoreham, New York. These findings were based on observations of activities, interviews, and document reviews, and have been discussed with Mr. J. Smith and other members of your staff.

No violations of NRC requirements were found. No reply is required.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1). The telephone notification of your intent to request withholding, or any request for an extension of the 10 day period which you believe necessary, should be made to the Supervisor, Files, Mail and Records, USNRC Region I, at (215) 337-5223.

Your cooperation is appreciated.

Sincerely,
Original Signed By:
T. J. Brunner for
Richard W. Starostecki, Director
Division of Project and Resident
Programs

Enclosure: NRC Region I Inspection Report
Number 50-322/82-23

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cc w/encl:
J. L. Smith, Manager of Special Projects
T. F. Gerecke, Manager, QA Department
Edward M. Barrett, Esquire
Jeffrey L. Futter, Esquire
J. Rivello, Plant Manager
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New York
Shoreham Hearing Service List

bcc w/encl:
Region I Docket Room (with concurrences)
Chief, Operational Support Section (w/o encl)
R. Gilbert, DOL, NRR
E. Weinkam, DOL, NRR

U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-322/82-23

Docket No. 50-322

License No. CPPR-95 Priority -- Category B

Licensee: Long Island Lighting Company

175 East Old Country Road

Hicksville, New York 11801

Facility Name: Shoreham Nuclear Power Station, Unit 1

Inspection at: Shoreham, New York

Inspection conducted: August 1 - September 8, 1982

Inspectors: JCHiggins 9/13/82
J. C. Higgins, Senior Resident Inspector date signed

JCHiggins FOR 9/13/82
P. H. Haynes, Resident Inspector date signed

date signed

Approved by: EJBrunner for 9/13/82
R. M. Gallo, Chief, Reactor Projects Section 1A date signed
Projects Branch #1, DPRP

Inspection Summary:

Inspections on: August 1 - September 8, 1982 (Inspection Report No. 50-322/82-23)
Areas Inspected: Routine onsite regular, backshift and weekend inspections by the resident inspectors (147 inspection hours) of work activities, preoperational testing, and plant staff activities including: tours of the facility, procedure review, test program implementation, review of NRC Bulletins, test results review, review of construction deficiency reports, review of Three Mile Island modifications, and followup on previously identified items.

Results: No violations were identified.

DETAILS

1. Persons Contacted

R. Gutman, Maintenance Engineer (L)
R. Jongeblood, Nuclear Engineer (L)
J. Kelly, Field QA Manager (L)
W. Matejek, Lead Advisory Engineer (S&W)
J. McCarthy, Section Supervisor - FQA (L)
M. Milligan, Project Engineer (L)
W. Museler, Manager, Construction and Engineering (L)
K. Nicholas, Lead Startup Engineer (GE)
R. Perra, Assistant Superintendent FQC (S&W)
R. Reen, Security Supervisor (L)
J. Ricardo, Lead Startup Engineer (S&W)
J. Riley, Operations Manager (GE)
J. Rivello, Plant Manager (L)
C. Seaman, Senior Asst. Project Engineer (L)
J. Smith, Manager, Special Projects (L)
R. Werner, OQA Engineer (L)
E. Youngling, Startup 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 including management, clerical, maintenance, operations, engineering, testing, health physics, security, quality assurance, and construction personnel.

2. Previous Inspection Item Update

2.1 Items Closed

2.1.1 (closed) Unresolved Item No. (322/80-09-01): Containment Systems Technical Specifications Items: The inspector reviewed the latest NRC draft of the Shoreham Technical Specifications and noted that section 3/4.6.2 contains a limiting condition for operation and surveillance requirements for the Drywell Floor Seal Pressurization System. Also in this same specification the 4.91 inches of water per minute acceptance criteria has been deleted and the wording now reads that leakage must be within the specified limit. This is acceptable since the acceptable limits are defined in correspondence between the licensee and the NRC relative to the Safety Evaluation Report open item number 26.

2.1.2 (closed) Violation No. (322/81-13-01): Startup Manual Control: This violation cited discrepancies associated with the updating of the various controlled copies of the Startup Manual. It was updated in inspection report 82-02 and then cited as a repeat violation in inspection report 82-08. The licensee stated that all manuals have been corrected and that the Administrative Aide to the Startup Manager is responsible for changes. All changes to controlled copies are to be made by Startup personnel. The inspector reviewed the following in selected copies of controlled Startup Manuals:

- Proper entry of Revision 17
- Proper entry of Revision 18
- Correct pages in place for selected earlier revisions
- Proper entry of the new Startup Instruction No. 9 into Appendix 4A of the Startup Manual
- Correct revisions of Startup Instructions No. 1 through No. 8 in the manuals

The inspector also reviewed the last two quarterly surveillances by Operational Quality Assurance (OQA) of the Startup Manual, No. 82-113 dated March 29, 1982 and No. 82-114 dated June 30, 1982. The inspector noted that the licensee had established adequate control of the Startup Manual and had no further questions. This violation is closed.

2.1.3 (closed) Unresolved Item No. (322/81-20-01): Hole in motor-operated valve (MOV) casing: The licensee issued Repair/Rework Request No. P42-170 to replace the damaged MOV motor casing for valve P42*MOV-036. This was completed June 14, 1982 and the valve was retested as appropriate. The inspector reviewed the documentation and observed the new motor installed on the MOV. This item is closed.

2.1.4 (closed) Violation No. (322/82-08-01): Document Control: This violation had two parts, control of the Startup Manual and use of an improperly approved procedure. The resolution of the Startup Manual control is discussed above under Violation No. (322/81-13-01). With respect to the procedure approval, the licensee initially took exception to the findings in letter SNRC-720 dated June 25, 1982; and then agreed to specified controls and approvals, which were documented in the letter from the NRC to LILCO dated August 9, 1982. In reviews performed during this inspection period, no examples were identified where appropriate procedures and approvals were not executed. Additionally, the Startup Manager has reviewed and approved the test results of the Inservice Reactor Pressure Boundary Leak Test performed between April 24 and May 3, 1982. This item is closed.

2.1.5 (closed) Violation No. (322/81-13-01): Startup Lifted Leads and Jumpers: The inspector observed the panels identified in this item and noted that all discrepancies were corrected. The inspector conducted tours throughout the plant during the inspection period and did not identify any unauthorized startup lifted leads or jumpers. The inspector also reviewed the Startup Jumper Tag File and the Jumper Log Book. No discrepancies were identified. The Startup organization has placed plastic insulating end-caps on spare leads in control room and relay room panels in place of the tape previously used in order to quickly and easily distinguish between spare leads and temporarily lifted leads. Discussions and document review indicate that program requirements have been re-emphasized to Startup Personnel. The inspector reviewed Operational Quality Assurance (OQA) 1982 documents associated with the lifted lead and jumper program including: the bi-monthly surveillances, corrective action requests, and LILCO deficiency reports (LDRs) and noted a decreasing number of identified problems. The inspector noted that the Startup response to two LDRs stated that particular lifted leads without tags were acceptable since the leads had not been lifted as part of a test procedure. The inspector did not agree with this narrow interpretation and discussed this with the Startup Manager, who agreed that any lifted lead or jumper in place overnight, should and would be tagged per the Startup Manual. Startup has initiated a weekly surveillance program to survey control room, relay room, and remote shutdown room panels for improper lifted leads and jumpers. The inspector reviewed the records of this program and noted that it appeared effective in reducing the number of identified problems. This violation is closed.

2.1.6 (closed) Unresolved Item No. (322/82-13-05): Annunciator Data List: The inspector reviewed the identified annunciators and a random selection of additional annunciators on the data list. All were appropriately yellow-lined. In addition the Startup Work Coordinator and the Startup technicians have been reinstructed in the proper updating of the annunciator data list. This item is closed.

2.2 <u>Items Remaining Open</u>

2.2.1 (open) Violation No. (322/81-14-01): Jumpers and Lifted Leads: This item was previously reviewed in report 81-20. Danger tagging and jurisdictional tagging were found acceptable then. No discrepancies have been identified in these areas during the current inspection either. The area of Startup jumpers and lifted leads was addressed above in paragraph 2.1.5 and was found acceptable. The remaining area is the plant staff jumper and lifted lead program. During the inspection 81-20 update discrepancies were identified in this program. During the current inspection, additional discrepancies were identified in this program as follows: Permits 81-11-4, 82-04-01, 82-08-03, and 82-08-06 had no expected duration; Permit 82-08-03 did not indicate the number of tags; several monthly audits performed since the issuance of the above permits did not identify the noted discrepancies; and in Panel WWP the lead to relay 3L-FW-VL was not tagged, three tags were attached to a terminal vice the lifted lead, and there were two lifted leads (the white and green/black leads of cable M42-NNC046) with no tags on the right side of the panel. This item remains open.

2.2.2 (open) Unresolved Item No. (322/80-04-04): Vendor Procedures: The licensee has revised the Startup Manual, paragraph 7.6.2 to require that the Joint Test Group review and approve any vendor procedure used for test activites. For the testing of the Carbon and HEPA filters the licensee has written and approved procedure CG.000.037 "In Place Testing of HEPA Filter and Carbon Adsorber Stage". The inspector reviewed the procedure against the FSAR, ANSI N-510, and Regulatory Guide (R.G.) 1.52, Rev. 2 and noted three minor discrepancies: (1) an incorrect reference on enclosure 1, Table 1 to step 8.5; (2) a missing equation on enclosure 6, page 2; and (3) the procedure does not require that air flow be continued after the charcoal adsorber test till residual gas effluent is less than 0.01 ppm per R.G. 1.52 paragraph C.5.d. This item remains open.

2.2.3 (open) Violation No. (322/82-13-04): Test Program: Item 1 - The inspector reviewed Checkout & Initial Operations (C&IO) package number E11-254A dated August 5, 1982, which performed the retests of relays E11*K45A and B and discussed the completeness of C&IO files with test engineers involved. No discrepancies were identified. Item 2 - The inspector reviewed C&IO package No. E11-254A which contained circuit wiring checks in accordance with E&DCR P-36309B. However, the inspector noted that as of September 7, 1982 the Yellow Line Master drawing for 1.61-136I had not yet been corrected. Item 3 - Not reviewed yet. This violation remains open.

3. Plant Tour

The inspector conducted periodic tours of accessible areas in the plant during normal, backshift, and week-end hours. During these tours, the following specific items were evaluated:

- Hot Work - Adequacy of fire prevention/protection measures used;
- Fire Equipment - Operability and evidence of periodic inspection of fire suppression equipment;
- Housekeeping - Maintenance of required cleanness levels of systems under or following testing;
- Equipment Preservation - Maintenance of special precautionary measures for installed equipment, as applicable;
- QA/QC surveillance - Pertinent construction and startup activities were being surveilled on a sampling basis by qualified QA/QC personnel;
- Security - Adequate site construction security;
- Weld Rod Control - Observations to determine weld rod was being controlled per site procedures; and
- Component Tagging - Implementation of appropriate equipment tagging for safety, equipment protection, and jurisdiction.

Minor discrepancies identified were brought to the licensee's attention and were corrected.

4. Three Mile Island (TMI) Modifications

4.1 Reactor Core Isolation Cooling (RCIC) Automatic Restart

This is item II.K.3.13 of NUREG-0737, "Clarification of TMI Action Plan Requirements". In letter SNRC-744 dated July 29, 1982 the licensee stated that this item was ready for confirmatory review. The inspector reviewed the completed RCIC Preoperational Test, PT.119.001-1, which verified the installation of the RCIC automatic restart modification. The inspector also reviewed station operating and alarm response procedures for the RCIC system and noted that: (1) SP.23.119.01, Rev. 0 and ARP 1062, Rev. 1 were not revised after the modification, and (2) ARP 1248, Rev. 0, Reactor High Water Level, does not mention the HPCI or RCIC turbine trips. Additionally, the inspector noted that prior to implementing this modification, all RCIC turbine trips gave the control room RCIC Turbine Tripped Alarm. This alarm scheme is still described in the FSAR, but has been degraded in the plant in that now a trip on high reactor water level (level 8) will not give the RCIC Turbine Tripped Alarm. The inspector stated that the intent of the TMI modification was not to remove existing alarm indications. This item remains open.

5. Containment Purge Valve Operability Test

In letter SNRC-636 to the NRC the licensee committed to the performance of an In-Situ operability test of a six inch containment purge valve. These valves are air-opened and spring-closed. This test was performed during the Structural Acceptance Test at a containment pressure of greater than 55 psig and demonstrated that the valve will close within 5 seconds as required by SER Open Item No. 36 and II.E.4.2. The inspector reviewed the test data, observed the valves in the plant (including the new debris screen at the inlet from the drywell), and discussed the test and valve arrangement with cognizant licensee personnel. During the test, difficulty was experienced with the opening of one purge valve located inside the drywell due to insufficient differential pressure. The test was performed satisfactorily using a valve physically located outside of the primary containment and it was determined that, had the inside valve been opened, it would close also. This is because both sides of the valve diaphragm are vented during closure and the spring is the only closing force. The inspector had no further questions in this area.

6. Construction Deficiency Reports

In letters to the NRC dated November 30, 1981 and July 23, 1982 the licensee described a reportable deficiency involving water hammer loads affecting the Control Rod Drive (CRD) System at Shoreham (Item No. 81-00-09). The inspector discussed various aspects of this problem with NRC regional management and with licensee representatives as follows. All design loads are to be specified by Stone & Webster (S&W), the licensee's Architect-Engineer. Also, the pipe stress analysis is to be performed by Stone & Webster. Pipe support design will be performed by Reactor Controls, Inc. (RCI) using the S&W supplied loads. Both the licensee and S&W have sent design/engineering personnel to RCI to review the design process to be utilized for the CRD pipe supports. Finally, the licensee has committed to performing an onsite design review of four CRD pipe supports: one modified support inside and one outside primary containment; and one unmodified support inside and one outside containment. This item will receive further review and remains open.

7. New Fuel Inspections

The inspector observed the uncrating of new fuel from the metal shipping containers, the inspection of the new fuel, and the dry storage of the fuel in the fuel pool area. During the witnessing the inspector noted that:

- License requirements were being met.
- Security measures were in place.
- Health physics controls were in place.
- Fire protection measures were being observed.
- Personnel were qualified.
- Inspections were performed per procedures and results were documented.
- Fuel assemblies were carefully handled.

A few discrepancies were identified by the new fuel inspectors which were typical of new fuel. These are scheduled for correction prior to use of the new fuel. Discussions with responsible personnel indicated that no mishaps had occurred during fuel handling and inspections. During the placement of the initial bundles in the fuel pool area, the inspector noted that the tag board already indicated the final position with all fuel assemblies in the pool. The licensee's representative agreed that the tag board should be maintained current as each fuel assembly was moved and changed the method of handling the tag board. Further inspections revealed no discrepancies. The inspector had no additional questions on inspection of new fuel at this time.

8. NRC Bulletins

Bulletin 80-04: This Bulletin, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition" involved an analysis discrepancy at a Pressurized Water Reactor (PWR). The licensee has reviewed this Bulletin and determined that it is not applicable to Shoreham, a Boiling Water Reactor. This Bulletin is closed.

9. Preoperational Test Program Implementation

During the inspection period the inspector observed portions of system and component testing, reviewed test documentation, held discussions with test engineers and startup management, observed equipment operation, reviewed the use and updating of controlled test procedures, and reviewed test organization and scheduling. No discrepancies were identified.

10. 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 inspector also attended selected entrance and exit interviews for four region-based inspections conducted during the inspection period.

Additionally, the inspector participated in the following meetings/reviews conducted with the licensee:

- Caseload Forecast Panel to review the licensee's current schedule.
- Meeting with LILCO President W. Uhl, et al. in Region I to discuss prerequisites for an operating license and other topics.
- NRR site visit to discuss post-accident discharge of radioactive material and secondary containment flooding.