UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	))))))	Docket No. 50-322-1 (OL)

AFFIDAVIT OF ROBERT G. LaGRANGE IN RESPONSE TO ALAB-788

I, Robert G. LaGrange, depose and say:

1. I am a Section Leader in the Equipment Qualification Branch, within the Division of Engineering, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission. A statement of my Professional Qualifications is attached. This affidavit is submitted in response to that portion of ALAB-788 dealing with "environmental qualification."

2. In ALAB-788 issued by the Atomic Safety and Licensing Appeal Board on October 31, 1984, the Appeal Board required the NRC Staff to advise the Licensing Board whether any non-safety related electrical equipment at Shoreham falls within the category defined by 10 CFR §50.49(b)(2) and, if so, the basis for the Staff's approval. (ALAB-788 at slip op. 105)

3. In compliance with the Appeal Board's requirements, the Board's attention is invited to Section 3.11.3 of the Shoreham SSER 7, issued in

September, 1984. In particular, Section 3.11 and specifically Section 3.11.3.1 of SSER 7, which was prepared under my supervision, and with which I concurred, describes the Staff's determination that no equipment at Shoreham falls into the category defined by 10 CFR §50.49(b)(2). Additionally, the basis for the Staff's determination in this regard is fully set forth in Section 3.11.3.1. In Section 3.11.3.1 we discussed the performance of a control systems failure study, a high energy line break/control system failure analysis, and the electrical isolation design philosophy at Shoreham. The staff has reviewed these areas and has found them to be acceptable as documented in Section 7.7 of SSER 4 and Section 7.6.6 of the SER. One of the purposes of these studies was to identify nonsafety-related equipment whose failure could affect the satisfactory accomplishment of safety functions by safety-related equipment. The resolution of these issues provides a sufficient basis to conclude that there is no equipment that falls into the category defined by 10 CFR §50.49(b)(2). I hereby certify (1) that the statements contained therein are true and correct to the best of my knowledge and belief, and (2) I know of no equipment at Shoreham that falls within the category of equipment described in 10 CFR § 50.49 (b)(2).

Robert G. LaGrange
Robert G. LaGrange

Subscribed and sworn to before me
this 13th day of November, 1984

Malinda L. McDonald
Notary Public

My commission expires: 7/1/86

PROFESSIONAL QUALIFICATIONS

OF

ROBERT G. LaGRANGE

I am Section Leader of the Environmental Qualification Section of the Equipment Qualification Branch, Division of Engineering, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission. I am responsible for planning, organizing and directing the activities of the section in performing technical reviews, analyses and evaluations of the adequacy of the environmental qualification of electrical and mechanical equipment whose failure, due to such environmental conditions as temperature, humidity, pressure and radiation, could adversely affect the performance of safety systems. I was previously a Senior Mechanical Engineer in the Seismic and Dynamic Loads Qualification Section of the Equipment Qualification Branch. My duties and responsibilities involved the review and evaluation of the structural integrity, operability and functional capability of safety related mechanical and electrical equipment under all normal, abnormal, and accident loading conditions, and in the event of seismic occurrences and other pertinent dynamic loads. Prior to my positions in the Equipment Qualification Branch, I was an Applied Mechanics Engineer in the Engineering Branch, Division of Operating Reactors. My duties and responsibilities included the review, analysis and evaluation of structural and mechanical aspects of safety issues related to reactor facilities licensed for power operation.

I have a B.S. degree in Mechanical Engineering from the University of Maryland (1972) and have done graduate work at both the University of Maryland and George Washington University.

Prior to my joining the NRC, I was associated with Bechtel Power Corporation as a Group Leader in the piping stress analysis group. My duties and responsibilities included performing and supervising stress analyses of nuclear power plant piping, and related activities, with emphasis on seismic analysis.

NUREG-0420

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1981

(1) Source range monitor
(2) Intermediate range monitor
(3) Local power range monitor
(4) Average power range monitor
(5) Rod block monitor
(6) Traversing in-core probe.

The applicant has documented in the Final Safety Analysis Report that the neutron monitor system is identical to previously licensed designs. However, we noted that the rod block monitor subsystem is a modified design and contains multiplexing circuitry which interfaces with the new reactor manual control system. The purpose of the rod block monitor subsystem is to supply an inhibit signal to the reactor manual control system to prevent control rod withdrawal if it is determined that a given rod withdrawal will exceed the minimum critical power ratio.

The multiplexing circuitry employed in the Shoreham rod block monitor and reactor manual control system processes and transmits information about reactor status, control rod position, rod block logic, and rod control logic through common electrical signals. In earlier BWR designs this was accomplished by individual circuits. The new design has a self testing capability to assure that this information is being processed correctly. We believe that the new multiplexing design is acceptable provided this self testing capability is formally implemented through technical specifications. The technical specifications will be reviewed to confirm that this is done. On this basis, we conclude that the rod block monitor design is acceptable.

7.6.5 Rod Sequence Control System

The rod sequence control system is a subsystem of the reactor manual control system. It receives inputs from the rod position information system and the rod drive control system, both of which are also subsystems of the reactor manual control system. The purpose of the rod sequence control system is to reduce the consequences of the postulated rod drop accident to an acceptable level by restricting the patterns of control rods that can be established to predetermined sets and to prevent a control rod withdrawal accident at low power.

We have noted in our discussions of the refueling interlocks and neutron monitoring system that the Shoreham reactor manual control system differs from earlier BWR designs in that it employs multiplexing as opposed to individual circuits (Sections 7.6.1 and 7.6.4). We also consider this design acceptable for the rod sequence control system function, on the basis that the self testing capability will be formally incorporated as a technical specification requirement.

7.6.6 Physical Independence

Section 3.12 of the Final Safety Analysis Report presents the separation criteria that have been followed to assure that the physical independence of redundant instrumentation, control and electrical equipment is maintained. The objectives of our review are to determine that the independence of redundant equipment is not jeopardized by a design basis event, such as a fire or missile, to the point that it may result in the loss of function. The basis for judging the acceptability of the applicant's separation criteria and design arrangement has been conformance with the acceptance criteria set forth in Section 7.1 of this report, in particular with IEEE Standard 384-1974, "Criteria for Separation of Class 1E

Equipment and Circuits," as augmented by Regulatory Guide 1.75, "Physical Independence of Electrical Systems," (Revision 1). The following items address the problem areas revealed during our review and the resolutions concerning them.

(1) The motor-generator sets that provide power to the reactor trip system are located in the relay room which contains the cables, instrumentation and control cabinets of the plant protection system. We informed the applicant that we were concerned that a failure of the motor-generator sets including their flywheel could lead to the generation of missiles which may strike the protective equipment cabinets. This could render plant protective channels inoperative. The applicant has documented the results of analysis which show that this rotating equipment is not a credible source of missiles. Furthermore, missile protection capability is inherently built in the design by having the motor-generator sets axis perpendicular to the cabinets. We consider the resolution of this issue acceptable.

(2) Recent reviews of designs similar to those in Shoreham revealed that a General Electric supplied computer system known as "Startrek" will be used to monitor plant start-up testing. It was noted that both safety and non-safety inputs will be connected to the computer and this may compromise the independence of Class 1E redundant divisions. This concern has been satisfactorily resolved in LaSalle which has the same design as Shoreham. We consider the Shoreham "Startrek" system acceptable with regard to isolation of the protection systems from the Startrek system.

(3) We noted that non-Class 1E motor space heater wiring was included in the Class 1E 4,160 volts switchgear breaker cubicles and in direct-current motor control centers. We expressed the concern that the Class 1E circuits may be degraded below an acceptable level as the result of a failure in the non-Class 1E wiring. We required that the applicant either demonstrate that such a wiring arrangement will not degrade the Class 1E circuits below an acceptable level, or modify the design to satisfy IEEE Standard 384-1974 as supplemented by Regulatory Guide 1.75 (Revision 1).

The applicant has stated that all wire used within the Class 1E 4,160 volts switchgear and direct-current motor control centers (whether it is associated with safety or non-safety functions) is qualified in accordance with Class 1E requirements. Current to the motor space heater is only applied when the breaker or starter is open (i.e., the motor is not running) and it is directed to the heater through an auxiliary contact of a Class 1E relay that changes state whenever the motor starts or stops. These auxiliary contacts are located in the switchgear and motor control centers and serve as isolation devices between non-Class 1E and Class 1E wiring. Each auxiliary contact is separated from other contacts in the same relay by dielectric barriers.

The removal of the non-Class 1E source of power to the space heater when the motor is running has eliminated the possibility of failures which otherwise may have challenged the capability of the Class 1E motor to perform its intended safety function when required. The applicant has agreed, at our request, to incorporate in the design the capability for monitoring the removal of power from each motor space heater which will be verified during the periodic testing of the Class 1E motor.

Based on our review of the provisions of the design discussed above and the low energy sources involved, it is our judgement that the non-Class 1E motor space heater wiring will not degrade the Class 1E circuits. We conclude that this is acceptable subject to the implementation of the power removal monitoring feature for the motor space heaters. (See Section 8.4.10)

On the basis of our review, we have concluded that the physical independence of redundant instrumentation, control and electrical equipment satisfy our requirements stated previously in this section, with the exception of the fire hazard analysis review.

7.6.7 High Pressure and Low Pressure System Interfaces

Low pressure systems that are connected to high pressure systems must be isolated when the design pressure of the low pressure system is exceeded. The isolation of the low pressure system under high pressure conditions is necessary to avoid damage by overpressurization or the potential for loss of integrity of the low pressure system and possible radioactive releases.

We reviewed the points of interface between high pressure and low pressure systems identified by the applicant and concluded that they satisfy our position set forth in Branch Technical Position ICSB 18 (PSB), "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," and, therefore, the design of the instrumentation, control and electrical equipment pertaining to these points of interface is acceptable.

7.6.8 Main Steam Isolation Valve Leakage Control System

The purpose of the main steam isolation valve leakage control system is to control and minimize the release of fission products, which could leak through the closed main steam isolation valves after a loss-of-coolant accident. This is accomplished by directing the leakage through a bleed line into an area served by the reactor building standby ventilation system for processing prior to release to the atmosphere.

The design of the leakage control system for the main steam isolation valves provides for two independent subsystems (upstream and downstream from outboard main steam isolation valves) powered from separate electrical divisions. The upstream and downstream subsystems each connect to the main steam lines through two isolation valves in series.

We noted that the serial isolation valves in either subsystem were powered from the same electrical division and thus, vulnerable to single failures which can result in the opening of both isolation valves during normal plant operation when the main steam isolation valves are open. Our review of a design similar to that of Shoreham revealed that a single failure of a relay will cause the opening of both serial isolation valves. This problem has been corrected at Shoreham by modifying the design to include additional relays.

7.6.9 Containment Vacuum Relief Valves Indication

We have proposed that redundant position indication be provided for all vacuum relief valves with indication and valve not closed alarm located in the main control room. This position indication system will inform the operator when

NUREG-0420
Supplement No. 4

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1983

7 INSTRUMENTATION AND CONTROLS

7.4 Systems Required for Safe Shutdown

7.4.3 Remote Shutdown System

On the basis of its review of the information furnished by the applicant regarding the remote shutdown panel (RSP) as reported in Section 7.4.3 of SSER 3, the NRC staff found that the design of the RSP would meet the regulatory requirements specified in GDC 19 and the guidance as detailed in the SRP Sections 7.4 II and III. As a confirmatory item, the staff required the applicant to provide final operating procedures and Technical Specifications and also perform a system operational verification test of the RSP with the assumption of the most limiting single failure in the equipment train controlled from the RSP or remote stations away from the RSP.

In a letter dated June 21, 1983, from J. L. Smith to Harold R. Denton, the applicant committed to (1) conduct a walk-through prior to fuel load to demonstrate RSP system operability (including stations remote from the RSP) with the assumption of the most limiting single failure, (2) revise the operating procedures prior to exceeding 5% power to reflect the final design of the RSP and its remote stations, and (3) address the RSP and its remote stations in the Shoreham Technical Specifications.

The staff has concluded that the above commitments are acceptable and that this confirmatory item is resolved.

However, the staff will condition the Shoreham license to require the applicant to (1) implement (and document) all of the required design changes discussed in Section 7.4.3 of SSER 3 by the end of the first refueling and (2) perform an acceptable procedure verification test for the new RSP design at that time.

7.5 Safety-Related Display Instrumentation

In SER Section 7.5, the NRC staff requested that the applicant review the adequacy of emergency operational procedures used by control room operators to attain safe shutdown upon loss of any Class 1E or non-Class 1E buses supplying power to safety- or nonsafety-related instruments and to control systems. The response to this request addressed Items 1 and 3 of IEB 79-27 regarding plant system design features. Based on a protection sequence for shutdown developed for Shoreham, the applicant demonstrated that only Class 1E systems are necessary to achieve cold shutdown and, therefore, enough equipment would remain available after the loss of any Class 1E or non-Class 1E electrical bus. This conclusion was accepted by the NRC staff in SSER 1.

In addition, the applicant committed to conduct a failure mode effects analysis of plant electrical buses and to determine whether emergency operating procedures are adequate for dealing with the resultant plant conditions. The applicant submitted the results of this analysis in letter, SNRC-761, dated August 27, 1982. The analysis demonstrated that failures or malfunctions of power sources or sensors providing power or signals to two or more control systems will not result in consequences outside the bounds of the FSAR Chapter 15 accident analysis. Plant personnel used the bus tree and load tables developed in the control systems failure analysis to verify that the plant operating procedures were adequate to deal with the identified transients. No procedure changes were required.

Finally, the applicant committed to review the Shoreham alarm response procedures for loss of power to Class 1E buses to ensure that these procedures identify the indications and symptoms resulting from postulated power failures on 4-kV, 480-V, and 125-V dc buses. The applicant concluded that the station operating procedures were adequate to address loss of power conditions on any Class 1E bus. CILAR B79-27 was logged closed for this item on September 29, 1982.

An NRC inspector reviewed the station normal operating, abnormal operating, and alarm response procedures for the 4-kV, 480-V, 125-V dc, 120-V ac instrument, and 120-V ac uninterruptible power supply systems as part of an inspection documented in Inspection Report No. 50-322/83-02. This review determined that the procedures provide sufficient, detailed instructions for the operator to: (1) identify the alarms, indicators, and symptoms needed to diagnose a loss of bus power; (2) restore bus power; and (3) identify alternate indications that may be used for plant control.

The NRC staff, therefore considers this item to be resolved.

7.6 Other Instrumentation and Control Systems Required for Safety

7.6.6 Physical Independence

7.6.6.1 Physical Independence Within NSSS Cabinets

During the NRC staff preparations for the Shoreham hearings, a concern developed regarding the lack of physical separation between non-Class 1E and Class 1E circuits inside the NSSS cabinets at Shoreham. It appeared to the NRC staff that the design of the Shoreham electrical system failed to provide adequate physical independence of circuits inside the NSSS cabinets, as established in current regulatory practice.

Section 4.6 of Standard 279-1971 of the Institute of Electrical and Electronics Engineers (IEEE), "Criteria for Protection Systems for Nuclear Power Generating Stations," requires, in part, that channels that provide signals for the same protective functions be independent and physically separated. Regulatory Guide (RG) 1.75, "Physical Independence of Electric Systems," describes a method acceptable to the NRC staff for complying with IEEE 279-1971 with respect to physical independence of the circuits and electrical equipment comprising or associated with the Class 1E power system, the protection system, systems actuated or controlled by the protection systems, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions.

In addition, in accordance with Section 4.6 of IEEE 384-1974, "IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits" (endorsed by RG 1.75), Class 1E circuits should not be degraded below an acceptable level by non-Class 1E circuits.

As a result of this regulatory guidance, the NRC staff's position was that failures in the non-Class 1E circuits should not prevent the proper functioning of Class 1E circuits or devices. Accordingly, the applicant was required to demonstrate that the separation inside the NSSS cabinets was adequate by using one of the following options:

(1) Correct the deficiency by meeting the separation criteria of RG 1.75, Revision 2, dated September 1978.

(2) Correct the deficiency by installing a separate barrier.

(3) Justify the deficiency by performing a specific analysis for each circuit where the minimum separation is not met to demonstrate that a failure will not propagate because of the insufficient separation.

In a letter dated August 31, 1982, from J. L. Smith to Harold R. Denton, the applicant transmitted an analysis that purported to provide justification for the lack of adequate separation inside the NSSS cabinets. The NRC staff evaluation of this analysis led to the conclusion that it was unacceptable and that, as previously stated, the applicant was required to demonstrate that the physical separation is adequate by using one of the three options listed above.

By letter dated January 21, 1983, from J. L. Smith to Harold R. Denton, the applicant provided a discussion regarding the separation of Class 1E and non-Class 1E circuits inside the NSSS cabinets and committed to utilize one of the following methods to resolve this concern:

(1) Provide adequate physical separation or barriers between the nonessential and essential circuits in these panels.

(2) Install redundant circuit protection for non-Class 1E circuits that are in close proximity to Class 1E circuits in the NSSS panels.

(3) Evaluate the non-Class 1E circuits and demonstrate that these circuits will not expend sufficient energy under fault conditions to damage adjacent essential circuitry.

To accomplish this, the applicant has committed to review all non-Class 1E cables/circuits internal to Shoreham's NSSS panels and evaluate whether they could impair the safety function of the Class 1E cables/circuits in these panels. Cables/circuits associated with annunciators and computer points are low energy (<5 watts) and will be eliminated from this evaluation because these circuits are not capable of expending energy of the magnitude necessary to damage cable insulation.

Also, to reduce the number of circuits requiring evaluation, the applicant has committed to install redundant circuit protection for all utility outlet and panel lighting circuits.

The applicant's preliminary evaluation of the remaining circuits indicates that they fall into one of the following categories:

(1) Nonessential instrument circuits that are not capable of generating energy of the magnitude necessary to damage cable insulation even under short circuit conditions. Examples of this type of circuit are 4-20 ma, 10-50 ma, 1-5 V circuits, thermocouple extension leads, and resistance temperature detection (RTD) circuits.

(2) Low ampacity nonessential control power circuits not capable of delivering sufficient current to exceed the ampacity rating of the cable used in the circuit, even when assuming a worst fault circuit condition and failure of a protective device.

(3) Nonessential circuits that meet the requirements of IEEE 384-1974 for associated circuits.

(4) Nonessential circuits that already have redundant circuit protection.

(5) Nonessential circuits that already have adequate physical separation or barriers provided.

(6) Nonessential 125-V dc circuits that have redundant circuit protection.

The applicant stated that the final review of all circuits would be completed by February 15, 1983. Any remaining non-Class 1E circuits that are not represented by one of the above categories will be provided with redundant circuit protection, adequate physical separation, or barriers to ensure the safety of any adjacent Class 1E circuits. If any of these modifications are required, the applicant has committed to intiate them during the first suitable outage, with the completion no later than the first refueling outage. This is considered acceptable because of the low probability of a non-Class 1E circuit failure degrading a Class 1E circuit or device during the first cycle of plant operation. These non-Class 1E circuits currently utilize appropriately sized circuit protection, and all NSSS cabinets are provided with fast-acting ionization-style fire detectors.

By letter dated April 7, 1983 (SNRC-869), the applicant confirmed that (1) all of the redundant protective devices utilized in the non-Class 1E circuits in the above analysis will be appropriately sized and located for the worst fault condition postulated and (2) the worst case condition postulated will encompass (a) open circuits, (b) short circuits, (c) transient conditions including electromagnetic interference (EMI), and (d) the application of the maximum credible fault voltage line to line and line to ground.

The staff has reviewed the Shoreham FSAR (Section 3.12), the applicant's responses to questions 223.12 (Revision 3 FSAR) and 223.67 (Revision 9 FSAR) and the additional information provided in the two letters noted above. Based on its review of the separation criteria utilized in the NSSS panels, the staff concludes that the design is acceptable, upon completion of the applicant's review and design modifications, and will meet the requirements of 10 CFR 50.55 a(h) (i.e., the requirements of IEEE 279).

The NRC staff will condition the license to require that the applicant implement all of the required design modifications by the end of the first refueling.

7.6.6.2 Electrical Separation Barriers

Deficiencies in separation for Shoreham electrical cables and raceways were identified in IE Inspection Report 50-322/79-07 and subsequent reports. As a result, the NRC staff required each deficiency to be corrected using one of the following four options:

(1) Correct the deficiency by meeting the electrical equipment separation criteria set forth in FSAR Section 3.12.

(2) Correct the deficiency by meeting RG 1.75, Revision 2, dated September 1978.

(3) Correct the deficiency by installing an acceptable barrier.

(4) Justify the deficiency by performing a specific analysis for each cable or raceway where the minimum separation is not met to demonstrate that a failure will not propagate because of the insufficient separation.

With regard to Option 3, the applicant, by letter dated January 14, 1983, provided its definition and basis (substantiated by test) for what constitutes an acceptable barrier. The applicant defined an acceptable barrier as a single conduit, tray cover, or wrapping of Siltemp woven-ceramic tape with 3M Scotch Branch No. 69 glass tape.

The NRC staff has reviewed Wyle Test Report No. 46287, "Test Report on Thermal Barrier and Short Circuit Test on 600 VAC Power and 120 VAC Control Cables," and Engineering and Design Coordination Report F-41238K, which describes the separation guidelines to be used for the installation of barriers at Shoreham. Based on the NRC staff's review of these reports, on discussions with the applicant, and on the conservatism of the proposed design, the NRC staff concludes that the applicant's definition of an acceptable barrier meets the objectives of IEEE 384-1974, as augmented by RG 1.75, and meets the independence requirements of GDC 17. It is, therefore, acceptable.

7.7 <u>Control Systems Not Required for Safety</u>

7.7.1 High-Energy Line Breaks (IE Bulletin 79-22, "Qualification of Control System")

If control systems are exposed to the environment resulting from the rupture of reactor coolant lines, steamlines, or feedwater lines, the control systems may malfunction in a manner that would cause consequences to be more severe than assumed in safety analyses.

The NRC staff requested the applicant to perform a review to determine what, if any, design changes or operator actions would be necessary to ensure that high-energy line breaks (HELBs) would not cause control system malfunctions and complicate the event beyond the FSAR analysis. In response to this concern, the applicant initiated a review to determine whether HELBs could have an effect on multiple control systems and to investigate the impact of failure of the applicable systems on the FSAR Chapter 15 analysis.

By letter dated November 8, 1982, from J. L. Smith to H. R. Denton (NRC), the applicant provided a report that presented the results of a design review, evaluation and plant walkdown addressing this concern.

The procedure that the applicant followed to perform the HELB analysis is as follows. The applicant

(1) Identified nonsafety-related control systems and components within these systems that may impact reactor pressure, water level, or critical power ratio and that may be vulnerable to functional damage from HELBs

(2) Established the assumptions and resulting criteria for high-energy line determination, break postulation, and consequence evaluation.

(3) Identified the locations (elevations/areas) that contain high-energy piping systems and in which components for the nonsafety-related control systems are located.

(4) Conducted a walkdown of the areas to verify the location of nonsafety-related control components and determined their proximity to high-energy lines.

(5) Postulated breaks in the areas having components from one or more of these nonsafety-related control systems and determined the resultant effect on the components, and ultimately the controlled equipment. Areas having no multiple system interactions within the constraints of the above criteria were not considered.

(6) Determined the resultant state of the reactor as a result of simultaneous failure of these nonsafety-related control systems.

(7) Compared this to events already analyzed and reported in FSAR Chapter 15, and determined if they are bounded. If not bounded, additional analysis was performed to determine if the effects are significant.

(8) Identified HELB/nonsafety-related control system events that were determined to be significant based on this analysis and indicated the corrective action to be taken.

The applicant performed the HELB study using the guidelines noted above. The results of the study indicated that all postulated events satisfy the criteria for infrequent events, i.e., that the dose consequences do not exceed 10% of the 10 CFR 100 criteria.

The most limiting event was found to be the loss of feedwater heating exacerbated by a turbine trip. This condition could be caused by a pipe break within the turbine building, which may simultaneously cause a partial loss of feedwater heating and a turbine trip, if the appropriate controls are disabled, leading to improper valve positions.

The loss of feedwater heating would cause a gradual increase in reactor power level which, without operator action, could eventually lead to a reactor trip at the APRM trip setpoint (117% power). Depending upon the specific timing of the event, the turbine trip may occur at a reactor power elevated between the operating value and the trip level of 117%.

When the turbine trip takes place, the bypass valves would start dumping 25% of the main steam flow to the condenser until the condenser pressure reaches 22.5 inches Hga. At this time, the bypass would also trip shut automatically. The bypass would be in operation for approximately 7 seconds after the turbine trip.

The staff was concerned about the consequences of an assumed worst case single failure concurrent with any of the postulated HELB events being more severe than those of the FSAR Chapter 15 analyses. The applicant provided the results of an analysis using the single failure assumption for the postulated worst case scenarios. The worst case postulated single failure analyzed was the complete loss of the turbine bypass system concurrent with the most limiting event noted above. This analysis shows that the results are well within the criteria for infrequent events. The applicant stated in a letter dated August 2, 1983 (from J. L. Smith to Harold R. Denton) that the dose consequences for this worst-case event will not exceed a small fraction (<10%) of the 10 CFR 100 criteria.

The staff questioned the applicant regarding the effects of humidity, pressure, and temperature on system components as a result of the HELB. In a letter dated May 11, 1983, from J. L. Smith to Harold R. Denton, the applicant stated that the effects of humidity, pressure, and temperature on the operability of these nonsafety-related control systems was addressed in formulating the conclusions reached in the original report. For additional clarification, the applicant stated that for small confined zones, it was assumed that any HELB would affect all nonsafety-related control components within the zone. Using this approach, it is apparent that the environmental effects on these components are directly enveloped within the scope of the report. In large, more open zones, only the components within the range of high-energy lines were assumed to fail simultaneously with the pipe break. Environmental effects on components outside the range of these HELB large open areas would tend to develop relatively slowly in comparison to the dynamic effects that would lead to rapid automatic and operator-initiated mitigative actions.

Based on its review and the conclusions of the applicant's study that indicate that the dose consequences will not exceed 10% of 10 CFR 100 criteria, the staff finds that SER Open Item 48, "High Energy Line Breaks," is resolved.

7.7.2 Multiple Control System Failures

SSER 3 noted that the applicant had committed to conduct a review to identify any power sources or sensors that provide power or signals to two or more control systems and to demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences beyond the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

By letter dated August 27, 1982, the applicant submitted a control systems failures evaluation report. The review performed for this report used the event-consequence logic of the Chapter 15 analyses, but started the logic chain from the specific source (i.e., a single bus failure) rather than a system condition.

This approach uncovered previously unanalyzed interactions. Although these new transient category events were postulated as a result of this study, it was concluded that the net effects were less severe than those of the original FSAR Chapter 15 events. The results of this report demonstrated that the previously reported limits of minimum critical power ratio, peak vessel, and main steamline pressures, and peak fuel cladding temperature for the expected operational occurrence category of events would not be exceeded as a result of common power source or sensor failures.

However, the staff remained concerned about control system malfunctions caused by a single failure of common hydraulic headers or impulse lines. The applicant submitted a report (letter dated June 20, 1983, from J. L. Smith to Harold R. Denton) addressing this issue. This report, supplemented by the existing FSAR Chapter 15 transient analysis, documents an evaluation of the Shoreham design related to postulated common sensor line failures (i.e., common hydraulic headers, impulse lines). Failures of common hydraulic headers, sensor taps, and instrument lines feeding two or more control system inputs were identified. Failure modes (broken or plugged lines were postulated for 28 individual identifications) and the resulting effects were analyzed.

All of the consequences of common instrument line failures were bounded by the previous analyses presented in FSAR Chapter 15, with the exception of a broken or plugged instrument standpipe on the feedwater heaters, which would reduce the feedwater temperature going into the reactor vessel and result in a possible turbine trip. Subsequent evaluation of this event indicates that the consequences are, in fact, bounded by the events considered in the Chapter 15 analyses.

The staff has reviewed the bases and results for the applicant's study and concludes, with reasonable assurance, that the consequences of single failures within the control systems are bounded by the analyses in FSAR Chapter 15. Therefore, the staff has concluded that SER Open Item 47, "Multiple Control System Failures," is resolved.

NUREG-0420
Supplement No. 7

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1984

cabinets and internals, auxiliary pump skids, and SDV vent and drain valves before the plant exceeds 5% power operation. The applicant must also complete the qualification for SDV solenoid valves before full power range testing during the power ascension program. Finally, qualification of the invessel rack must be complete before the first refueling outage.

The applicant will continue to provide a monthly updated equipment qualification summary list until this equipment has been qualified.

3.11 Environmental Qualification of Electrical and Mechanical Equipment

3.11.1 Background

SSER 3 identified several issues relating to justifications for interim operation with equipment that is not fully qualified and to qualification of the GE 200 series electrical penetrations that required resolution before an operating license is issued. On February 22, 1983, a new rule, 10 CFR 50.49, became effective that defined requirements for the environmental qualification of electrical equipment important to safety; this rule imposed several new requirements that applicants must address before licensing. The following paragraphs describe the staff evaluation of the applicant's responses to these outstanding items and to the new rule, and describe the staff's bases for concluding that the applicant has demonstrated conformance with 10 CFR 50.49.

3.11.2 Outstanding Items from SSER 3

3.11.2.1 Justification for Interim Operation

SSER 3 identified a number of open items relating to the justifications for interim operation (JIOs) with equipment that is not fully qualified. Many of these were requests for backup documentation used to support statements made in the JIOs or other minor clarifications. These have been resolved as a result of information in a letter from the applicant dated February 18, 1983 (SNRC-838), with the exception of the Anaconda flex conduit.

The applicant indicated that this item had been "successfully tested to the applicable service conditions." In a meeting with the applicant on July 29, 1983, the staff reviewed the qualification file for this item. Although a test report was available, the test was inadequate because only the electrical continuity of an assembly consisting of a junction box, conduit, and terminal blocks was measured during exposure to steam. The insulation resistance of the assembly, which could be reduced to unacceptable values for some instruments by failure of the plastic sleeve on the flexible conduit, was not measured. The applicant had performed additional analysis to demonstrate that the conduit construction is adequate for preventing moisture intrusion during a pipe break outside containment. The staff finds this acceptable only for justifying interim operation until additional type testing can be completed.

The applicant's original justification for interim operation was unacceptable because nonconservative handbook temperature ratings for the plastic sleeve of the conduit were used. As a result, the staff required that the applicant review all JIOs to determine if similar practices were utilized on other equipment items. The few cases where this method was utilized were found to be acceptable by the applicant and were so verified by the staff.

3.11.2.2 Interim Operation

The staff also requested that the applicant define the periods of interim operation with mechanical equipment not fully qualified, as identified in a letter dated November 19, 1982. In SNRC-838 dated February 18, 1983, the applicant indicated that full qualification would be accomplished by the end of the first refueling outage. The staff finds this schedule acceptable.

3.11.2.3 GE Series 200 Penetrations

The staff identified two outstanding items relating to the qualification of the GE series 200 electrical penetrations. The applicant addressed these items in a letter dated January 21, 1983 (SNRC-821) as follows:

- Surveillance testing: The staff requested that the applicant commit to a program for periodically monitoring the electrical integrity of these penetrations so significant age-related degradation can be detected and appropriate corrective action taken before failures occur. In SNRC-821, the applicant described an acceptable program to be utilized for this purpose.

- I^2R heating: The applicant provided information to show that the I^2R heating during qualification testing was greater than the heating effect that could be experienced in service. The response is acceptable.

3.11.3 Conformance with 10 CFR 50.49

10 CFR 50.49 contains several provisions not previously addressed by the applicant in the NUREG-0588 qualification program. In letters dated June 24, August 3 and 15, and September 9, 1983, the applicant discussed the effect of the rule on the existing environmental qualification program. The staff evaluated this response for those areas where a change to the program could occur. The staff's evaluation follows.

3.11.3.1 Scope of Equipment

10 CFR 50.49(b) and (c) define the scope of equipment to be included in the environmental qualification program. 10 CFR 50.49(c) limits the scope of equipment to that located in the harsh environments produced by design-basis events (DBEs) that is, therefore, susceptible to common mode failures.

Thus, a large portion of the electrical equipment important to safety is not covered by the rule and is not evaluated in this report. Conformance with existing requirements--such as the General Design Criteria (GDC, in Appendix A to 10 CFR 50), Appendix B to 10 CFR 50 (particularly Section III, "Design Control") and Regulatory Guide (RG) 1.33, Revision 2 ("Quality Assurance Program Requirements (Operation)") and other regulatory guides--is sufficient to ensure that electrical equipment located in mild environments performs adequately. The staff evaluation of this equipment is a part of the overall evaluation performed in accordance with the Standard Review plan (SRP, NUREG-0800).

10 CFR 50.49(b)(1) requires that safety-related equipment* be included in the program. The definition of safety-related is consistent with that used in the environmental qualification program.

Safety-related equipment that is not required to function to mitigate an event that produces a harsh environment need not be qualified for that harsh environment, as stated and implied in 10 CFR 59.49(d)(1), (e)(1), and (e)(4), provided that failure of that equipment has no impact on plant safety. This requirement agrees with that defined in the equipment classifications of NUREG-0588, Appendix E, Items 2a, 2b, and 2c. These classifications were used in the development of the Shoreham environmental qualification program, with the exception of a broader scope of DBEs to be evaluated, as discussed later in this report.

10 CFR 50.49(b)(2) requires qualification of nonsafety-related equipment whose failure could prevent the satisfactory accomplishment of safety functions by the safety-related equipment. The applicant has indicated that no Shoreham equipment is in this category. The applicant has referenced the control systems failure study, the high energy line break/control system failure analysis, and the electrical isolation design philosophy at Shoreham, which comply with RG 1.75, Revision 1.

The review of the first two areas is discussed in SER Section 7.7. The staff review has now been completed, and all issues have been satisfactorily resolved.

Position C.4 RG 1.75, Revision 1 states

> Associated circuits installed in accordance with Section 4.5.1 [of IEEE Standard 384-1974] should be subject to all requirements placed on Class 1E circuits such as cable derating, <u>environmental qualification</u> (emphasis added), flame retardance, splicing restrictions, and raceway fill unless it can be domonstrated that the absence of such requirements could not significantly reduce the availability of Class 1E circuits.

Associated circuits are defined as non-Class 1E circuits (i.e., nonsafety-related circuits) that share power supplies, enclosures, etc., with Class 1E circuits or that are not physically separated from Class 1E circuits. Other non-Class 1E circuits are not connected to Class 1E power supplies or are electrically isolated from Class 1E supplies to prevent malfunctions in one section of a circuit from causing unacceptable influences in other sections of the circuit.

The staff finds that conformance with this standard is sufficient to demonstrate compliance with 10 CFR 59.49(b)(2). Other interactions between safety-related and nonsafety-related equipment are covered in parts of the SRP, including Sections 3.5.1, 3.5.2 (missiles), 9.5.1 (fires), and 3.6.1 (pipe breaks).

Operating plants licensed in accordance with safety classification criteria less definitive than those applied to recently licensed plants may contain improperly classified equipment that would be covered by 10 CFR 50.49(b)(2). However, the staff review of the classification of structures, systems, and

---

*Safety-related equipment is defined as equipment that is relied on to remain functional during and following design-basis events to ensure certain safety functions.

components in Section 3.2.1 of the Shoreham Final Safety Analysis Report (FSAR) provides reasonable assurance that the equipment at Shoreham has been classified using the proper criteria.

The last type of equipment to be included in the environmental qualification program is the Category 1 and 2 instrumentation addressed in RG 1.97, Revision 2. The applicant has identified installed equipment in this category and provided justifications for interim operation with unqualified equipment. The staff has reviewed the identified items in the same way that other equipment in the program has been identified.

3.11.3.2 Scope of Design-Basis Events

10 CFR 50.49 requires that equipment be qualified for DBEs that produce a harsh environment, subject to certain limitations specified in 10 CFR 50.49(c). In accordance with Commission directives, the applicant based the Shoreham program on LOCAs and pipe breaks inside and outside containment only. The applicant also has reviewed additional events and their impact on the program, and described the results to the staff. Some events create environments that are different from normal plant operating conditions but that are not "significantly more severe" than the normal environment. Qualification in accordance with the new rule is not required because a harsh environment is not created. One event, control rod drop, results in a 6-month integrated gamma dose in the steam tunnel of 3.4×10^5 rems. Equipment required to mitigate this event and achieve shutdown is either (1) included in the applicant's existing environmental qualification program with operability required at significantly high radiation levels, or (2) located in a mild environment.

On the basis of its review, the staff does not require the applicant to change the harsh environment qualification program.

Instrument line breaks in the secondary containment have also been considered as a result of the rule, but these are enveloped by the breaks postulated in FSAR Appendix 3C.

3.11.3.3 List of Equipment

10 CFR 50.49(d) directs applicants to prepare a list of equipment covered by the rule. The applicant provided this list to the staff, and the latest revision (in the applicant's June 27, 1983 letter (SNRC-917)) is acceptable.

3.11.3.4 Completion of Qualification

Previous staff evaluations indicated that a license condition would be imposed requiring full qualification by the end of the first refueling outage. However, because 10 CFR 50.49(g) does not specify schedule requirements for holders of operating licenses, the following license condition will be imposed on the applicant and will supersede the previous commitment:

> The applicant shall environmentally qualify all electrical equipment within the scope of 10 CFR 50.49 in accordance with the implementation requirements of 10 CFR 50.49(g).

All other requirements in the rule are bounded by the existing qualification program. The staff, therefore, finds that the applicant conforms with 10 CFR 50.49.

3.12 Reactor Building Internal Flooding

The staff has completed its review of the internal flooding analysis in the Shoreham probabilistic risk assessment (PRA) study and the Shoreham flooding submittal dated December 2, 1982.* The applicant had found the Shoreham core vulnerable frequency initiated by flooding to be about 4×10^{-6} per reactor-year.

For the most part, the staff found the assumptions and methodology used by the applicant to be reasonable. However, in its review, the staff used more recent licensee event report (LER) data and used a different model in re-evaluating the flood-initiating frequency. The staff model used a Markov process model to determine the frequency of flood precursor events, and used time-phased event trees to account for the effects of flooding to different levels.

The staff recognizes that there are many uncertainties in the analysis, particularly the human error in initiating a flood and in not taking proper corrective actions during a flood. Therefore, the staff has performed an uncertainty analysis using the SAMPLE program (NUREG-75/014). The staff estimates that the mean value of the core vulnerable frequency of accidents initiated by flooding in the reactor building at Shoreham is 2×10^{-5} per reactor-year, and the 95% upper limit is 7.5×10^{-5} per reactor-year. The core vulnerable frequency as a result of maintenance-induced flooding has a mean value of 7×10^{-6} per reactor-year, while the corresponding value for pipe break-induced flooding is 1.3×10^{-5} per reactor-year.

The staff's complete evaluation is in Appendix A of this report, which includes the evaluation of the applicant's PRA study on flooding performed by personnel at Brookhaven National Laboratory (BNL).

On the basis of its review, the staff concludes that although there are discrepancies between the applicant's core vulnerable frequencies and those determined by the staff, this item is satisfactorily resolved. The staff review has determined that this issue provides no basis for further investigation or for the denial of an operating license.

3.13 Long-Term Operability of Deep Draft Pumps

Bulletin IE 79-15 (dated July 1979), issued by the NRC office of Inspection and Enforcement (IE) (IEB 79-15), identified problems with deep draft pumps in operating facilities. These vertical turbine pumps are usually 30 to 60 feet long with impellers in casing bowls at the lowest elevation of the pump and the motor (driver) at the highest elevation; the discharge is just below the motor. This configuration has experienced excessive vibration and bearing wear, which have been attributed to

---

*See Appendix A.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
LONG ISLAND LIGHTING COMPANY	)	Docket No. 50-322-1 (OL)
	)	
(Shoreham Nuclear Power Station, Unit 1)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF'S REPORT PURSUANT TO LICENSING BOARD ORDER OF NOVEMBER 5, 1984, ON REMANDED ISSUES FOR ALAB-788, AND MOTION TO ACCEPT INTO THE EVIDENTIARY RECORD THE AFFIDAVITS OF ANDREW SZUKIEWICZ AND JERRY J. MAUCK" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 14th day of November, 1984:

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

Dr. George A. Ferguson **
Administrative Judge
School of Engineering
Howard University
2300 - 6th Street, N.W.
Washington, D.C. 20059

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

Jonathan D. Feinberg, Esq.
New York State Department of
  Public Service
Three Empire State Plaza
Albany, NY 12223

Fabian G. Palomino, Esq.
Special Counsel to the Governor
Executive Chamber
State Capitol
Albany, NY 12224

Howard L. Blau, Esq.
217 Newbridge Road
Hicksville, NY 11801

W. Taylor Reveley III, Esq.
Hunton & Williams
707 East Main Street
Richmond, VA 23212

Cherif Sedkey, Esq.
Kirkpatrick, Lockhart, Johnson
  & Hutchison
1500 Oliver Building
Pittsburgh, PA 15222

Stephen B. Latham, Esq.
John F. Shea, III, Esq.
Twomey, Latham & Shea
Attorneys at Law
P.O. Box 398
33 West Second Street
Riverhead, NY 11901

Herbert H. Brown, Esq.
Lawrence Coe Lanpher, Esq.
Karla J. Letsche, Esq.
Kirkpatrick and Lockhart
1900 M Street, N.W.
8th Floor
Washington, DC 20036

Atomic Safety and Licensing
Board Panel*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Docketing and Service Section*
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Atomic Safety and Licensing
Appeal Board Panel*
U.S. Nuclear Regulatory Commission
Washington, DC 20555

James B. Dougherty, Esq.
3045 Porter Street, NW
Washinton, DC 20008

Peter S. Everett, Esq.
Hunton & Williams
2000 Pennsylvania Avenue, NW
Washington, DC 20036

Gerald C. Crotty, Esq.
Ben Wiles, Esq.
Counsel to the Governor
Executive Chamber
State Capitol
Albany, NY 12224

Robert Abrams, Esq.
Attorney General of the State
of New York
Attn: Peter Bienstock, Esq.
Department of Law
State of New York
Two World Trade Center
Room 46-14
New York, NY 10047

Bernard M. Bordenick

Bernard M. Bordenick
Counsel for NRC Staff

COURTESY COPY LIST

Edward M. Barrett, Esq.
General Counsel
Long Island Lighting Company
250 Old County Road
Mineola, NY 11501

Mr. Brian McCaffrey
Long Island Lighting Company
Shoreham Nuclear Power Station
P.O. Box 618
North Country Road
Wading River, NY 11792

Marc W. Goldsmith
Energy Research Group, Inc.
400-1 Totten Pond Road
Waltham, MA 02154

Martin Bradley Ashare, Esq.
Suffolk County Attorney
H. Lee Dennison Building
Veteran's Memorial Highway
Hauppauge, NY 11788

Ms. Nora Bredes
Shoreham Opponents Coalition
195 East Main Street
Smithtown, NY 11787

Chris Nolin
New York State Assembly
Energy Committee
626 Legislative Office Building
Albany, New York 12248

MHB Technical Associates
1723 Hamilton Avenue
Suite K
San Jose, CA 95125

Hon. Peter Cohalan
Suffolk County Executive
County Executive/Legislative Bldg.
Veteran's Memorial Highway
Hauppauge, NY 11788

Mr. Jay Dunkleberger
New York State Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Leon Friedman, Esq.
Costigan, Hyman & Hyman
120 Mineola Boulevard
Mineola, NY 11501