DEC 18 1984

Docket No. 50-322

Mr. J. D. Leonard
Vice President - Nuclear
Long Island Lighting Company
175 East Old Country Road
Hicksville, New York 11801

Dear Mr. Leonard:

SUBJECT: ISSUANCE OF SUPPLEMENT NO. 8 TO THE SAFETY EVALUATION REPORT RELATED TO OPERATION OF THE SHOREHAM NUCLEAR POWER STATION, UNIT 1

Printed copies of Supplement No. 8 to the Safety Evaluation Report related to operation of the Shoreham Nuclear Power Station, Unit 1 are enclosed. This report, NUREG-0420 Supplement No. 8, was issued by the Office of Nuclear Reactor Regulation in connection with your application for operating license for the Long Island Lighting Company.

Copies of the report will be placed in the Commission's Public Document Room located at 1717 H Street N.W., Washington, D.C. 20555, and in the local public document room at the Shoreham-Wading River Public Library, Route 25A, Shoreham, New York 11786.

Sincerely,

Original signed by:

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Division of Licensing

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NUREG-0420 Supplement 8 (20)

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NUREG-0420
Supplement No. 8

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

December 1984

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
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Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

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The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

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Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

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Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $4.75

NUREG-0420
Supplement No. 8

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

December 1984

ABSTRACT

Supplement 8 (SSER 8) to the Safety Evaluation Report on Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement addresses several items that have been reviewed by the staff since the previous supplement was issued.

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ACRONYMS

ASME	American Society of Mechanical Engineers
ANSI	American National Standards Institute
BWR	boiling water reactor
CFR	Code of Federal Regulations
FSAR	Final Safety Analysis Report
GDC	General Design Criteria(on)
GE	General Electric
HELB	high energy line break
HPCI	high pressure coolant injection
ISEG	independent safety engineering group
ISMG	instrumentation setpoint methodology group
LILCO	Long Island Lighting Company (applicant)
LRG	Licensee Review Group
MCPR	minimum critical power ratio
MSIV	main steam isolation valve
NRB	Nuclear Review Board
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OL	operating license
QA	quality assurance
RG	regulatory guide
RO	reactor operator
ROC	Review of Operations Committee
SER	Safety Evaluation Report
SLCS	standby liquid control system
SRP	Standard Review Plan
SRO	senior reactor operator
SSER	Safety Evaluation Report Supplement
STA	shift technical advisor
TER	technical evaluation report
TIP	travelling incore probe

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission's Safety Evaluation Report (SER) (NUREG-0420) on the application by Long Island Lighting Company (LILCO or applicant) to operate the Shoreham Nuclear Power Station was issued by the Nuclear Regulatory Commission staff (NRC staff) on April 10, 1981. Supplement 1 (SSER 1) to the Shoreham SER was issued in September 1981; SSER 2 was issued in February 1982; SSER 3 was issued in February 1983; SSER 4 was issued in September 1983; and SSER 5 was issued in April 1984; SSER 6 was issued in July 1984; and SSER 7 was issued in September 1984.

Each of the sections in this SSER 8 is numbered the same as the section of the SER that is being updated. The discussions in this report are supplementary to and not in lieu of the discussions in the SER, except where specifically noted.

Copies of this report are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C. 20555 and at the Shoreham-Wading River Public Library, Route 25A, Shoreham, New York 11786. Copies are also available for purchase from the sources indicated on the inside front cover. The NRC documents and other project-related documents cited in this report are available as described on the inside front cover.

The NRC Project Manager assigned to the operating license (OL) application for Shoreham is Ralph Caruso. He may be contacted by calling (301) 492-7000 or writing to the following address:

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1.7 Outstanding Issues

In Section 1.7 of the SER, the NRC staff identified 61 outstanding issues that were not resolved at the time of issuance of the SER. This report discusses the resolution of a number of these items previously identified as open. The items identified in Section 1.7 of the SER are listed below, with status of each item. If the item is discussed in this supplement, the section where the item is discussed is identified. The resolution of the remaining outstanding issues will be discussed in future supplements to the SER.

Item		Status	Section
(1)	Pool dynamic loads	Resolved	
(2)	Masonry walls	Resolved	
(3)	Piping vibration test program - small bore piping/instrumentation lines	Resolved	
(4)	Piping vibration test program - safety-related snubbers	Resolved	
(5)	LOCA loadings on reactor vessel supports and internals	Resolved	
(6)	Downcomer fatique analysis	Resolved	
(7)	Piping functional capability criteria	Resolved	
(8)	Dynamic qualification	Resolved with license condition	3.10
(9)	Environmental qualification	Resolved with license condition	
(10)	Seismic and LOCA loadings	Resolved	
(11)	Supplemental ECCS calculations with NUREG-0630 model	Resolved with license condition	
(12)	ODYN-Generic letter 81-08	Resolved	
(13)	NUREG-0619 - feedwater nozzle and control rod return line cracking - Generic Letter 81-11	Resolved	
(14)	Jet pump holddown beam	Resolved	
(15)	Inservice testing of pumps and valves	Resolved	

Item		Status	Section
(16)	Leak testing of pressure isolation valves	Resolved	
(17)	SRV surveillance program	Resolved	
(18)	NUREG-0313	Resolved	
(19)	Preservice inspection	Resolved	
(20)	Appendix G - IV.A.2.a	Resolved	
(21)	Appendix G - IV.A.2.c	Resolved	5.3.1
(22)	Appendix G - IV.A.3	Resolved	5.3.1
(23)	Appendix G - IV.B	Resolved	5.3.1
(24)	Appendix H - II.C.3b	Resolved	
(25)	RCIC	Resolved	
(26)	Suppression pool bypass	Resolved	
(27)	Steam condensation downcomer lateral loads	Resolved	
(28)	Steam condensation oscillation and chugging loads	Resolved	
(29)	Quencher air clearing load	Resolved	
(30)	Drywell pressure history	Resolved	
(31)	Impact loads on grating	Resolved	
(32)	Steam condensation submerged drag loads	Resolved	
(33)	Pool temperature limit	Resolved	
(34)	Quencher arm and tie-down loads	Resolved	
(35)	Containment isolation	Resolved	6.2.3, 6.2.5
(36)	Containment purge system	Resolved	
(37)	Secondary containment bypass leakage	Resolved	

Item		Status	Section
(38)	Fracture prevention of containment pressure boundary	Resolved	
(39)	Emergency procedures	Resolved	
(40)	LOCA analyses	Resolved	
(41)	LPCI diversion	Resolved	
(42)	Flow meter	Resolved	
(43)	Loss of safety function after reset	Resolved	7.3.6
(44)	Level measurement errors	Resolved	
(45)	Fire protection	Resolved	
(46)	IE Bulletin 79-27	Resolved	
(47)	Control system failures	Resolved	
(48)	High-energy line breaks	Resolved	
(49)	DC system monitoring	Resolved	
(50)	Low and/or degraded grid voltage condition	Resolved	
(51)	Fracture toughness of steam and feedwater line materials	Resolved	
(52)	Management organization	Resolved	13.4
(53)	Emergency planning (onsite)	Resolved pending confirmation	
(54)	Security	Resolved	
(55)	Q-list	Resolved	
(56)	Financial qualification	Resolved	
(57)	TMI-2 requirements:		
	Shift technical advisor	Resolved with license condition	
	Shift supervisor administrative duties	Resolved	

Item	Status	Section
Shift manning	Resolved	
Upgrade operator training	Resolved	13.1
Training programs - operators	Resolved	13.2
Organization and management	Resolved	
Procedures for transients and accidents	Resolved	
Shift relief and turnover procedures	Resolved	
Control room access	Resolved	
Dissemination of operating experiences	Resolved	
Verify correct performance of operating activities	Resolved	
Vendor review of procedures	Resolved	
Emergency procedures	Resolved	
Control room design review	Resolved with license condition	
Training during low-power testing	Resolved	
Reactor coolant system vents	Resolved	
Plant shielding	Resolved	
Post-accident sampling	Resolved with license condition	
Degraded core training	Resolved	
Hydrogen control	Resolved	
Relief and safety valves	Resolved	
Valve position indication	Resolved	
Dedicated hydrogen penetrations	Resolved	
Containment isolation dependability	Resolved	

Item	Status	Section
Accident-monitoring instrumentation		
Attachment 1	Resolved with post-implementation review	
Attachment 2	Resolved	
Attachment 3	Resolved	
Attachment 4	Resolved	
Attachment 5	Resolved	
Attachment 6	Resolved	
Inadequate core cooling	Resolved	
IE Bulletins		
Item 5	Resolved	
Item 10	Resolved	
Item 22	Resolved	
Item 23	Resolved	
Bulletins and Order Task Force		
Item 3	Resolved	
Item 13	Resolved	
Item 16	Resolved	
Item 17	Resolved	
Item 18	Resolved	
Item 21	Resolved	
Item 22	Resolved	
Item 24	Resolved	
Item 25	Resolved	
Item 27	Resolved	

Item		Status	Section
	Item 28	Resolved	
	Item 30	Resolved	
	Item 31	Resolved	
	Item 44	Resolved	
	Item 45	Resolved	
	Item 46	Resolved	
	Emergency preparedness - short term	Under review	
	Upgrade emergency support facilities	Resolved	
	Emergency preparedness - long term	Under review	
	Primary coolant outside containment	Resolved	
	Improved iodine monitoring	Resolved	
	Control room habitability	Resolved	
(58)	Reactor vessel materials toughness	Resolved	
(59)	Control of heavy loads - Generic Letter 81-07	Resolved	9.1.5
(60)	Station blackout - Generic Letter 81-04	Resolved	Appendix B (USI A-44)
(61)	Scram system piping	Resolved	
(62)	Remote shutdown system	Resolved with license condition	7.4.3
(63)	Design verification	Resolved	
(64)	Loose parts monitoring system	Resolved	
(65)	Reactor building flooding	Resolved	
(66)	Deep draft pumps (IEB-79-15)	Resolved	
(67)	Reactor internal and core support material	Resolved with license condition	
(68)	GHOSH code	Resolved	
(69)	LPCI annunciator	Resolved	

Item		Status	Section
(70)	Core spray logic	Resolved	7.3.10
(71)	Nearby industrial transportation and military facilities	Resolved	2.2.2
(72)	Instrument setpoints	Resolved	7.2.6
(73)	Physical separation in NSSS panels	Resolved	7.6.6

2 SITE CHARACTERISTICS

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.2 Nearby Facilities

The staff has reviewed the fire hazards associatd with the onsite oil storage tank. The tank contains No. 2 fuel oil and has a nominal capacity of 23,100 barrels. The tank is surrounded by a 150-foot-diameter, 8-foot-high steel dike that has a nominal capacity of 25,200 barrels. The staff has estimated the thermal fluxes as a result of a postulated fire following a fuel tank failure. The thermal flux at the nearest safety-related structure was calculated to be approximately 10 kW/m^2. Hence, the postulated fire does not pose a significant risk to the safe operation of the plant. The staff considers this item resolved.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.10 Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

3.10.1 Background

In Section 3.10.4 of SSER 7, the staff reported that the applicant had made significant progress toward completing the equipment seismic and dynamic qualification program. However, several pieces of equipment remained to be qualified.

By letter dated November 9, 1984 (SNRC-1105), the applicant reported that qualification of the following had been completed: (1) the scram discharge volume vent and drain valves, (2) the scram discharge volume solenoid valves, and (3) the power range monitor panel (H11-P608/1H11*PNL608). The applicant also reported that the high pressure coolant injection (HPCI) turbine qualification would be completed in mid-November, and committed not to use the invessel storage rack (F16-E006/1F16*FAK-09) as an invessel storage area for fuel bundles until seismic qualification of the rack is complete. On the basis of these reports, the staff finds the qualification of these items to be complete, except that the staff will condition the Shoreham license to prohibit the use of the invessel storage rack until its qualification is completed.

3.10.2 Exemption Request

Additionally in its November 9, 1984 submittal, the applicant requested an exemption from the provisions of General Design Criterion (GDC) 2 regarding the seismic qualification of the radiation monitoring panels (1D11*PNL-117A and B) and the radiation monitoring pumps (1D11*P-126 and 134). A discussion of each of these pieces of equipment follows.

(1) Radiation Monitoring Panels (Mark 1D11*PNL-117A and B)

The recorder power supplies (1D11*E/S-117A and C) provide power to the recorders that are located in the control room cabinets (1D11*PNL-117A and B). These power supplies are the only items in the cabinets for which seismic qualification is not complete. The recorders are used to keep a historical log of the readings generated at panels 1D11*PNL-126 and 134, and this information is then utilized in determining the release of radioactivity to the surroundings. The applicant reported that the subcomponent requiring qualification is currently involved in a test program that is expected to be completed, with a report available, in the first quarter of 1985. Should an accident occur and lead to failure of the recorder power supplies, the indicating devices in the control room cabinet will not be affected because both physical and electrical isolation is provided between devices. Therefore, the failure of these devices will not degrade the monitoring function of any other components.

Failure of the devices will require that an operator periodically record information from the indicating devices mentioned above so that the estimate of the release of radioactivity to the surroundings can be generated. In SSER 7, the staff found that interim operation of the cabinets and internals is acceptable for power levels not to exceed 5% power.

(2) Radiation Monitoring Pumps (Mark 1D11*P-126, 134)

The specific items of concern with this equipment are the auxiliary pump skids used to supply the sample air to the post-accident station vent and reactor building standby ventilation system exhaust monitors. If there is seismic failure of the pump skids, alternate means--such as sampling via the post-accident sampling system, grab sampling of the effluents, and normal range monitors--are available to determine the gaseous effluent releases from the plant. It should be noted that the buildup of radioactivity inventory during operation at a power level up to 5% will be comparatively small. In view of these considerations, the staff found the equipment acceptable for interim operation for power levels not to exceed 5% power. These findings were documented in SSER 7.

3.10.3 Evaluation

Operation of the plant pending the qualification of this radiation monitoring equipment is as safe as operation with qualified panels. During the Phases I and II low power testing activities, there are no accidents for which this equipment must function. The only function this equipment serves during operation beyond Phases I and II is post-accident monitoring. Thus, this exemption has no effect on the probability of any postulated accident. The consequences of any accident are not affected because the monitoring function can be accomplished by alternate methods as described above.

Therefore, the staff has concluded that the operation at up to 5% of full power without having this equipment fully qualified is as safe as operation in full compliance with the regulations, and is acceptable.

As set forth in the Commission's decision in Shoreham (Long Island Lighting Company) (Shoreham Nuclear Power Station, Unit 1), CLI-84-8 (May 16, 1984), the Commission regards the use of the exemption authority under Title 10 of the Code of Federal Regulations Part 50.12 (10 CFR 50.12) as extraordinary. The availability of an exemption requires a finding of exigent circumstances that favor the granting of an exemption. Pursuant to the Commission's *Shoreham* decision, a determination as to whether exigent circumstances warrant an exemption should include a consideration of the stage of the facility's life, any financial or economic hardships, any internal inconsistencies in the regulation, the applicant's good faith effort to comply with the regulation from which an exemption is sought, the public interest in adherence to the Commission's regulations, and the safety significance of the issues involved.

With regard to the stage of the facility's life, construction of the Shoreham Nuclear Power Station is complete and the facility is ready for fuel loading and low power testing. Absent the requested exemption and consequent authorization to load fuel and conduct low power testing, the facility essentially would remain idle, unused, and untested until the seismic qualification of the radiation monitoring pumps and valves is completed and full compliance with GDC 2 is shown. Thus, without the requested exemption, fuel loading and attendant useful facility testing would be delayed pending the completion of work on this equipment. In this circumstance, the stage of the facility's life would appear to favor issuance of the exemption.

With regard to financial or economic hardship, in its October 29, 1984 Initial Decision (ASLBP No. 77-347-01C-OL), the Atomic Safety and Licensing Board noted that it is almost self evident that there must be financial hardships to someone when there is a physically completed nuclear facility that is standing unused and nonproductive because of substantial licensing delays. If this exemption is not granted, the applicant will be subjected to financial and economic hardships. On the other hand, the staff has identified no financial or economic hardships that would result if the exemption were granted. Financial and economic considerations thus appear to favor issuance of the exemption.

No internal inconsistencies in the regulation are apparent and, in this instance, this factor appears to weigh neither in favor of, nor against, a finding of exigent circumstances and issuance of the requested exemption.

As to the applicant's good faith efforts to comply with GDC 2, the NRC staff has known that this equipment might not be qualified before licensing, and, in fact, in SSER 7 specifically addressed the matter and determined that the equipment did not have to be qualified before 5% of rated power is exceeded. The applicant is making a bona fide effort to qualify this equipment and achieve compliance; the applicant cannot be faulted if a heroic effort was not made to complete the work sooner, considering the previous staff position expressed in SSER 7. In these circumstances, the equities lie in favor of granting the exemption.

Finally, although the public interest favors adherence to the Commission's regulations, the staff has concluded that in this instance, where a limited and temporary exemption from compliance with GDC 2 for fuel loading and low power testing has no adverse safety significance (as noted above) and yet would allow the efficient and expeditious testing of facility components and systems, it is not contrary to the public interest to grant the requested exemption.

In accordance with the Commission's directions in Shoreham then, taking into account the equities of the situation, the staff finds that those equities weigh in favor of granting the requested exemption. In sum, the staff finds, based on the readiness of the facility for fuel loading and low power testing, the usefulness of such testing, the potential for adverse economic impacts absent an exemption, the applicant's good faith efforts at compliance with the regulations, and the lack of adverse safety significance or any detriment to the public interest from granting the requested exemption, that exigent circumstances exist that favor the granting of an exemption under 10 CFR 50.12(a).

Based on the foregoing, and in accord with the Commission's decision on Shoreham, CLI-84-8, and 10 CFR 50.12(a), the staff has concluded that the partial exemption from the requirements of GDC 2 as discussed above is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

4 REACTOR

4.2 Fuel System Design

4.2.3 Design Evaluation

4.2.3.13 Channel Box Deflection

Boiling-water reactor (BWR) fuel channels provide structural stiffness for the fuel assemblies and distribute the coolant flow between the assemblies and channel bypass regions. The channels are subject to time-dependent, permanent dimensional changes (i.e., deflections) that result from irradiation, creep, and stress-relaxation effects. The resultant bulge (from long-term thermal creep) or bow (from differential irradiation-induced axial growth) reduces the size of the gap available for control rod insertion. Channel box deflection is thus a phenomenon that can limit channel life because of the potential for interfering with control blade motion.

In a generic topical report (NEDE-21354-P), General Electric (GE) describes a channel lifetime prediction method and makes a backup recommendation for periodic channel deflection measurements that consist of settling friction tests. After consideration of the factors involved, the staff concluded that settling friction tests or an acceptable alternative (such as channel dimensional deflection measurements) should be performed. In a memorandum from L. Rubenstein dated September 18, 1981,* the staff outlined a method that could be used to resolve the channel box deflection issue for several near-term BWR operating license applications. Basically, the staff advocated a multistep procedure that had been proposed by the Zimmer applicant. The key ingredient of the Zimmer plan was a commitment to (1) perform some control rod settling friction tests, which would provide an exact profile of control rod drive friction versus position at refueling outages, or (2) make some actual channel dimensional measurements.

In a letter from J. L. Smith (LILCO) to H. R. Denton (NRC), dated February 16, 1983, the applicant described a channel box management program that was incorporated into the Shoreham station procedures. The program is basically identical to the one adopted by Licensee Review Group (LRG) II, which is almost identical to the Zimmer plan that was approved by the NRC staff (Rubenstein, August 19, 1982).

The applicant's program includes the following features:

(1) Records will be kept of channel locations and exposure for each cycle of operation.

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*Correspondence cited herein is available as described on the inside front cover.

(2) Channels shall not reside in the outer row of the core for more than two operating cycles (because flux gradients are largest near the core periphery and, therefore, differential irradiation-induced growth and bowing will be greatest at those locations).

(3) At the beginning of each fuel cycle, the combined outer row residence time for any two channels in any control rod cell shall not exceed four peripheral cycles.

(4) Channels that reside in the outer row for more than one cycle shall be situated in core locations that rotate the channel so that a different side faces the core edge.

(5) Channels that reside in the outer row of the core for three or more cycles should not be shuffled inward.

In addition, by letter dated July 11, 1984, the applicant proposed to perform a control rod drive friction test for those cells that exceed these general guidelines or contain channels with exposures greater than recommended exposures. The applicant also stated that fuel channel deflection measurements might be used to identify the amount of remaining lifetime for those channels that exceed the general guidelines and exposure level.

Because of the similarity of the proposed program to the programs approved for LRG-II and Zimmer, the staff concludes that the Shoreham channel box management program is acceptable. Thus, the channel box license condition is removed, and the item is considered resolved.

However, the staff is continuing its review of the channel box deflection phenomenon, and if the review indicates that modification of the proposed steps is necessary, the applicant will be so notified.

4.4 Thermal-Hydraulic Design

4.4.1 Evaluation

4.4.1.1 Thermal-Hydraulic Analysis Methods and Thermal-Hydraulic Stability

To ensure that the thermal-hydraulic safety design criteria regarding the thermal-hydraulic stability margin will be met for operations beyond the first cycle core, SER Section 4.4.2 stated: "Operating beyond Cycle 1 is not permitted until a new stability analysis is provided and approved for the second cycle of operation."

The applicant has been notified that the existing analyses do not support operation beyond Cycle 1. By letter dated October 19, 1984 (SNRC-1095), the applicant agreed to submit for staff review the similar analytical results including the minimum critical power ratio (MCPR) limits and thermal-hydraulic stability margin, as part of the reload licensing application for operation beyond Cycle 1 core operation. On the basis of this agreement, the staff has concluded imposition of the license condition quoted above is not necessary. The staff will review the analytical results when they become available, and provide the evaluation results in a future safety evaluation.

4.4.1.2 Single Loop/Natural Circulation Operation

In Sections 4.4.1 and 4.4.2 of the SER, the staff determined that operation of the Shoreham reactor in either the natural circulation mode or with only one operating recirculation loop could not be permitted until supporting safety analyses had been submitted and approved. The SER further stated that the license would be conditioned to prohibit such operation. Because Section 3.4.1.1 of the Shoreham Technical Specifications requires that two reactor coolant system recirculation loops must be in operation at all times when the reactor is in operational conditions 1 and 2 (except during special tests), the imposition of this license condition would be redundant and is, therefore, not necessary.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

In Section 5.3.1 of SSER 3 and SSER 4, the staff reported the results of its review of the Shoreham reactor vessel materials. The staff indicated that exemptions would be required to Paragraphs III.B.3, III.B.4, III.C.2, IV.A.2.c, IV.A.3, and IV.B of Appendix G, 10 CFR 50 and Paragraph II.B of Appendix H, 10 CFR 50. Since those evaluations were proposed, however, Appendices G and H have been revised. The revisions became effective on July 26, 1983.

In lieu of the requirements in Appendix G (which were discussed in the staff's previous safety evaluations), the revised Appendix G requires that the fracture toughness program meet the edition and addenda of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME Code), as permitted by 10 CFR 50.55a. As discussed in the staff evaluation, the fracture toughness test program for Shoreham does not comply with the ASME Code fracture toughness requirements. However, Paragraph III.A of Appendix G permits, for a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, that the fracture toughness data and data analyses may be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of the Appendix. Shoreham was constructed to an ASME Code that was earlier than the Summer 1972 Addenda of the 1971 Edition.

In the safety evaluation, the staff considered the fracture toughness data and data analyses that were presented by the applicant. The staff considers that the data presented by the applicant demonstrate that the fracture toughness properties of the ferritic reactor coolant pressure boundary materials are equivalent to those required by the Appendix. Hence, exemptions to Appendix G are no longer required.

As a result of changes to Appendix H to 10 CFR 50, the Appendix H exemption is no longer required because the licensee's reactor vessel surveillance program complies with the revised Appendix H requirements.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.3 Containment Isolation System

6.2.3.1 Background

In Section 6.2.3 of SSER 3 and SSER 4 the staff reported the results of its review of the containment isolation provisions for certain instrument lines that penetrate reactor containment. In those evaluations, the staff determined that certain classes of instrument lines did not meet the explicit requirements of GDC 56 and did not satisfy the requirements of the acceptable alternative methods described in Regulatory Guide (RG) 1.11. However, the staff did determine that the facility could operate in its present condition until the first refueling outage, at which time the instrument lines would have to be modified. By letter dated November 9, 1984 (SNRC-1105), the applicant formally requested an exemption from the provisions of GDC 56 for these instrument lines.

6.2.3.2 Discussion

The containment isolation capability required by GDC 56 is intended to preclude the release of radioactive material from the containment following an accident inside the containment. This is normally accomplished by providing two isolation valves on each line. Although the instrument lines in question have only a single manual valve for isolation purposes, all instruments (including the internal pressure boundary of the instrument), sensing lines, and standpipes identified as extensions of the primary containment boundary are capable of maintaining pressure boundary integrity during and following postulated design-basis events concurrent with a seismic event. During plant operation, the integrity of the instrument lines is demonstrated by the proper operation of the instruments. Following an accident, loss of integrity can be postulated only as a random passive failure of the instrument line pressure boundary. This is an extremely unlikely event. The peak pressure that would be experienced by the lines and instruments during an accident is limited to the projected peak containment pressure of 46 psig, whereas the design pressure of the instruments and tubing is much greater than the projected peak containment pressure. The integrated leak rate test and other tests conducted on these lines during the construction or pre-operational testing phases demonstrate this capability. Thus, the design provides substantial double barrier protection.

The staff notes, however, that one exception to this analysis exists: a static O-ring for pressure switch 1T49-PS085. However, this instrument is used for periodic containment leak rate testing, does not provide a safety function, and has its own manual isolation valve. The individual manual isolation valve upstream of this instrument will be maintained normally closed until the O-ring is demonstrated to maintain pressure boundary integrity under the conditions described above.

In addition, each instrument line in question is sized so that if a postulated failure of the piping or of any component (including any valve body in the line outside primary reactor containment) were to occur during normal reactor operation

(1) The resulting leakage will be reduced to the maximum extent practical consistent with other safety requirements.

(2) The integrity and functional performance of the secondary containment and its associated safety systems (e.g., filters) will be maintained.

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

During operation in Phases I and II of low power testing, primary containment isolation capability is not required because primary containment is not, and need not be, established. Indeed, there can be no release of fission products because none will be generated during Phase I and a negligible quantity exists during Phase II. Thus, this exemption request has no impact on the safety of the plant during these phases.

Beyond Phases I and II, operation of Shoreham with the current design would be, in substance, as safe as operation of a plant in full compliance with GDC 56 or RG 1.11 (which provides guidance for complying with GDC 56). First, viewed in the context of the overall safety of the plant, the exemption request has little safety significance. Only a relatively small number of the plant's containment instrument line penetrations do not meet GDC 56. Moreover, the condition will only exist for a small fraction of the life of the plant because the staff will condition the Shoreham license to require the applicant to make the necessary modifications to the plant prior to startup after the first refueling outage. Because loss-of-coolant accidents are extremely unlikely events, the probability that one would occur during the first cycle of operation is very remote. Second, the current design provides single barrier protection with a backup manual isolation valve on the instrument lines in question. Third, each instrument line is sized to minimize the radiological consequences of a rupture.

Although operation with the proposed exemption would not provide exactly the same margin of safety as would operation in full compliance with GDC 56, the staff is not required to deny an exemption if granting it would reduce a margin of safety by only an insignificant amount. As discussed in its Initial Decision (ASLBP No. 77-347-01C-OL) dated October 29, 1984, the Atomic Safety and Licensing Board determined that "the question of 'as safe as' must be approached in a functional sense (does it serve the purpose of protecting public health and safety) rather than in an absolute sense (is it the very best possible machine available for the purpose)" (Initial Decision, p. 26). In the staff's view, on the basis of the technical discussion presented above and in SSER 3 and SSER 4, the current configuration of the containment isolation provisions for these instrument lines is substantially as safe as a configuration that would meet the literal requirements of GDC 56 or RG 1.11. For that reason, the staff concludes that operation of Shoreham, until the first refueling outage without the installation of additional containment isolation valves in the instrument lines discussed in SSER 4, would be substantially as safe as operation in full compliance with the regulations and, therefore, the standard set forth by the Commission in CLI-84-8 is satisfied. It is, therefore, acceptable.

As set forth in the Commission's decision in Shoreham (Long Island Lighting Company) (Shoreham Nuclear Power Station, Unit 1), CLI-84-8 (May 16, 1984), the Commission regards the use of the exemption authority under 10 CFR 50.12 as extraordinary. The availability of an exemption requires a finding of exigent circumstances that favor the granting of an exemption. Pursuant to the Commission's Shoreham decision, a determination as to whether exigent circumstances warrant an exemption should include a consideration of the stage of the facility's life, any financial or economic hardships, any internal inconsistencies in the regulation, the applicant's good faith effort to comply with the regulation from which an exemption is sought, the public interest in adherence to the Commission's regulations, and the safety significance of the issues involved.

With regard to the stage of the facility's life, construction of Shoreham is complete and the facility is ready for fuel loading, low power testing, and eventual commercial operation. Absent the requested exemption and consequent authorization to load fuel, conduct low power testing, and generate power, the facility would remain idle, unused, and untested until the additional containment isolation valves are installed and testing is complete. Thus, without the requested exemption, testing and eventual operation would be delayed. In this circumstance, the stage of the facility's life would appear to favor issuance of the exemption.

With regard to financial or economic hardships, in its October 29, 1984 Initial Decision (ASLBP No. 77-347-OIC-OL), the Atomic Safety and Licensing Board noted that it is almost self-evident that there must be financial hardship to someone when there is a physically completed nuclear facility standing unused and nonproductive because of substantial licensing delays. If this exemption is not granted, the applicant will be subjected to financial and economic hardships. On the other hand, the staff has identified no financial or economic hardships that would result if the exemption were granted. Financial and economic considerations thus appear to favor issuance of the exemption.

No internal inconsistencies in the regulation are apparent and, in this instance, this factor appears to weigh neither in favor of, nor against, a finding of exigent circumstances and issuance of the requested exemption.

As to the applicant's good faith efforts to comply with the regulations, the NRC staff explicitly approved a delay in installing these isolation valves in SSER 4, which was issued in September 1983. The applicant is otherwise in compliance with the provisions of GDC 56, as well as with GDC 55, GDC 57, and RG 1.11, all of which apply to containment isolation. In these circumstances, the equities lie in favor of granting the exemption.

Finally, while the public interest favors adherence to the Commission's regulations, the staff has concluded that in this instance, where a limited and temporary exemption from compliance with GDC 56 for operation until the first refueling outage has no adverse safety significance (as noted above) and yet would allow the efficient and expeditious testing and operation of the facility, it is not contrary to the public interest to grant the requested exemption.

In accordance with the Commission's directions in Shoreham then, taking into account the equities of the situation, the staff finds that those equities weigh in favor of granting the requested exemption. In sum, the staff finds--based on the readiness of the facility for fuel loading, low power testing, and eventual commercial operation, the potential for adverse economic impacts absent an exemption, the applicant's good faith efforts at compliance with the regulations, and the lack of adverse safety significance or any detriment to the public interest from granting the requested exemption--that exigent circumstances exist that favor the granting of an exemption under 10 CFR 50.12(a).

Based on the foregoing, and in accord with the Commission's decision on Shoreham, CLI-84-8 and 10 CFR 50.12(a), the staff has concluded that the partial exemption from the requirements of GDC 56, as discussed above, is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

6.2.5 Containment Leakage Testing

6.2.5.1 Main Steam Isolation Valves

Appendix J to 10 CFR 50 requires local leak rate testing of boiling water reactor main steam isolation valves (MSIVs) at the peak calculated containment pressure for the design-basis accident (Appendix J II.H.4 and III.C.2). Furthermore, Appendix J requires that the measured leak rates be included in the summation of the local leak rate test results (III.C.3). In Section 6.2.5.1 of the SER, the staff concluded that an exemption is justified at Shoreham to allow local leak rate testing of the MSIVs at a reduced pressure and to exclude the measured leakage from the combined local leak rate for the local leak rate test results.

By letter dated November 9, 1984 (SNRC-1105), the applicant formally requested an exemption from the above provisions of Appendix J to 10 CFR 50.

Testing the MSIVs at a reduced pressure was accepted by the staff because testing at peak calculated containment pressure produces demonstrably inaccurate and misleading results. This is true because each main steamline has two MSIVs configured so that post-accident pressure in the steamlines serves to seat the valves rather than open them. To test the two MSIVs simultaneously, however, the line between the valves must be pressurized, so the pressure applied to the inboard valve is in the direction reverse to that expected during an accient. This testing in the reverse direction tends to unseat the inboard valve, lifting the disc, and permitting leakage past it at peak calculated containment pressure. Thus, testing at the peak calculated containment pressure would be meaningless because the inboard valve would unseat, allowing excessive leakage.

To remedy this problem, the proposed test calls for a test pressure of 25 psig (instead of peak calculated containment pressure of 46 psig) to avoid lifting the disc of the inboard valve. The total observed leakage through both valves (inboard and outboard) is then assigned to the penetration. If the combined leakage of two valves exceeds the Technical Specification-allowable value for any one MSIV, further testing is done to discriminate leakage between the valves. This ensures that no single valve exceeds the allowable leakage that is assumed for radiological consequences by the safety analysis. In addition, because the inboard valve is tested in the reverse direction, the post-accident leak rate is likely to be less than the test leak rate, because the valve would tend to seat more firmly under accident conditions. Although this phenomenon has not been quantified, the effect of the reduction in test pressure would clearly tend to be offset by the effects of testing in the reverse direction.

Excluding the measured MSIV leakage from the combined local leak rate for the local leak rate test results is justified because this type of leakage and its radiological consequence has been separately accounted for in the safety analysis. In the event of a loss-of-coolant accident, the MSIV leakage control system will maintain a negative pressure between the MSIVs. Any leakage into this space will be discharged into a volume where it will be processed by the reactor building standby ventilation system before it is released to the environment. A separate radiological analysis for this potential source of containment atmosphere leakage was performed and the results are documented in the Shoreham FSAR, Chapters 6 and 15. The periodic local leak rate test will ensure that the leakage assumed in the analysis is not exceeded.

Based on this discussion, the staff has concluded that operation of Shoreham with this exemption is as safe as operation would be without the exemption and is, therefore, acceptable. The exemption does not have any impact on the operation of plant equipment. With respect to the adequacy of testing, as shown above, the testing that will be performed is conservative and, therefore, will yield results similar to or more conservative than those that would have been obtained by literal Appendix J testing.

As set forth in the Commission's decision in Shoreham (Long Island Lighting Company) (Shoreham Nuclear Power Station, Unit 1), CLI-84-8 (May 16, 1984), the Commission regards the use of the exemption authority under 10 CFR 50.12 as extraordinary. The availability of an exemption requires a finding of exigent circumstances that favor the granting of an exemption. Pursuant to the Commission's Shoreham decision, a determination as to whether exigent circumstance warrant an exemption should include a consideration of the stage of the facil s life, any financial or economic hardships, any internal inconsistencies in the regulation, the applicant's good faith effort to comply with the regulation from which an exemption is sought, the public interest in adherence to the Commission's regulations, and the safety significance of the issues involved.

With regard to the stage of the facility's life, construction of Shoreham is complete and the facility is ready for fuel loading, low power testing, and eventually, commercial operation. Absent the requested exemption and consequent authorization to load fuel, conduct low power testing, and generate power, the facility would remain idle, unused, and untested until the MSIVs could be replaced or modified to allow such testing as is strictly required by Appendix J. The staff knows of no similar valves that have been modified to allow such testing, because exemptions such as this one have frequently been granted to other applicants and licensees. Replacement of the valves would require extensive rework of existing systems and would delay operation significantly. In this circumstance, the stage of the facility's life would appear to favor issuance of the exemption.

With regard to financial or economic hardship, in its October 29, 1984 Initial Decision (ASLBP No. 77-347-01C-OL), the Atomic Safety and Licensing Board noted that it is almost self-evident that there must be financial hardship to someone when there is a physically completed nuclear facility standing unused and nonproductive because of substantial licensing delays. If this exemption is not granted, the applicant will be subjected to financial and economic hardships. On the other hand, the staff has identified no financial or economic hardships that would result if the exemption were granted. Financial and economic considerations thus appear to favor issuance of the exemption.

No internal inconsistencies in the regulation are apparent, but the staff notes that this exemption from Appendix J is not unique to Shoreham. In fact, it is included as part of the Standard Technical Specifications and is consistent with current regulatory practice for boiling water reactors. This factor, therefore, appears to weigh in favor of a finding of exigent circumstances and the issuance of the proposed exemption.

As to the applicant's good faith efforts to comply with Appendix J, the staff explicitly approved this exemption in the SER, which was issued in April 1981, so the applicant cannot be faulted for not attempting to make a heroic effort to fully comply. In all other respects, Shoreham complies with Appendix J. In these circumstances, the equities lie in favor of granting the exemption.

Finally, while the public interest favors adherence to the Commission's regulations, the staff has concluded that in this instance, where exemption from strict compliance with Appendix J testing has no adverse safety significance (as noted above) and yet would allow the efficient and expeditious testing and operation of the facility, it is not contrary to the public interest to grant the requested exemption.

In accordance with the Commission's directions in Shoreham then, taking into account the equities of the situation, the staff finds that those equities weigh in favor of granting the requested exemption. In sum, the staff finds--based on the readiness of the facility for fuel loading, low power testing and commercial operation, the potential for adverse economic impacts absent an exemption, the applicant's good faith efforts at compliance with the regulations, and the lack of adverse safety significance or any detriment to the public interest from granting the requested exemption--that exigent circumstances exist which favor the granting of an exemption under 10 CFR 50.12(a).

Based on the foregoing, and in accord with the Commission's decision on Shoreham, CLI-84-8 and 10 CFR 50.12(a), the staff has concluded that the partial exemption from the requirements of Appendix J to 10 CFR 50, as discussed above, is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.6 Instrument Setpoints

During the OL review of the Shoreham Technical Specifications, the staff identified a concern regarding the values selected for protection system instrument setpoints and, in general, the methodology used to establish the reactor protection system setpoints. It was determined that additional information would be required to confirm the applicant's conformance with the Commission's regulations relevant to the issue of protection system setpoints.

The applicable regulations are: GDC 20, 10 CFR 50.36, and 10 CFR 50.46. GDC 20 states: "...the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety." 10 CFR 50.36 states: "limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." 10 CFR 50.46 specifies the performance criteria for the emergency core cooling systems. These criteria include a maximum peak cladding temperature, a maximum cladding oxidation, a maximum total amount of hydrogen generated, and requirements that core geometry remain amenable to cooling for long-term decay heat removal. Guidance on acceptable methods for complying with these regulations is contained in RG 1.105.

In an effort to conserve resources and to take advantage of an ongoing review effort, the applicant joined with several other BWR owners that had formed a Licensee Review Group (LRG)--the Instrumentation Setpoint Methodology Group (ISMG)--so that the requested information could be provided to the staff. The applicant's commitment to join the ISMG was provided in a letter dated November 23, 1983 from J. L. Smith (LILCO) to Harold R. Denton (NRC).

On July 14, 1983, the staff met with the ISMG at the request of the ISMG. At this meeting, the ISMG presented an outline of a setpoint methodology. In response to additional questions from the staff, another meeting was held on January 31, 1984. By letter dated May 15, 1984, from T. M. Novak (NRC) to J. F. Carolan (Chairman, ISMG), the staff provided its assessment of ISMG methodology. The staff evaluation identified several deficiencies in the methodology presented and requested that the ISMG provide additional information in response to 10 specific concerns. In response to the staff's evaluation, by letter dated June 29, 1984 (from J. F. Carolan to T. M. Novak), the ISMG provided an action plan for resolving the outstanding issues. By letter dated

July 23, 1984 (from B. J. Youngblood (NRC) to J. F. Carolan), the staff accepted the proposed action plan.

By letter dated November 12, 1984 from J. P. Leonard (LILCO) to Harold R. Denton (NRC), the applicant committed to utilize the ISMG methodology being developed (if it is suitable for Shoreham) or to develop a plant-specific methodology for Shoreham (if the ISMG methodology is not suitable) to close out the remaining items of this concern. This information is to be submitted to the staff within 6 months of the completion of the generic effort. The final acceptability of the protection system instrumentation setpoints will be addressed after the staff completes it review of the information.

On the basis of meetings with the ISMG and the applicant's commitment, the staff concludes that there is reasonable assurance that the information on the setpoint methodology being developed will verify the acceptability of the proposed setpoints. In the interim, the staff finds the proposed setpoints acceptable.

7.3 Engineered Safety Feature Systems

7.3.6 Loss of Function After Reset

In Section 7.3 of SSER 7 the staff reported that it would condition the Shoreham operating license to require that the applicant modify the design of the traveling incore probe (TIP) system before startup after the first refueling outage. This modification would prevent reinsertion of the TIP probes upon reset of an engineered safety feature actuation signal. By letter dated November 9, 1984 (SNRC-1105), the applicant reported that the modifications to the TIP system have been completed. Thus, this item is resolved.

7.3.10 Core Spray Valve Logic and Setpoint

In Section 7.3.10 of SSER 7, the staff reported that it would condition the Shoreham license to require additional testing of certain core spray system check valves. This testing will be incorporated into the Shoreham Technical Specifications, and, therefore, it is not necessary to include it in a separate license condition.

7.4 Systems Required for Safe Shutdown

7.4.3 Remote Shutdown System

In Section 7.4.3 of SSER 3 and SSER 7, the staff reported the results of its review of the Shoreham remote shutdown system. The staff found that the design of the remote shutdown system would meet GDC 19 and Sections 7.4.II and III of the Standard Review Plan after certain additional design changes were made by the end of the first refueling outage and after an acceptable procedure verification test had been performed for the new remote shutdown system. By letter dated November 9, 1984 (SNRC-1105), the applicant formally requested an exemption from GDC 19 until these modifications could be completed.

GDC 19 requires that equipment at appropriate locations outside the control room shall be provided with (1) a design capability for prompt hot shutdown of the reactor, including instrumentation and control necessary to maintain the unit in a safe condition during hot shutdown and (2) a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Section 7.4 of the Standard Review Plan states that the remote control stations and equipment used to maintain safe shutdown should be designed to accommodate a single failure. Because GDC 19 contains no explicit requirement that the remote shutdown capability be able to withstand a single failure, the applicant did not agree that exceptions to the single failure criterion from the remote shutdown panel constitute noncompliance with NRC regulations. This issue was the subject of a contention before the Atomic Safety and Licensing Board presiding over the Shoreham operating license proceeding. The contention was settled by the parties before it could be litigated.

The "Resolution of Suffolk County Contention 1--Remote Shutdown Panel" and Section 7.4.3 of SSER 3 listed the instrumentation and controls that are needed to meet the single failure criterion noted above. Some of these items have been identified by the applicant as already existing in the plant. The applicant committed to (1) provide additional instrumentation and controls prior to completion of the first refueling outage and (2) upgrade the qualification of other instrumentation. This proposal was agreed upon by Suffolk County and the staff in the "Resolution of Suffolk County Contention 1--Remote Shutdown Panel." This settlement agreement was accepted by the Licensing Board on December 21, 1982 (Tr. 17,198). The staff published its rationale for accepting the proposal in Section 7.4.3 of SSER 3. First, there is an extremely low probability that an event requiring evacuation of the control room will occur concurrently with a single failure in the primary shutdown path at the remote shutdown panel during the first cycle of operation. The staff also found that the redundant systems themselves will still be operable from remote locations. Only the redundancy in the indication for certain parameters will not be available until the first refueling outage. Thus, this exemption request does not affect the ability of plant equipment to perform its required function. Furthermore, as documented in SSER 7, a system operational verification test of the remote shutdown panel was successfully performed assuming a single worst case failure.

Operation of the plant during Phases I and II is unaffected by the exemption request. During Phase I, the reactor is in a cold shutdown condition. Thus, the remote shutdown panel has no effect on the plant's ability to reach cold shutdown. Similarly, during Phase II operation, use of the remote shutdown panel would not be necessary to reach cold shutdown because cold shutdown would be attained by merely transferring the reactor mode switch to shutdown. Thus, for Phases I and II, operation of the plant with this exemption would be as safe as operation with the monitoring equipment in question installed.

Operation of Shoreham beyond Phases I and II during the first cycle with the present remote shutdown systems is, in substance, as safe as operation would be utilizing a remote shutdown system which strictly meets the single failure criterion. In a letter dated November 23, 1981 (SNRC-638), the applicant documented a single failure analysis for the Shoreham remote shutdown system that demonstrated that sufficient equipment was available to ensure that safe shutdown could be achieved assuming a single failure. As noted above, certain additional monitoring and control equipment is to be added at the first refueling outage. Although this additional equipment may be useful, its absence does not prevent the operator from safely shutting down the plant. As a result, the plant's ability to reach cold shutdown using the remote shutdown system is not adversely affected by the granting of this exemption request.

Although operation with the proposed exemption would not have exactly the same margin of safety as would operation in full compliance with GDC 19 and the NRC Standard Review Plan, the staff is not required to deny an exemption if granting it would reduce a margin of safety by only an insignificant amount. As discussed in its Initial Decision (ASLBP No. 77-347-OIC-OL) dated October 29, 1984, the Atomic Safety and Licensing Board determined that "the question of 'as safe as' must be approached in a functional sense (does it serve the purpose of protecting public health and safety) rather than in an absolute sense (is it the very best possible machine available for the purpose)" (Initial Decision, p. 26). In this context, the Board found that if the standard to be used would justify a comparable level of protection, then the "as safe as" determination could be made. Therefore, based on this standard and the technical discussion presented above and in SSER 3 and SSER 7, the staff has concluded that operation of Shoreham until the first refueling outage without the additional remote shutdown system instrumentation discussed in SSER 3 would be as safe as operation in full compliance with the regulations. It is, therefore, acceptable.

As set forth in the Commission's decision in Shoreham (Long Island Lighting Company) (Shoreham Nuclear Power Station, Unit 1), CLI-84-8 (May 16, 1984), the Commission regards the use of the exemption authority under 10 CFR 50.12 as extraordinary. The availability of an exemption requires a finding of exigent circumstances that favor the granting of an exemption. Pursuant to the Commission's Shoreham decision, a determination as to whether exigent circumstances warrant an exemption should include a consideration of the stage of the facility's life, any financial and economic hardships, internal inconsistencies in the regulation, the applicant's good faith effort to comply with the regulation from which an exemption is sought, the public interest in adherence to the Commission's regulations, and the safety significance of the issues involved.

With regard to the stage of the facility's life, construction of Shoreham is complete and the facility is ready for fuel loading, low power testing, and eventual commercial operation. Absent the requested exemption and consequent authorization to load fuel, conduct low power testing, and generate power, the facility would remain idle, unused, and untested until the additional remote shutdown system instrumentation is installed and testing is complete. Thus, without the requested exemption, testing and eventual operation would be delayed. In this circumstance, the stage of the facility's life would appear to favor issuance of the exemption.

With regard to financial or economic hardship, in its October 29, 1984 Initial Decision (ASLBP No. 77-347-OIC-OL), the Atomic Safety and Licenisng Board noted that it is almost self-evident that there must be financial hardship to someone when there is a physically completed nuclear facility standing unused and nonproductive because of substantial licensing delays. If this exemption is not granted, the applicant will be subjected to financial and economic hardships. On the other hand, the staff has identified no financial or economic hardships that would result of the exemption were granted. Financial and economic considerations thus appear to favor issuance of the exemption.

No internal inconsistencies in the regulation are apparent, but the staff has noted that the applicant does not agree with the staff position that noncompliance with the single failure criterion in the Standard Review Plan constitutes noncompliance with GDC 19. This factor, therefore, appears to weigh slightly in favor of a finding of exigent circumstances and the issuance of the proposed exemption.

As to the applicant's good faith efforts to comply with GDC 19, a similar argument applies. The applicant considers that it does comply with the GDC, but to avoid unnecessary licensing delays, the applicant has agreed to modify the system to meet the staff concerns. The applicant negotiated the settlement agreement in good faith and has made significant progress in making the necessary modifications. Based on the staff agreement with the schedule for completing the modifications by the end of the first refueling outage, as stated in SSER 3, SSER 7, and the settlement agreement, the applicant cannot be faulted for not making a heroic effort to comply earlier. In these circumstances, the equities lie in favor of granting the exemption.

Finally, while the public interest favors adherence to the Commission's regulations, the staff has concluded that in this instance, where a limited and temporary exemption from compliance with GDC 19 for the first cycle of operation has no adverse safety significance (as noted above) and yet would allow the efficient and expeditious testing and operation of the facility, it is not contrary to the public interest to grant the requested exemption.

In accordance with the Commission's directions in <u>Shoreham</u> then, taking into account the equities of the situation, the staff finds that those equities weigh in favor of granting the requested exemption. In sum, the staff finds, based on the readiness of the facility for fuel loading, low power testing, and eventual commercial operation, the potential for adverse economic impacts absent an exemption, the applicant's good faith efforts at compliance with the regulations, and the lack of adverse safety significance or any detriment to the public interest from granting the requested exemption, that exigent circumstances exist that. favor the granting of an exemption under 10 CFR 50.12(a).

Based on the foregoing, and in accord with the Commission's decision on <u>Shoreham</u>, CLI-84-8 and 10 CFR 50.12(a), the staff has concluded that the partial exemption from the requirements of GDC 19, as discussed above, is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

7.5 <u>Safety-Related Display Instrumentation</u>

In Section 7.5 of the SER, the staff stated that the applicant would be expected to upgrade post-accident monitoring instrumentation in accordance with Revision 2 to RG 1.97. It was further stated that an evaluation of the new instrumentation proposed by the applicant in accordance with these requirements would be issued upon submittal of an acceptable design.

By letter dated April 14, 1984 (SNRC-863), the applicant responded to Generic Letter 82-33. The extent of Shoreham's conformance to RG 1.97 is discussed in that response.

In its submittal, the applicant took exception to some of the guidance of RG 1.97, Revision 2, with regard to neutron flux, reactor water level, reactor coolant system soluble boron concentration, drywell sump and drywell drain sump levels, primary containment isolation valve position, radiation level in circulating primary coolant, analysis of primary coolant radiation exposure rate, suppression chamber spray and drywell spray flows, suppression pool water level, core spray system flow, standby liquid control system (SLCS) storage tank level, reactor building area radiation, radiation and radioactivity in the plant environs, and primary coolant and sump grab sampling. The staff is reviewing the applicant's proposal, hence, the staff will condition the Shoreham license to require the applicant to implement any additional modifications that arise from that review before startup following the first refueling, unless prior approval of these exceptions is granted by the staff.

7.6 <u>Other Instrumentation and Control Systems Required for Safety</u>

7.6.6 Physical Independence

7.6.6.1 Physical Independence Within NSSS Cabinets

In Section 7.6.6.1 of SSER 4, the staff reported that the applicant had committed to review and verify that all redundant protective devices used in non-Class 1E circuits in panels in the nuclear steam supply system (NSSS) are appropriately sized and located for the worst fault condition postulated. The worst case conditions postulated would include open circuits, short circuits, transient conditions including electromagnetic interference, and the application of the maximum credible fault voltage line to line and line to ground.

In Inspection Report No. 50-322/84-14, dated July 16, 1984, the staff inspector reported the results of his inspection of Unresolved Item 81-17-01, concerning separation requirements for internal panel wiring. This report stated:

> To further insure that the electrical separation within the nuclear steam supply system (NSSS) panels installed at Shoreham are consistent with Shoreham's licensing commitments and in particular with IEEE-279-1971 an evaluation was performed by GE of suspected deviations from electrical separation requirement. This evaluation consisted of a detailed inspection of all Shoreham's NSSS panels by GE systems engineering personnel. This report ("NSSS Panel Design Evaluation for Electrical Separation 1E to [non-]1E Interface") concludes that the divisional separation of electrical wiring inside the NSSS panels is entirely adequate. This report does make one recommendation to further enhance the reliability of the standby liquid control system (SLCS). This recommendation to install redundant circuit protection in the wiring for both SLCS circuits will be incorporated by E&DCR No. P-4359.

The inspector reviewed the results of the applicant's study of the non-Class 1E circuits inside the NSSS cabinets and determined that the applicant had properly identified the additional redundant protective devices that had to be added, as well as other necessary modifications. The inspector reviewed the work packages for the modifications and further determined that all changes had been implemented, except for the SLCS control wiring. The completion of the modifications to the SLCS wiring was documented in Station Modification Package 83-048, dated July 27, 1983. This item is resolved.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.5 Overhead Heavy Load Handling System

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was developed. Following the issuance of NUREG-0612, a generic letter dated December 22, 1980 was sent to the licensees of all operating plants, all applicants for operating licenses, and all holders of construction permits requesting that they indicate the degree of their compliance with guidelines of NUREG-0612. The responses were to be made in two stages. The first response (Phase I, based on Section 5.1.1 of NUREG-0612) was to identify the load handling equipment within the scope of NUREG-0612 and describe the associated general load handling operations (such as safe load paths; procedures; operator training; special and general purpose lifting devices; the maintenance, testing, and repair of equipment; and the handling equipment specifications). The second response (Phase II) was to show that either single-failure-proof handling equipment was not needed or that single-failure-proof equipment has been provided. The following discussion and the Technical Evaluation Report (TER) reproduced as Appendix A to this SSER constitute the staff's evaluation of the applicant's Phase I response. Phase II will be evaluated in a future safety evaluation.

By letter dated November 11, 1983 (SNRC-980), the applicant stated that, as a minimum, the guidelines of Section 5.1.1 of NUREG-0612 (except for Guideline 4, "Special Lifting Devices") would be fully implemented before fuel load. By letter dated February 21, 1984 (SNRC-1007), the applicant stated that the modifications and testing required to meet Guideline 4 will be implemented before the beginning of the first scheduled refueling outage. The staff has reviewed the applicant's schedule for implementation and concludes that it is acceptable.

The staff and its consultant, EG&G Idaho, Inc. (EG&G), have reviewed the applicant's submittals. As a result of its review, EG&G has issued its TER (Appendix A). The staff has reviewed the TER and concurs with its findings that the guidelines of NUREG-0612 Section 5.1.1 have been satisfied. With the issuance of this TER, the staff concludes that Phase I of NUREG-0612 for Shoreham has been completed acceptably. Based on the improvements in heavy loads handling obtained from Phase I and the results of the Phase II pilot program at other selected power plants, which identified no further heavy loads concerns, the staff has determined that no additional Phase II work is necessary for Shoreham, and that the issue of control of heavy loads is resolved.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.1 General

In SSER 1 the staff provided its evaluation of the applicant's corporate and plant staff organizations that existed at that time (1981). Since then, the applicant has revised these organizations and has staffed many of the management positions with different people.

The staff has reviewed FSAR Revision 33, which was submitted in September 1984 and which included extensive revisions to FSAR Chapter 13. The staff's evaluation of the revised Chapter 13 is discussed below.

13.1.2 Corporate Organization

The corporate organization and the lines of responsibility for operation of the nuclear station are shown in Figures 13.1 and 13.2 of this supplement. (These figures are FSAR Figures 17.2.1-1 and 13.1.1-1, respectively.) The senior corporate officer in charge of LILCO's nuclear program is Mr. John D. Leonard, the Vice President, Nuclear Operations. He is responsible for all nuclear activities within LILCO related to the operation and maintenance of Shoreham. He reports directly to the Executive Vice President, who is responsible to the President of the company. Fossil plant operations and matters related to transmission of power are the responsibilities of others within the LILCO corporate organization, as shown in Figure 13.1.

Mr. Leonard's naval nuclear experience, spanning 12 years, includes service as a qualified engineering watch officer on a land-based prototype propulsion reactor and as commander of two nuclear-powered submarines. Following naval service, he worked for 15 months as corporate supervisor of operational quality assurance (QA) for a public utility that had two operating pressurized water reactors and two under construction. For the next 8 years, he was directly involved with the operation of the Fitzpatrick Nuclear Power Plant, a large BWR, as Resident Manager, Emergency Plan Director and, later, as Assistant Chief Engineer responsible for engineering assistance to the operating departments. He joined LILCO in May 1984 in his present position.

Reporting to Mr. Leonard are the Manager, Nuclear Operations Support Department; the Manager, Nuclear Engineering Department; and the Shoreham Plant Manager.

The corporate quality assurance activities, including those for Shoreham, are managed by the Director, Quality Assurance, Safety and Compliance, who reports directly to the Executive Vice President.

13.1.2.1 Nuclear Engineering

The Manager, Nuclear Engineering is Mr. Edward Youngling. He is responsible for providing the engineering, fuel management, and radiation protection technical support required to supplement the technical staff assigned permanently to the plant staff in support of the operation of Shoreham. Mr. Youngling, in his present position and in his previous positions with LILCO, has approximately 14 years of nuclear experience, primarily related to Shoreham design, construction, and preoperational testing activities. He received senior reactor operator (SRO) certification on the Duane Arnold Energy Center BWR while participating as shift engineer during startup and power ascension testing. He served previously as a test engineer on the S3G prototype naval reactor at Knolls Atomic Power Laboratory.

Reporting to Mr. Youngling, as shown in Figure 13.3 (FSAR Figure 13.1.1-2), are five divisions: Nuclear Systems Engineering, Nuclear Project Engineering, Radiation Protection, Engineering Assurance, and Nuclear Fuel.

These divisions will have as many staff specialists as required to support initial fuel loading and the safe operation of the plant. Technical design and evaluation expertise will be available in the areas of nuclear instrumentation, nuclear materials engineering, nuclear mechanical engineering, plant modifications, radiation protection and shielding, reactor physics, transient analysis, accident analysis, engineering assurance, and nuclear fuel management. According to the applicant, these five divisions have a present total staff of approximately 65 people.

13.1.2.2 Nuclear Operations Support

The Manager, Nuclear Operations Support, Mr. Jeffrey L. Smith, is responsible for six divisions as shown in Figure 13.4 (FSAR Figure 13.1.1-3). The applicant has stated that these divisions have a present total staff of approximately 130 people.

Mr. Smith has been employed by LILCO for the past 18 years, except for a 2-year period with the U.S. Army. His nuclear experience extends over the most recent 10-year period, in progressively more responsible positions involved with QA and regulatory matters, all related to Shoreham. He was certified as an SRO in the GE BWR Simulator Program in 1979.

13.1.2.3 Quality Assurance

The Director, Quality Assurance, Safety and Compliance is responsible to the Executive Vice President for the LILCO QA program, the Independent Safety Engineering Group (ISEG), and the Reliability Group. He also serves as Chairman of the Nuclear Review Board.

The FSAR does not describe the functions or staffing of the Reliability Group. However, at the staff's request, the applicant submitted, by letter dated November 12, 1984, a brief description of the functions and goals of the Reliability Group. The functions of the group, which is now being staffed, are:

(1) Ensure that reliability goals are established, in conjunction with Nuclear Operations, for nuclear plant systems and components to improve plant

reliability and ensure that programs for improvement in these goals are functional and operating.

(2) Ensure that these programs are continually monitored and evaluated relative to the nuclear plant and industry experience.

(3) Select and monitor plant performance indicators and develop recommendations for cost-effective programs to enhance equipment and plant reliability.

(4) Maintain a set of evaluation and computer software systems for tracking and cataloging failure, outage, and repair data for nuclear plant components and systems and for monitoring and evaluating reliability experience and maintenance trends and problems.

(5) Assist in establishing and maintaining a data base that is consistent with or part of the systems developed above to provide administrative and operations support.

(6) Perform critical reviews of data reports taken from the data base to ensure they properly describe events, accurately reflect all contributing factors, and duly indicate relative importance.

The ISEG and the Nuclear Review Board are discussed in Section 13.4 below. The staff's evaluation of the Shoreham QA program is in Section 17 of this SSER.

The Director, Quality Assurance, Safety and Compliance, is Mr. Robert A. Kubinak. Mr. Kubinak has been employed by LILCO since 1955. From then until 1969, he served in progressively more responsible positions in the operation and maintenance of LILCO's fossil-fueled power stations. From 1969 to 1978, he was Shoreham Plant Manager. During this period, he received SRO training and was assigned for 15 months to the Dresden Nuclear Station of Commonwealth Edison Company as a preoperational test engineer and shift engineer on the GE startup team for Dresden Units 2 and 3. He was certified as a reactor operator (RO) for both units. For the next 6 years he was Manager, Nuclear Operations Support.

13.1.2.4 <u>Other Technical Support</u>

The LILCO Vice President, Engineering and Administration has five engineering divisions plus computer and purchasing services reporting to him (see Figure 13.1). At least 10 engineering staff members have been designated whose first priority will be to respond to the needs of the Shoreham plant, as required. Their areas of expertise include mechanical, electrical, and environmental engineering plus design and drafting.

In addition, LILCO has committed to contracting with a qualified architect/engineer who will provide supplementary engineering support.

13.1.2.5 <u>Qualifications</u>

Of the corporate staff that manages the technical support for Shoreham operations, LILCO has provided resumes of only the Vice President, Nuclear Operations; the Managers, Nuclear Operations Support and Nuclear Engineering; and

Director, Quality Assurance, Safety and Compliance. Of these four, only the Vice President, Nuclear Operations has extensive experience with a large operating commercial BWR. The experience of the other three has been primarily with the design, construction, and preoperational testing of the Shoreham plant.

In SSER 1, the staff noted that it would require the then-Vice President, Nuclear to be assisted by an experienced advisor because that Vice President, Nuclear had no large commercial power reactor operational experience. The new Vice President, Nuclear Operations, Mr. Leonard, has significant operating experience, as noted in Section 13.1.2 of this report. Because of his experience, the staff concludes that an advisor to the Vice President, Nuclear Operations is not needed.

13.1.2.6 Conclusions

On the basis of its review of FSAR Section 13.1.1, Revision 33, the staff concludes that there is no longer a need for the applicant to provide an experienced advisor to the Vice President, Nuclear Operations. The staff further concludes that the applicant's corporate organization and staffing have been improved over those the staff reviewed in 1981, and can adequately support the operation of the Shoreham plant.

The staff evaluation in this area was based on Section 13.1.1 of the NRC Standard Review Plan (SRP, NUREG-0800) and on NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" (draft report dated September 1980).

13.1.3 Plant Staff Organization

The Shoreham Nuclear Power Station Operations Department, as shown on Figures 13.5 through 13.9 (FSAR Figures 13.1.2-1 through 13.1.2-5), consists of 12 main sections that report, through their respective section heads, to one of five Division Managers (Operations, Maintenance, Radiological Controls, Outage and Modifications, and Operations Staff). The Division Managers, in turn, report to the Plant Manager.

13.1.3.1 Operations Division

The Operations Division consists of the Operations and Reactor Engineering Sections (see Figure 13.6).

The Operating Engineer directs the activities of the Operations Section, which primarily consist of the routine operation of the station systems and equipment. The section will include a minimum of 32 supervisors and operators who are responsible for the operation of the station. A Watch Engineer will direct the operation of each shift through the Watch Supervisor, Nuclear Station Operator, and Nuclear Assistant Station Operator. The Watch Engineer reports to the Operating Engineer.

The Reactor Engineering Section is supervised by the Reactor Engineer. The Section will include a minimum of eight engineers and Shift Technical Advisors who will function in the areas of core physics, fuel management, post-refueling startup surveillance testing, and accident assessment/transient analysis.

13.1.3.2 Maintenance Division

The Maintenance Division is composed of the Instrument and Control, Computer Engineering, and Maintenance Sections (see Figure 13.8).

The Instrument and Control Section will have a minimum of 12 persons--engineers, a foreman, and technicians--who will be responsible for the calibration, maintenance, and testing of instruments and control systems.

The Computer Engineering Section, directed by the Computer Engineer, will repair, test, and maintain all hardware, software, and firmware associated with process, emergency, and administrative computer and teleprocessing systems. In addition to the Computer Engineer, the section will include a minimum of six engineers and technicians. Additional technicians may be used to supplement the computer work force as required.

The Maintenance Section will have a minimum of 26 people experienced in the mechanical and electrical maintenance of large steam-electric generating stations. The force will be supervised by the Maintenance Foreman who, in turn, will report to the Maintenance Engineer. This number of maintenance personnel is adequate for normal maintenance, but will be supplemented by additional competent maintenance personnel from other LILCO power stations or organizations, or outside contractors, as required.

13.1.3.3 Radiological Controls Division

The Radiological Controls Division consists of the Health Physics, Radiochemistry, and Radwaste Sections (see Figure 13.7).

The Health Physics Section, directed by the Health Physics Engineer, will have a minimum of 14 engineers and technicians to implement Shoreham's radiation protection program, including the preparation of radiation work permits, performance of radiological surveillance, maintenance of personnel exposure records, and calibration and maintenance of fixed and portable radiation detection instrumentation.

The Radiochemistry Section will consist of a minimum of 11 engineers and technicians. The Radiochemistry Engineer will supervise the section activities, such as detection and control of environmental releases, assessment of radiation doses to the public, and station chemical and radiochemical activities.

The Radwaste Engineering Section, directed by the Radwaste Engineer, will consist of a minimum of five engineers and technicians. The section will be responsible for the processing, handling, and preparation for shipment of all radioactive waste.

13.1.3.4 Outage and Modifications Division

As shown in Figure 13.9, the Outage and Modifications Division consists of the Modifications Engineering, Planning and Scheduling, and Outage Planning Sections, which will implement changes to plant systems and equipment that are required by regulatory agencies or that are designed to improve plant operation and reliability.

The Modifications Engineering Section, directed by the Modification Engineer, will request and implement station modifications, and coordinate post-modification retesting and return to service. This section is composed of the Modifications Engineer, engineers, and engineering aides.

The Planning and Scheduling Section, directed by the Planning and Scheduling Engineer, will plan and schedule plant maintenance activities. This section is composed of the Planning and Scheduling Engineer, engineers, engineering aides, planners, and clerical personnel.

The Outage Planning Section, directed by the Outage Planning Engineer, will coordinate the planning of scheduled plant outages. This section is composed of the Outage Planning Engineer and other engineers as assigned.

13.1.3.5 Operations Staff Division

The Operations Staff Division (see Figure 13.5) is composed of the Administrative and Operational Compliance Sections, which provide station administrative support and assurance that the station is in compliance with the requirements of the operating license.

The Plant Administrative Section, directed by the Administrative Coordinator, reports directly to the Operations Staff Manager. This section is responsible for the administration and direction of the office organization, including plant personnel records, plant filing system, office procedures, and reproduction equipment. The section administers the flow of correspondence, specifications, and drawings into and out of the plant.

The Operational Compliance Section, directed by the Operational Compliance Engineer, will implement the station surveillance programs, including leak rate testing, inservice testing, and snubber testing. The Operational Compliance Engineer will review surveillance activities to ensure compliance with the station's Technical Specifications. The Operational Compliance Section is composed of the Operational Compliance Engineer and other engineers as assigned.

13.1.3.6 Conclusions

On the basis of its review, the staff concludes that the plant staff organizational arrangement and staffing levels meet NUREG-0731 and the acceptance criteria of SRP 13.1.2/13.1.3 and are, therefore, acceptable.

13.1.4 Plant Staff Qualifications

FSAR Section 13.1.3, Revision 33, specifies the proposed qualification requirements for the plant operating and support staff. Section 13A of FSAR Revision 33 includes the resumes of the top five levels of plant management (the Plant Manager and the Managers of Operations, Maintenance, Radiological Controls, and Outage and Modifications).

The staff has evaluated the proposed requirements against those of SRP 13.1.2/ 13.1.3, which are, by endorsement of RG 1.8, the requirements of ANSI N18.1-1971. The staff has also evaluated the qualifications of the five managers listed above, as described in their resumes, against both ANSI N18.1-1971 and the applicant's proposed requirements.

Except for the requirements for the Operations Manager, the applicant's proposed qualification requirements meet the staff's criteria. Of the 8 years of responsible power plant experience required by ANSI N18.1-1971 for an Operations Manager, ANSI permits only 2 years of academic training to be substituted for 2 years of power plant experience. The applicant has proposed to permit a 4-year degree in engineering or a related scientific field to satisfy 4 years of the 8-year requirement. These proposed requirements do, however, meet the most recent standard, ANSI/ANS 3.1-1981, which is expected to be endorsed soon by the staff in a new revision to RG 1.8. Therefore, the staff concludes that the applicant's proposed qualification requirements for the Operations Manager are acceptable. However, the staff's initial review of the FSAR resume of the present Operations Manager, Mr. John A. Scalice, indicated that there was no evidence that Mr. Scalice had "participated in the management activities of an operating nuclear power plant during two months operation above 20% power," as is required by the applicant's proposed qualification requirements. (Mr. Scalice meets all the other requirements.) On the other hand, Mr. Scalice does meet all the requirements of ANSI N18.1-1971 for his present position.

By letter dated November 12, 1984, the applicant provided a revised resume for Mr. Scalice that indicates that he participated in the management activities of the Brunswick Steam Electric Plant during 2 months of operation above 20% power. Therefore, the staff concludes that Mr. Scalice meets all the requirements for the position of Operations Manager.

The staff's initial review of the FSAR resume of the Outage and Modifications Manager, Mr. Jack A. Notaro, indicated that Mr. Notaro met some but not all of the qualification requirements the applicant has proposed for this position. The resume did not provide evidence that Mr. Notaro had "participated in the management activities of an operating nuclear power plant during two months operation at 20 percent power, routine refueling outage (one to two months), and initial plant start-up testing or postrefueling outage start-up testing." However, by letter dated November 12, 1984, the applicant provided a revised resume for Mr. Notaro that indicates that he participated (at the Millstone, Dresden Unit 2, and Vermont Yankee nuclear power plants) in all of the activities noted above. Therefore, the staff concludes that Mr. Notaro meets all of the qualification requirements that the applicant has proposed for the position of Outage and Modifications Manager.

The applicant has proposed that neither the Plant Manager nor the Operations Manager will hold an SRO license, but that both will acquire the experience and training necessary to be examined for the SRO license. This is acceptable because the Operating Engineer, who is responsible for directing the actual day-to-day operation of the unit, will hold an SRO license. However, the applicant has not indicated that the "Engineer who reports to the Operating Engineer and acts as his alternate" (see Figure 13.6) will hold an SRO license. It is the staff's position that this Engineer should be SRO-licensed in order to be the alternate to the Operating Engineer and because, as stated in the FSAR, "his duties are similar to those of the Operating Engineer," and Figure 13.6 of this SSER shows the Watch Engineers reporting through the Engineer to the Operating Engineer. As a result of recent conversations between the staff and the applicant, the applicant has proposed, by letter dated November 12, 1984, that, until the Engineer (alternate) obtains an SRO license, the Operations Manager (SRO-licensed) will serve as the alternate to the Operating

Engineer, and the Engineer (alternate) position will not be in the line of command between the Operating Engineer and the Watch Engineers. The Shoreham Technical Specifications will include a station organization chart that reflects the noncommand function of the Engineer (alternate). The staff concludes that this is an acceptable arrangement.

The applicant did not provide, in the FSAR, the resume of the individual who fills the position of Operating Engineer. However, the applicant provided this resume by letter dated November 12, 1984. On the basis of its review of this resume, the staff concludes that the present Operating Engineer meets all the qualification requirements of that position.

The applicant has proposed that, if a Watch Engineer has 60 college credits (or a Watch Supervisor has 30 credits) in certain technical subjects (in addition to having the required high school diploma or equivalent), a Shift Technical Advisor (STA) need not be assigned to that Watch Engineer's or Watch Supervisor's shift. This proposed application of the "dual role" for the Watch Engineer has not been accepted by the Commission. Therefore, until the Commission takes final action on this generic issue, the staff will require LILCO to provide an STA for each operating shift. In a letter dated November 12, 1984, the applicant committed to meeting this requirement.

The applicant informed the staff by telephone on October 23, 1984, that there are 16 licensed senior operators (SROs) and 11 licensed reactor operators (ROs) assigned to the operating shifts. 10 CFR 50.54(m)(2) requires two SROs and two ROs on each shift during plant operation. It is apparent, therefore, that LILCO can fully staff six operating shifts, which is adequate to accommodate unexpected illnesses, vacations, and time for retraining. The applicant confirmed orally that six shifts are being used. In addition, LILCO stated that a total of eight other individuals on the plant staff are SRO-licensed, thus providing additional backup personnel for staffing the operating shifts, if necessary. An additional 10 shift ROs and 2 backup SROs are presently in training and are expected to be given NRC license examinations in February 1985. The staff concludes that there are sufficient licensed personnel for operation of the Shoreham plant.

In SSER 1, the staff noted the lack of substantive operating experience on BWRs exhibited by the plant operations staff. There have been some changes in the individuals occupying the top levels of plant management since SSER 1 was issued. The staff's evaluation of the experience of these individuals, as evidenced by their resumes provided in FSAR Revision 33, is given below.

The present Plant Manager had been the previous Chief Operating Engineer. His resume does not indicate any significant change in his BWR operating experience except for the 3 years of additional time preparing the Shoreham plant for operation.

The present Operations Manager had been Reactor Engineer at Shoreham. Prior to that, he had been involved, from 1970 to 1974, in LILCO's fossil plant operations and, from 1974 to 1979, in Shoreham design, construction, and startup. In 1980, he participated in a 16-week assignment at the Brunswick BWR nuclear plant during a refueling outage and post-refueling startup. He also participated, for 4 weeks in 1984, in management and planning of power operation at Brunswick.

The present Maintenance Manager has extensive equipment operation and maintenance experience on fossil-fueled power and propulsion plants. Since 1973, he has been the Maintenance Engineer for Shoreham. He received SRO certification in 1976 from the GE training program. He has no experience on an operating nuclear plant.

The present Outage and Modifications Manager has been at Shoreham since 1973. He has been QA Engineer for 5 years, Operating Engineer for 5 years, and Chief Operating Engineer for 1 year. His training assignments include approximately 7 months at other BWRs that were either operating at normal power levels or were shutting down for and starting up from refueling or other outages. He received an SRO license in 1982.

The Radiological Controls Manager has been assigned to Shoreham since 1973. Prior to that he had been involved with LILCO's fossil power plants since 1966. While at Shoreham, he spent 4 months at the U.S. Department of Energy's Savannah River Plant, receiving operational training in environmental monitoring, waste management, reactor shutdown radiation control, emergency operations, dosimetry, and analytical laboratory work. He also spent 8 months at the Dresden nuclear station in chemical and radiochemical work and in routine health physics activities.

On the basis of the experience discussed above, the staff determined that there was little BWR operating experience represented in the Shoreham plant's management. On that basis, and because the earlier Vice President, Nuclear had no significant nuclear experience, the staff had required, as delineated in SSER 1, that the applicant provide additional operating experience in the form of an advisor to the Plant Manager or Operations Manager. However, the new Vice President, Nuclear Operations has extensive BWR operating experience, as noted in Section 13.1.2 of this SSER, and he is located on the site. He and the Plant Manager are in close contact daily, so that the Plant Manager has direct benefit from the Vice President's experience. The Plant Manager's scope of responsibilities has been narrowed. For example, he is no longer responsible for quality assurance, security, and training. Regional personnel who are familiar with the Shoreham organization and personnel believe that the present Plant Manager has matured in his previous and present positions and is capable of performing his functions without the need of an advisor. Therefore, the staff concludes that there is no longer a need for the applicant to provide an experienced advisor to the Plant Manager or the Operations Manager.

13.1.5 Shift Crew Composition

The Commission has recently become concerned about the possible lack of hot operating experience among the operators on shift at newly licensed nuclear power plants. This has led to an evaluation of (1) the operating experience on shift proposed by the applicant and (2) the interim use of shift advisors to supplement the operating shift crews.

13.1.5.1 Operating Experience on Shift

Dialogue with the industry was begun in late 1982 to find a way of ensuring that each operating shift at a newly licensed plant had at least one senior operator with previous hot operating experience. On February 24, 1984, an industry working group representing utilities with nuclear power plants under construction or ready for operation presented a proposal to the Commission on the amount of previous operating experience considered to be the minimum desirable on each shift and how that experience could be obtained. On June 14, 1984, the Commission accepted the industry proposal with certain clarifications. Information regarding the Commission action was forwarded to the industry as Generic Letter 84-16, dated June 27, 1984. The objective is that, at the time of fuel load, each operating shift will have at least one senior operator with a minimum of 6 months of hot operating experience, including startup/shutdown experience and at least 6 weeks experience above 20% power. However, for plants in the late stages of licensing with insufficient time to meet the objective, the temporary use of experienced shift advisors is acceptable. The minimum qualifications for shift advisors are 4 years of power plant experience (including 2 years of nuclear power plant experience) and 1 year of hot operating experience as a senior reactor operator (or reactor operator, if found suitably qualified) on a large commercial nuclear power plant of the same type. All shift advisors are to be trained on the systems, procedures, and Technical Specifications of the plant for which they are to provide advice, and they are to be certified to the NRC as being qualified to act as shift advisors.

The applicant has selected an initial group of four individuals to act as Shoreham shift advisors. (The applicant uses the term onshift advisors.) These individuals have all had experience as senior reactor operators at operating boiling water reactors. The staff has reviewed the qualifications of the four proposed shift advisors and the plant-specific training provided to them at Shoreham and concludes that, subject to certain clarifications, they meet the shift advisor guidelines adopted by the Commission. These clarifications are discussed in Section 2a below.

In addition, because Shoreham does not now have senior reactor operators on each shift who meet the minimum guidelines for hot operating experience, the staff will condition the operating license to require shift advisors until the requisite experience has been obtained.

13.1.5.2 Shift Advisor Program

By letters dated July 2, 1984; July 25, 1984; September 5, 1984; September 6, 1984; and September 20, 1984 the applicant submitted information regarding the shift advisor program. The staff has reviewed this information for conformance to Generic Letter 84-16. In performing the review, the staff used additional information regarding qualifications and training of shift advisors that was developed during the staff review of shift advisor programs at several other plants.

The staff review of the Shoreham shift advisor program comprised four main areas: shift advisor qualifications, the training program for shift advisors (including written and oral examinations), the procedure used to define shift advisor duties and responsibilities, and other requirements pertaining to the use of shift advisors. These are discussed below.

(1) Shift Advisor Qualifications

Three of the current group of four individuals amply meet the shift advisor experience requirements of the industry working group proposal of February 24, 1984 as clarified by the Commission on June 14, 1984. All four have more than 4 years of equivalent nuclear power plant experience, and three of the four have had well over 1 year on shift as a senior reactor operator at a large operating BWR. The fourth individual has approximately 6 months of experience as a senior reactor operator, 16 months of experience as a reactor operator, and 3 years of experience as an unlicensed auxiliary operator at a large operating BWR. By virtue of their considerable onshift operating experience, all four individuals are considered qualified to participate in the Shoreham shift advisor program.

Although the current group of Shoreham shift advisors is considered qualified (pending satisfactory completion of oral examinations), the staff is concerned that there are six shift crews and only four shift advisors. Conducting the startup program using three 8-hour shifts per day will place the fourth shift advisor in a permanent "relief" status. The staff prefers to see at least as many shift advisors as shift crews because this allows each advisor to be assigned continuously to the same crew and makes it much easier to accommodate shift advisor sick leave, vacation time, and working hour limitations (the working hour limitations in the Technical Specifications would apply to shift advisors just as they apply to licensed operators and other individuals who perform safety-related functions). It is also the staff's position that the shift advisors should participate in the licensed operator requalification program. The use of only four advisors to cover three shifts per day would not appear to permit this.

The staff has concluded that the applicant should take steps to increase the number of shift advisors to at least five and to have them participate in the licensed operator requalification program as much as possible. The staff understands that a fifth advisor has already begun the training program and that the applicant has access to additional qualified personnel. The applicant is further encouraged to increase the number of shift advisors to at least six so that they can be assigned continuously to the same operating crew. The staff intends to monitor the applicant's performance in this regard.

(2) Shift Advisor Training Program

The initial Shoreham shift advisor training program was comprised of approximately 4 weeks of classroom training covering plant systems, emergency operating procedures, administrative procedures, and Technical Specifications. At the end of the classroom training, comprehensive written and oral examinations were given. The staff concludes that the classroom portion of the training program was adequate to familiarize an experienced operator with the Shoreham facility.

The facility-administered certification examinations (written and oral) were monitored by an examiner from Region I. The content and conduct of the final written examination were found to be excellent. The examiner reviewed the grading and parallel graded some random questions to verify the results. There were no significant deviations between the grades awarded by the facility examiners and by the Region I examiner.

The final oral examination showed that, although the shift advisors seem to have adequate knowledge of Shoreham systems, they did not have adequate familiarity with Shoreham-specific procedures, alarms, indications, reference material, or control room boards. The shortcomings are believed to be the result of a lack of sufficient control room time for these individuals.

The applicant has proposed to correct this deficiency by placing the initial group of shift advisors in the control room for at least one complete shift cycle (day, swing, and mid-shift). Upon completion of the additional control room training, the oral examination board will be reconvened and will include a Watch Engineer (senior reactor operator) and representation from Operations Division management and Training Division management. Before administering the oral examinations, the problems to be posed will be submitted to the staff. The initial group of four shift advisors will be re-examined, and the staff will be invited to send observers. In addition, before 5% power is exceeded, each shift advisor will complete a Shift Advisor Qualification Guide developed by the applicant to ensure that the advisor is familiar with the Shoreham control room.

The Shoreham shift advisor training program, supplemented by the applicant's commitments described in the preceding paragraph, meets the training guidelines of the Industry Working Group as clarified by the Commission on June 14, 1984. Prior to operation above 5% power, the applicant should certify to the staff the names of advisors who have been examined and determined to be competent to provide advice to operating shifts. Certification to the staff of the initial group of shift advisors will be made a condition of the license.

(3) Shift Advisor Procedures

The duties and responsibilities of the shift advisor are in Shoreham Temporary Procedure 21.001.02. This procedure establishes training and qualification criteria, log-keeping and shift-turnover requirements, and detailed duties and responsibilities associated with the shift advisor position.

Shift advisor responsibilities given in Section 8.5 of TP 21.001.02 include

- recommending reactor shutdown or other action when considered prudent (procedural limitations preclude the shift advisor from directing activities that require an operator's license)
- staying aware of plant conditions and their relationship to Technical Specifications
- providing advice on planned evolutions

- reviewing shift turnover sheets, various logs, and other documents for completeness and accuracy

- pursuing the resolution of disagreements over the proper course of action

The staff agrees with the limitation that restricts the shift advisor from directing licensed activities. The staff particularly agrees that the shift advisor should recommend reactor shutdown when such action is considered prudent and should pursue the resolution of disagreements about the proper course of action to be taken in specific situations.

The staff understands that procedure TP 21.001.02 has been or will be revised and resubmitted. As part of its license condition, the staff will require that the shift advisors and operating crews be trained in the revised procedure before initial criticality.

(4) Additional Shift Advisor Requirements

The applicant has made the following additional commitments, which will strengthen the effectiveness of the shift advisor program:

- The applicant has committed to a monthly evaluation by the Operations Division of the performance of each shift advisor. A procedure on shift advisor performance evaluations will be issued.

- The applicant has committed to a monthly evaluation of each operating shift crew by a shift advisor.

- The applicant's shift advisors will be medically qualified in accordance with 10 CFR 55. Shift advisors possessing a current operator's license at the time of contract or employment with Long Island Lighting Company will be understood to meet the requirements.

13.1.6 Conclusions

On the basis of its review, the staff concludes that the applicant's FSAR commitments, as modified by its November 12, 1984, letter, are acceptable in regard to the staffing of the Shoreham plant organization.

13.2 Training

13.2.1 Requalification Program for Licensed Personnel

In SSER 1, the staff stated that the applicant's plans for requalification training would conform to the requirements of Appendix A to Title 10 of the Code of Federal Regulations Part 55 (10 CFR 55). In addition, the applicant would provide responses to the TMI Action Items.

By letter dated August 2, 1983 (SNRC-942), the applicant submitted a Licensed Operator Requalification Program. The staff reviewed the program and, by letter dated March 14, 1984, requested additional information. In letters of

July 31 and August 31, 1984, the applicant provided a response and resubmitted the Requalification Program.

During its review, the staff has used Appendix A to 10 CFR 55, NUREG-0737, Standard 3.1 of the American National Standards Institute/American Nuclear Society (1978 and 1981, "Selection, Qualification and Training of Personnel of Nuclear Power Plants"), Regulatory Guide 1.149 (Nuclear Power Plant Simulators for Use in Operator Training), Generic Letter 83-17, and NUREG-1021.

The applicant has modified the Licensed Operator Requalification Program (SP 12.0104.07, Revision 3) in the following areas:

Section 8.1.4: Waivers for attendance for specific retraining will be granted for an individual whose grade in the area is equal to or greater than 80% on the annual exam.

The staff finds this change acceptable to meet the conditions of Section 2 of Appendix A to 10 CFR 55.

Section 8.2.1: The applicant has combined lecture series into one section to include applicable subjects in Section 2 of Appendix A to 10 CFR 55 and subjects addressed in NUREG-0737. However, the applicant has omitted related nuclear industry operating experience and changes to facility design and license, which were in Revision 1 of SP 12.014.07.

The staff finds that the change made in response to the staff's request is acceptable; however, the staff recommends reinstating the review of significant industry events and changes in facility design in the lecture series when appropriate (see Item I.C.5, NUREG-0737).

Section 8.2.4: Revision 3 of SP 12.014.07 includes participation in plant drill scenarios each training week. In Revision of SP 12.014.07, the applicant had committed to three plant drill scenarios each training week.

The staff concurs with this modification, which allows flexiblity in conducting drill scenarios.

Section 8.3.1: The applicant has referenced Appendix 12.2 of the revised program, which describes the control manipulations required by Section 3a of Appendix A of 10 CFR 55, as modified by NUREG-0737 (H.R. Denton letter of March 28, 1980), and includes evaluations of performance during simulator training. Appendix 12.2 has been modified to include additional control manipulations that the applicant has determined to be desirable for operator experience.

The staff concludes that these changes clarify control manipulation and evaluations. The additional control manipulations are expected to provide additional experience in potential plant transients.

Section 8.3.2.1: The applicant has clarified methods to keep personnel informed of plant design, procedural, and license changes.

Section 8.3.2.3: In response to the staff's request, the applicant has added to the drill scenarios the participation of all licensed evaluators as responders and has included the objectives of plant drills in the drill scenarios.

The applicant has provided a satisfactory response to the staff request.

Section 8.3.2.4: In its request of March 14, 1984, the staff stated that a regulatory guide would be issued that could apply to backup operators. Although the guide has not been published, the applicant has stated an intention to review the guide when it is issued and modify the program as necessary.

Section 8.4: The applicant has incorporated the elements of Generic Letter 83-17 in the requalification program.

Section 8.4.1: The applicant has added a section that provides for an annual performance evaluation and incorporated accelerated retraining when required.

This change complies with Sections 4c and 4e of Appendix A to 10 CFR 55.

Section 8.4.5: The applicant has limited the number of waivers for those licensed individuals who prepare and grade annual examinations.

Section 11: The applicant has included additional references that were contained in the staff's March 14, 1984 request for additional information.

The applicant has also provided information related to participation of licensed instructors in the requalification program and stated that the simulator meets Regulatory Guide 1.149. The staff finds these responses acceptable.

With these modifications, the staff concludes that the applicant's requalification program submitted on August 31, 1984, meets the conditions in Appendix A of 10 CFR 55 and in Item I.A.2.1. of NUREG-0737. Therefore, this item is resolved.

13.4 Review and Audit

The applicant has made and is making provisions for the review and audit of plant safety-related activities, as detailed below.

13.4.1 Review of Operations Committee

The Review of Operations Committee (ROC) has been established and is functioning. The Operations Manager is the Chairman and a committee member. The Maintenance Manager is an Alternate Chairman and a member. The Plant Manager is an Alternate Chairman but, otherwise, is not a committee member. The other committee members are the Operating Engineer, Reactor Engineer, Maintenance Engineer, Instrument and Control Engineer, Radiological Controls Manager, and the Health Physics Engineer.

The functions of the ROC are specified in the Shoreham Technical Specifications, which are based on the Standard Technical Specifications.

Although SSER 1 stated that the FSAR should be revised to indicate that the applicant would utilize the 1976 version of ANSI N18.7/ANS 3.2, FSAR Revision 33 still refers (in Section 13.4.2) to the 1972 version. However, by letter dated November 12, 1984, the applicant committed to meeting the 1976 version and stated that the FSAR would be revised accordingly.

The staff concludes that the applicant's commitment for the ROC is acceptable.

13.4.2 Nuclear Review Board

FSAR Revision 33 provides a description of the proposed organization, functions, and responsibilities of the Nuclear Review Board (NRB).

The staff's review of this information indicates that the provisions for independent review by the NRB do not meet the staff's acceptance criteria in SRP 13.4. The acceptance criteria require that provisions for independent review should meet those described in Section 4.3 of ANSI N18.7-1976 (ANS 3.2), and the qualification requirements for those performing these reviews should meet or exceed those described in Section 4.7 of ANSI/ANS 3.1-1978 and RG 1.8, Revision 1-R.

The staff has several major areas of concern. The applicant has not specified the membership of the NRB or whether it will be a standing committee or an organizational unit. The review area competency does not include nondestructive testing. The minimum qualifications of the NRB Chairman should include at least 3 years of nuclear power plant experience. Conference telephone calls should not substitute for face-to-face meetings. There is no indication that the NRB membership will include at least one individual from outside LILCO's or its contractors' organizations and at least one individual with substantial BWR operating experience. (This latter requirement was noted by the staff in SSER 1.)

The staff has been working with the applicant in the preparation of the Administrative Controls section of the Shoreham Technical Specifications, which are based on the staff's Standard Technical Specifications. Before an OL is issued, the Administrative Controls section will include specifications, required by the staff, that resolve the staff's concerns.

13.4.3 Independent Safety Engineering Group (ISEG)

The applicant has revised slightly the FSAR discussion of the ISEG from that addressed by the staff in SSER 1.

The ISEG is established and functional. It is composed of a minimum of five dedicated multidiscipline full-time engineers (one supervisor and four engineers) located on the site. Each has a bachelor's degree in engineering or a related science and at least 2 years of professional experience in his field, with at least 1 year of experience in the nuclear field. The ISEG is functionally responsible to the Director, Quality Assurance, Safety and Compliance, as shown in Figure 13.2.

The principal functions of the ISEG include:

(1) assessment of the operating experience of the station and stations of similar design

(2) examination of appropriate plant operating characteristics and industry/NRC issuances

(3) review of plant activities, such as maintenance, modifications, operational problems, and operational analysis

(4) surveillance of plant operations and maintenance activities to verify that these activities are performed correctly and with minimum human error

(5) review of other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety

(6) where useful improvements can be achieved, development and presentation of detailed recommendations in such areas as revised procedures or equipment modifications, to corporate management through the Director, Quality Assurance, Safety and Compliance.

On the basis of its review of FSAR Revision 33, the staff concludes that the proposed ISEG meets the requirements of NUREG-0737 Item I.B.1.2 and is, therefore, acceptable.

13.4.4 Conclusions

On the basis of its review, the staff concludes that the applicant's commitment for the ROC is acceptable. Adequate provisions for independent review by the NRB will be established by the staff in the Shoreham Technical Specifications, and the proposed ISEG meets the requirements of Item I.B.1.2 of NUREG-0737. Therefore, the staff finds the applicant's provisions for the review and audit of plant safety-related activities acceptable.

13.5 Plant Procedures

13.5.1 Administrative Control Procedures

The Licensed Operator Staffing Rule in 10 CFR 50.54(m)(2)(iii) requires that at least one senior operator be present in the control room at all times when a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the Technical Specifications. Additionally, the senior operator in the control room is expected normally to spend most of the time in that portion of the control room where there is direct and prompt access to information on current plant conditions and where the operator at the controls can be supervised.

The senior operator should have the flexibility to move periodically for a brief period of time to other parts of the control room as long as the senior operator is at all times within the vital security area of the control room and either is in sight of or in the audible range of the reactor operator(s) at the controls, or is in the audible range of the control room annunciators.

The staff has reviewed recent OL applications with the above consideration in mind, and has included appropriate limitations in the Technical Specifications for some of these plants.

By letter dated November 15, 1984, the applicant provided the following statement:

> Administrative procedures are in place which require either the Watch Engineer or the Watch Supervisor (licensed SROs) to be in possession of the control room command function. The individual in possession of the control room command function is within the physical confines of the Control Room and within sight or audible range of the Operator "at the controls" or in audible range of the Control Room annunciators.

The staff consiuers this an acceptable commitment for ensuring that the senior operator in the control room will remain in areas from which appropriate supervision can be maintained.

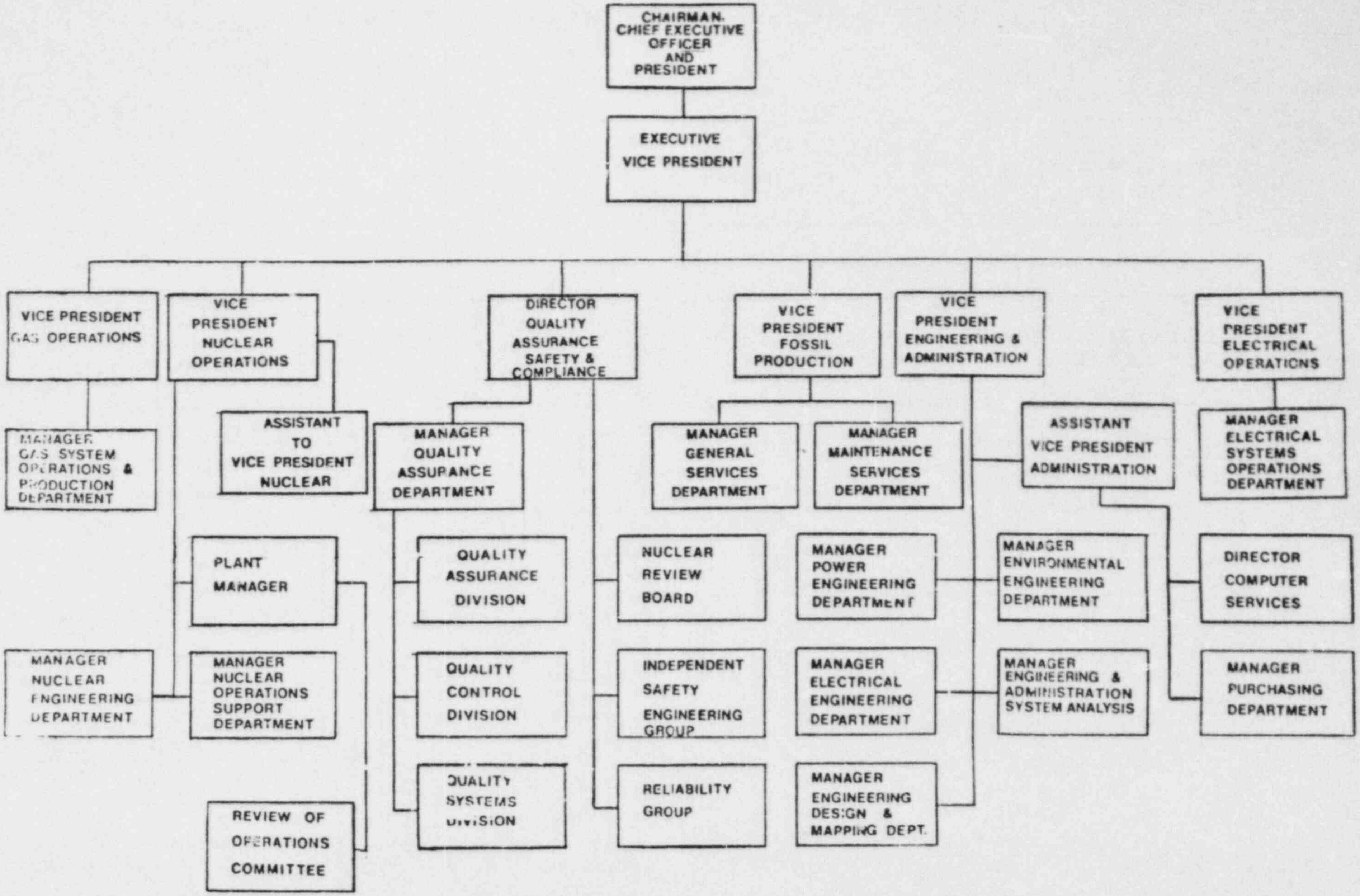

Figure 13.1 LILCO organization for quality assurance, Shoreham Unit 1

Source: FSAR Figure 17.2.1-1

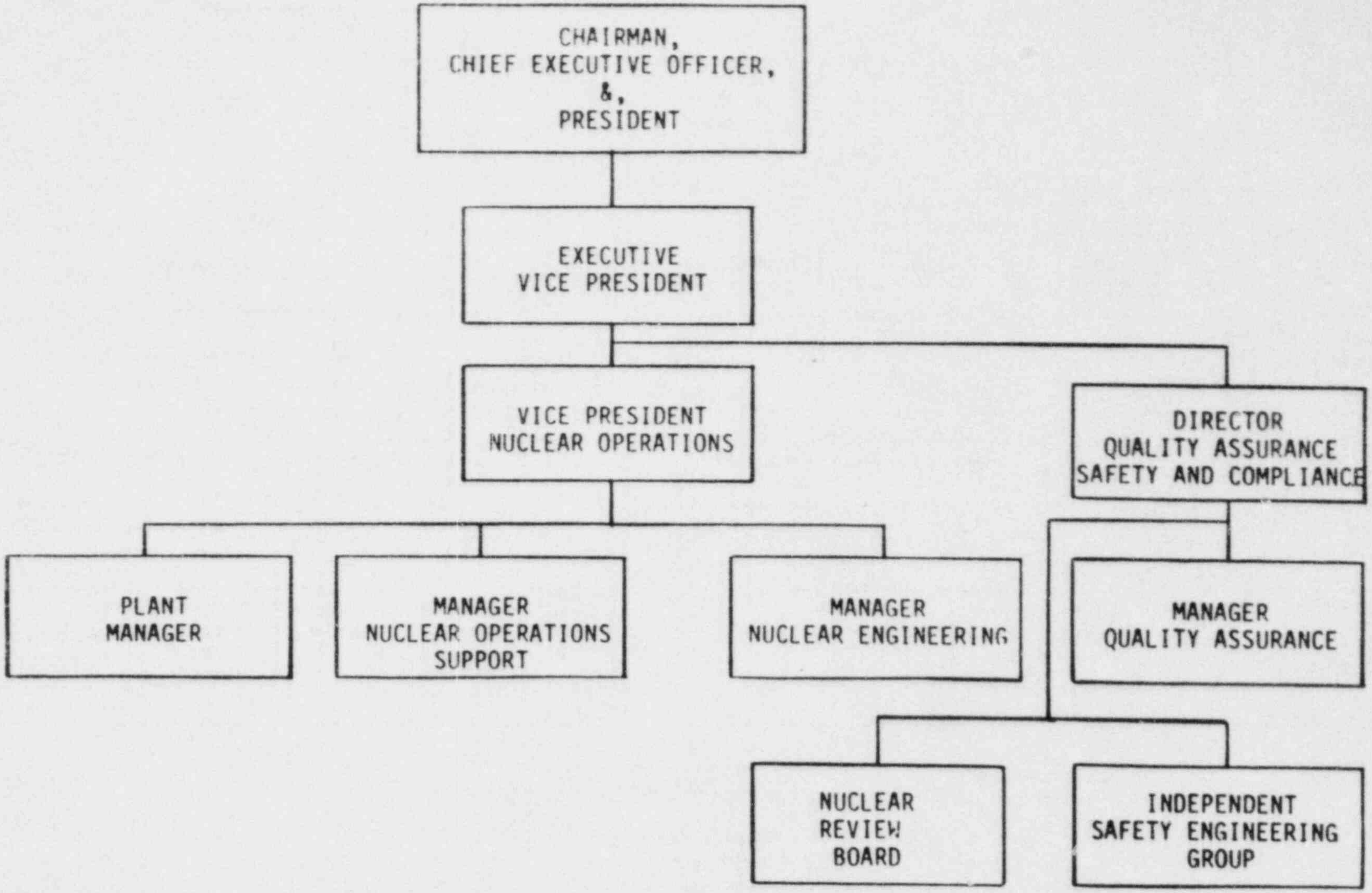

Figure 13.2 Executive responsibility for Shoreham Unit 1

Source: FSAR Figure 13.1.1-1

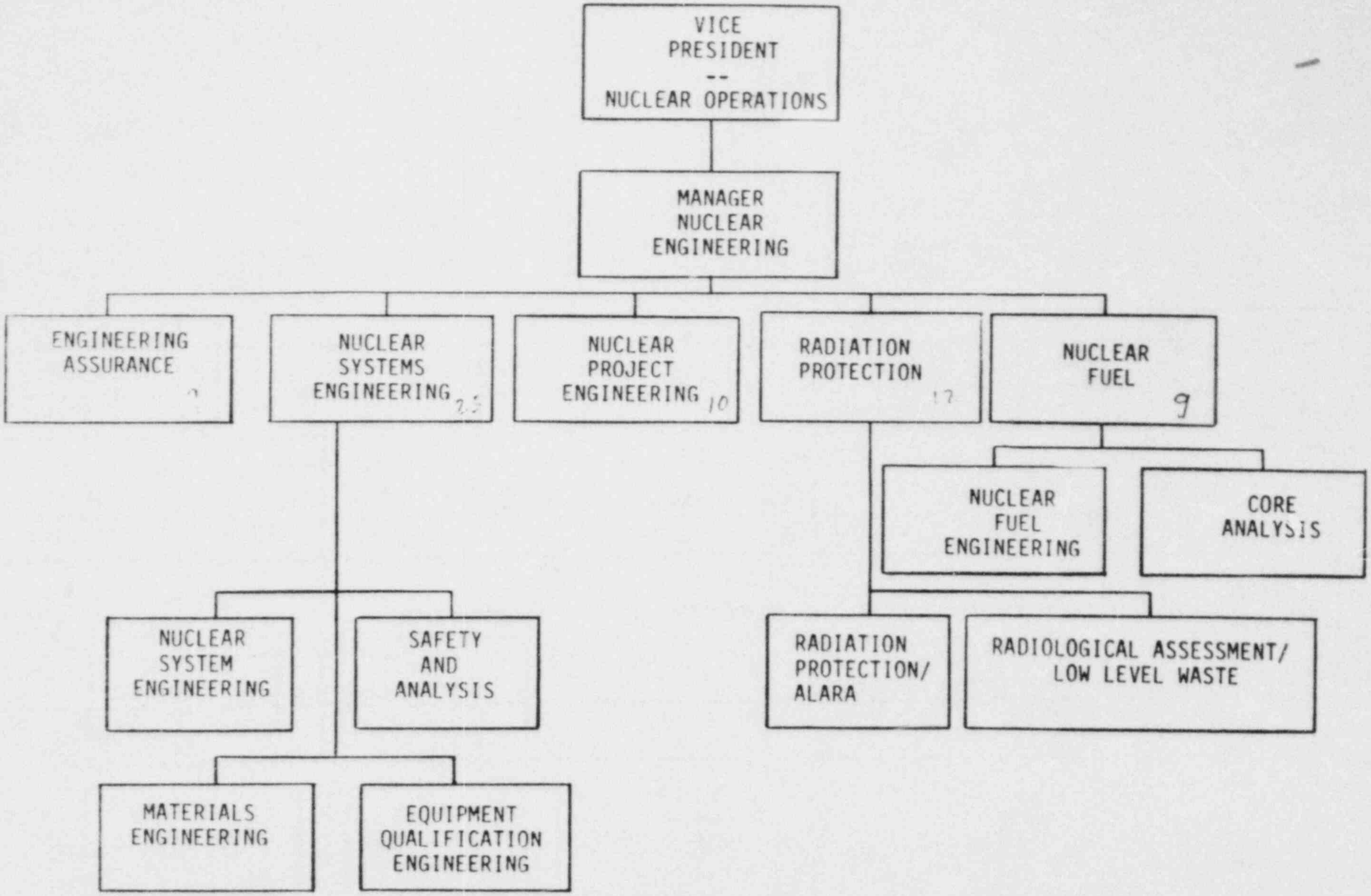

Figure 13.3 Nuclear Engineering Department

Source: FSAR Figure 13.1.1-2

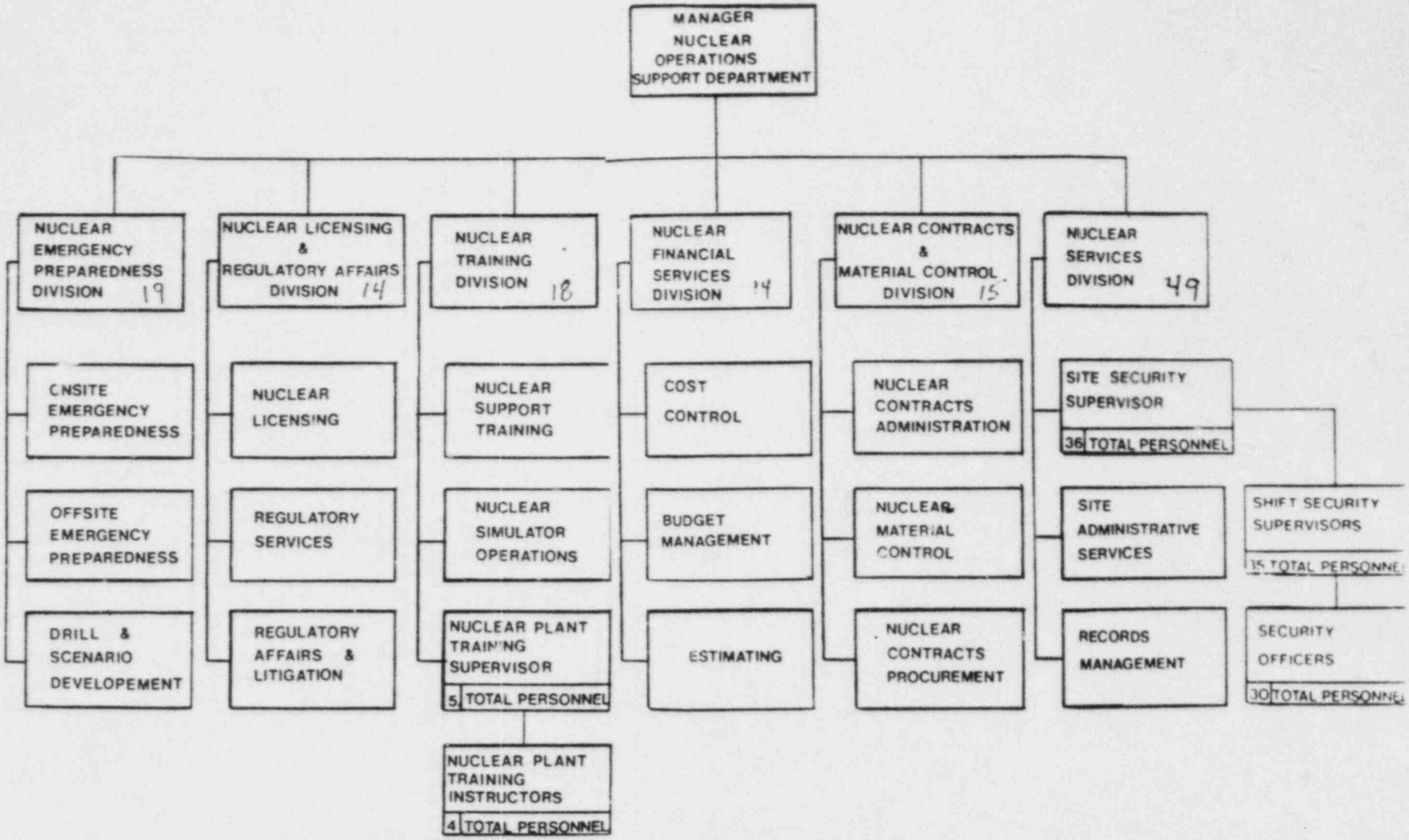

Figure 13.4 Nuclear Operations Support Department

Source: FSAR Figure 13.1.1-3

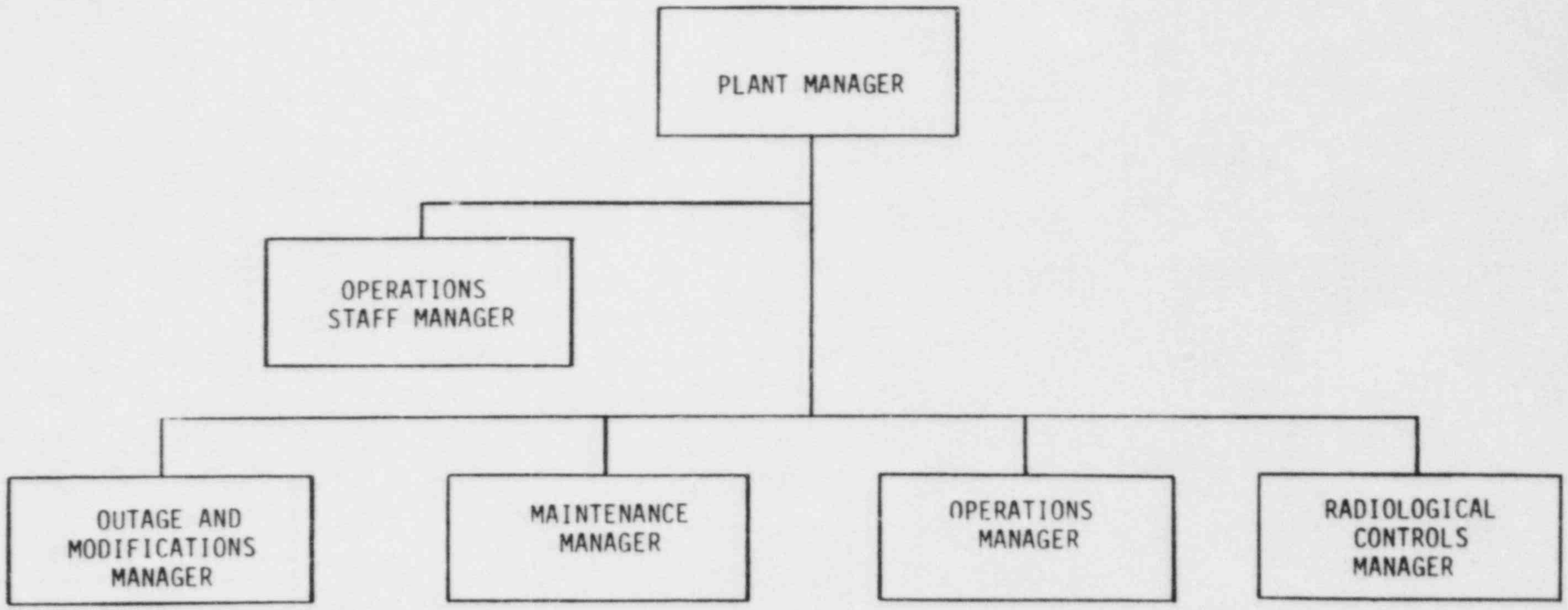

Figure 13.5 Shoreham plant staff

Source: FSAR Figure 13.1.2-1

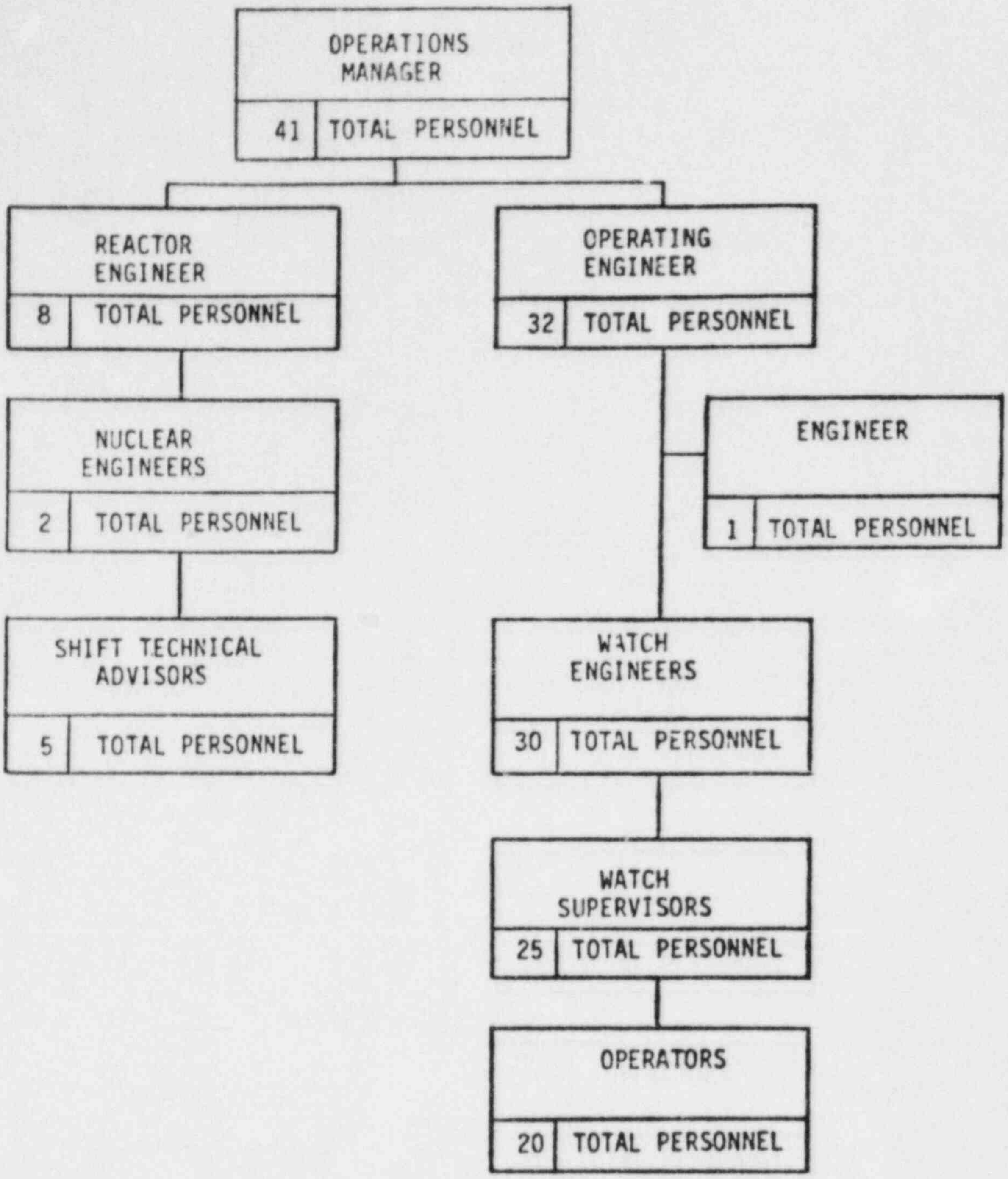

Figure 13.6 Operations Division

Source: FSAR Figure 13.1.2-2

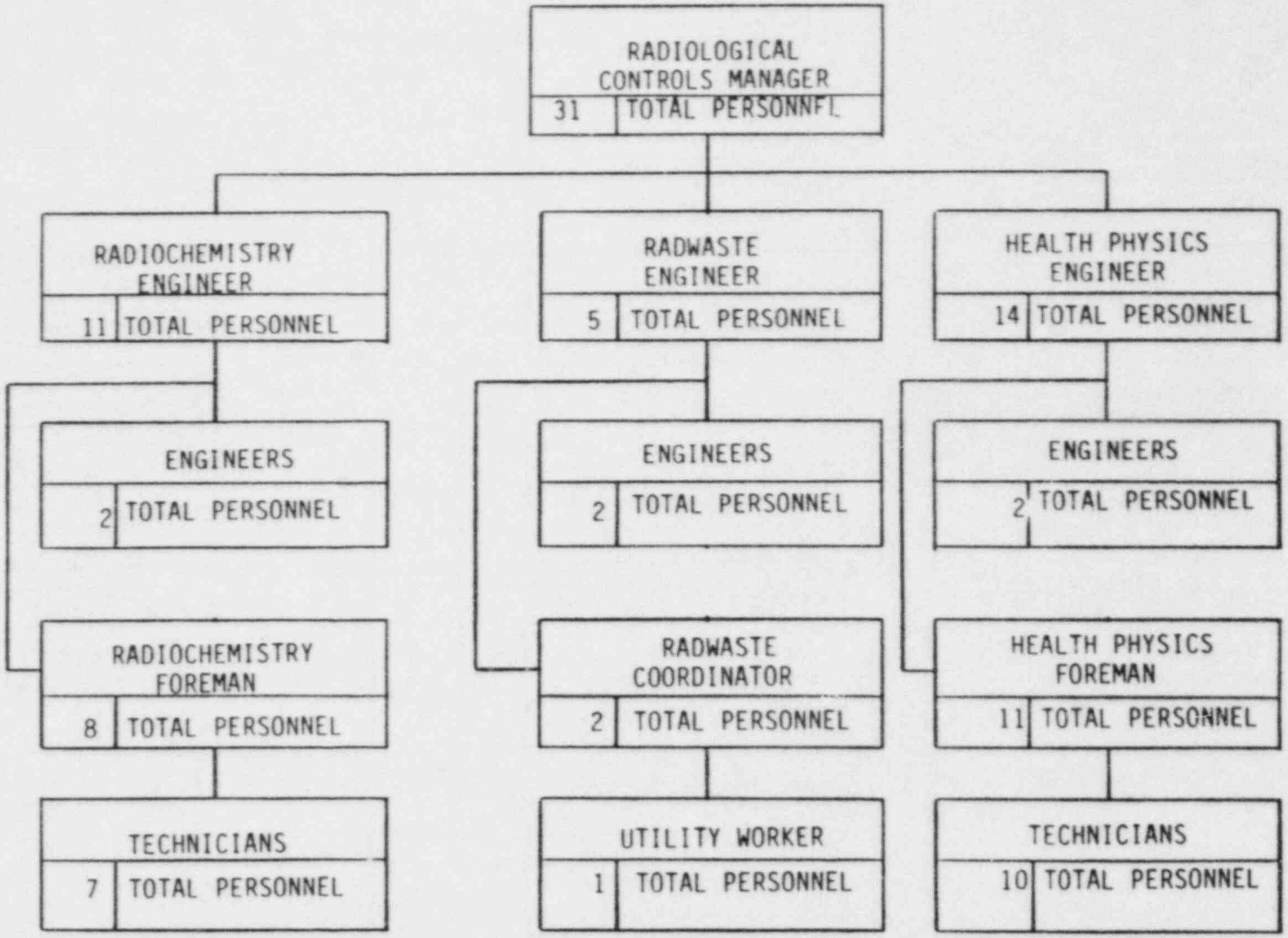

Figure 13.7 Radiological Controls Division

Source: FSAR Figure 13.1.2-3

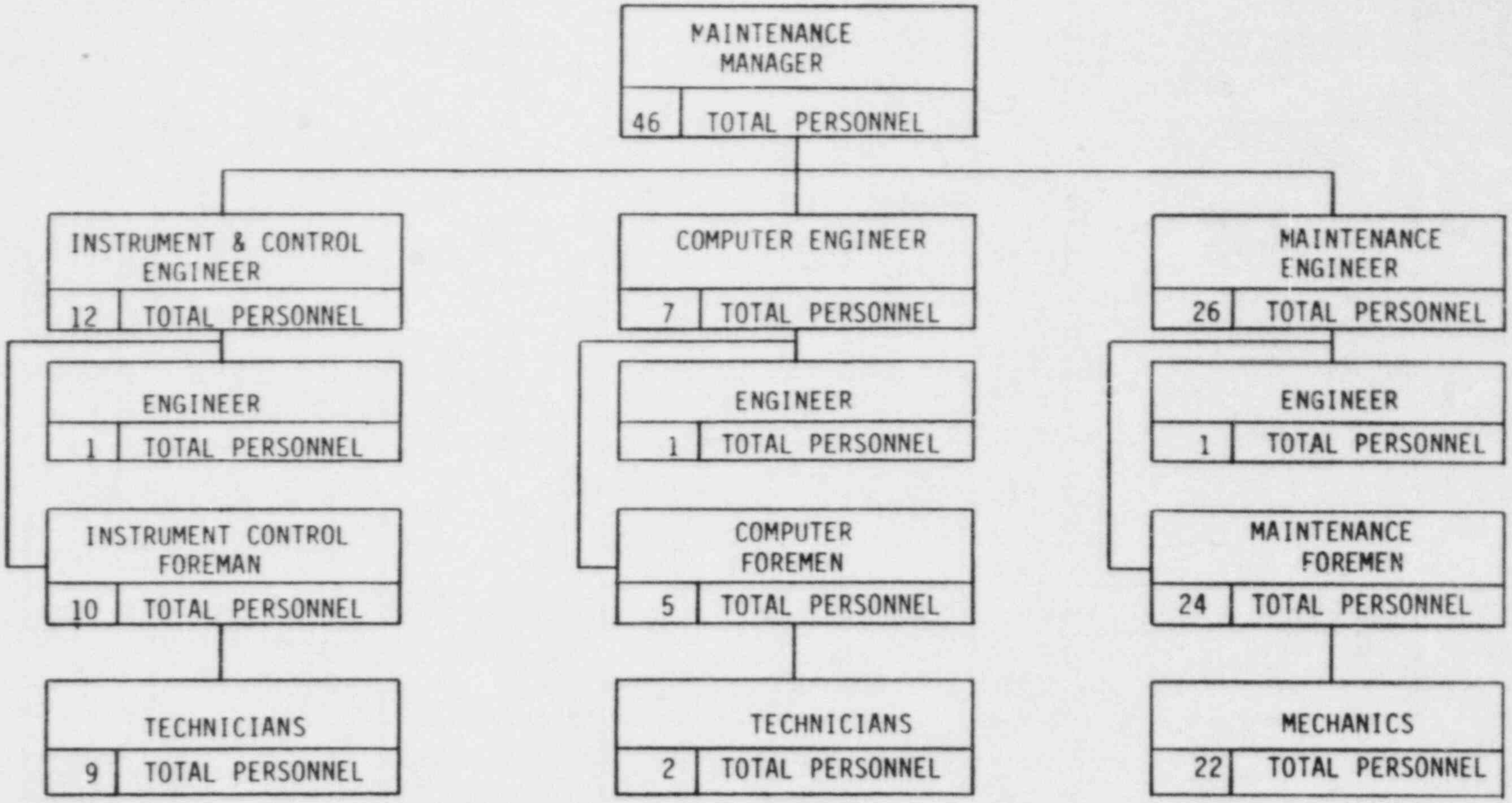

Figure 13.8 Maintenance Division

Source: FSAR Figure 13.1.2-4

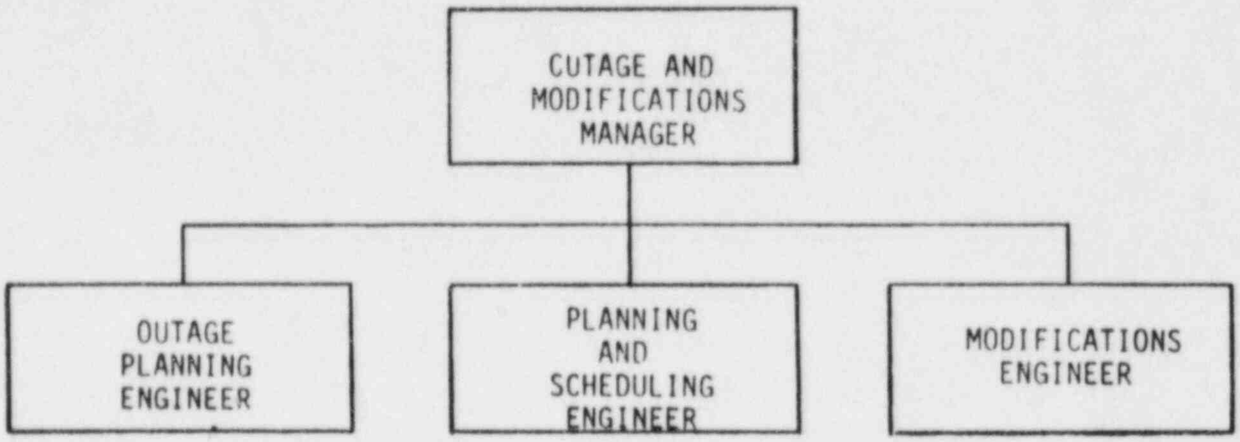

Figure 13.9 Outage and Modifications Division

Source: FSAR Figure 13.1.2-5

17 QUALITY ASSURANCE

17.1 General

The description of the quality assurance (QA) program for the operations phase of the Shoreham Nuclear Power Station is in Section 17.2 of the Final Safety Analysis Report (FSAR). This evaluation of the QA program is based on the staff review of Revision 33 of the Shoreham FSAR and discussions with representatives of the applicant and updates the evaluations published in the SER, SSER 1, and SSER 7.

The staff assessed the applicant's QA program for operations phase to determine whether it complies with the requirements of Appendix B to 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"; the applicable QA-related regulatory guides and ANSI standards listed in Table 17-1; and Standard Review Plan Section 17.2, "Quality Assurance During the Operations Phase."

17.2 Organization for the Quality Assurance Program

The structure of the organization responsible for the operation of Shoreham and for the establishment and execution of the operations phase QA program is shown in Figure 17.1. The Vice President, Nuclear Operations, who reports to the Executive Vice President, has overall responsibility for the engineering, modification, licensing, testing, startup, operation, and maintenance of the plant. This responsibility includes ensuring that organizations and personnel under his jurisdiction comply with the applicant's QA program requirements in the performance of their duties.

The Plant Manager of the Shoreham Nuclear Power Station reports to the Vice President, Nuclear Operations and is responsible for enforcing, within the station, those QA program requirements applicable to station functions and duties.

The Director, Office of Quality Assurance, Safety and Compliance reports directly to the Executive Vice President and is responsible for the development and implementation of the overall QA program during design, construction, preoperational testing, operation, and modification of the nuclear power plant. The Director of Quality Assurance, Safety and Compliance has delegated the quality assurance functions to the QA Manager who has three quality divisions--Quality Control, Quality Systems, and Quality Assurance--for ensuring full implementation of the QA program.

These three divisions have the authority to: (1) identify quality problems; (2) initiate, recommend, or provide solutions; (3) verify implementation of solutions; and (4) stop unsatisfactory work or further processing of unsatisfactory material. The QA organization is responsible for: (1) reviewing and approving quality-related documents (e.g., instructions, procedures, and specifications); (2) performing vendor QA prequalifications; (3) ensuring that procurement documents contain quality requirements that can be inspected and controlled; (4) surveillance and auditing of vendors; (5) documenting and reporting to management nonconformances discovered during surveillance or audit; (6) ensuring that corrective actions are effective and accomplished in a timely manner; and (7) auditing maintenance and operation activities.

The staff concludes that the QA organization has the necessary independence and authority to establish and implement the QA program. It is, therefore, acceptable.

17.3 Quality Assurance Program

The QA program description for the operation of Shoreham is described in the applicant's QA Manual and is supplemented by QA procedures and instructions that provide the detailed instructions and checklists necessary to implement or verify implementation of the quality assurance program requirements.

The applicant's QA program is structured to be in accordance with Appendix B to 10 CFR 50 and with the regulatory guidance shown in Table 17.1. These documents, coupled with the QA program description in the FSAR, form the foundation from which the overall QA program is formulated and describe how the requirements of Appendix B to 10 CFR 50 are satisfied. The program is implemented via the applicant's QA Manual and implementing procedures. These documents control quality-related activities involving safety-related items to satisfy the requirements of Appendix B to 10 CFR 50.

The QA program requires that QA documents encompass detailed controls for: (1) translating codes, standards, and regulatory requirements into specifications, procedures and instructions; (2) developing, reviewing, and approving procurement documents, including changes; (3) prescribing all quality-affecting activities by documented instructions, procedures, or drawings; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing material, equipment, processes or services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping of items; (11) identifying the inspection, test, and operating status of safety-related items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

The Quality Assurance Manager is responsible for the establishment and continuous implementation of the QA indoctrination and training program to ensure that persons involved in safety-related activities are knowledgeable in QA instructions and implementing procedures and demonstrate a high level of competence and skill in the performance of their quality-related activities.

Quality is verified through surveillance, inspection, testing, checking, and audit of work activities. The QA program requires that quality verification and inspections be performed by individuals from the plant staff who are not directly responsible for performing the actual work activity. Inspections are performed with procedures, instructions, and/or checklists by inspectors who have been qualified and certified in accordance with codes, standards, or the applicant's training programs. A technical QA review is performed on spare and replacement parts to ensure that controls for inspection and testing safety-related items are equal to or better than the original equipment.

The applicant's QA organization is responsible for QA audits, which includes planning, preparation, scheduling, performance, reporting, and verifying implementation of corrective and preventive action measures. These audits are performed with written procedures or checklists by qualified personnel not having direct responsibility in the areas being audited. The QA program establishes a comprehensive audit system to ensure that the QA program requirements and related supporting procedures are effective and properly implemented during the plant's operational phase. Audits will include an evaluation of QA practices, procedures, and instructions; work areas, activities, processes, and items; the effectiveness of implementation of the QA program; and conformance with policy directives.

The QA program requires documentation of audit results and written review by management having responsibility in the area audited to determine and take corrective action as required. Re-audits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences. Audit findings, which indicate quality trends and the effectiveness of the QA program, are reviewed by the Quality Assurance Manager and the Plant Manager and reported to appropriate management levels on a periodic basis.

17.4 Conclusions

The staff review of the applicant's QA program description for the operations phase has verified that the criteria of Appendix B to 10 CFR 50 have been addressed in the Shoreham QA program.

On the basis of its review and evaluation of the QA program description in FSAR Section 17.2 the staff concludes:

(1) The applicant's QA organization provides independence from cost and schedule (when opposed to safety considerations), authority to effectively carry out the operations QA program, and access to management at the level necessary for QA personnel to perform their quality assurance functions.

(2) The QA program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR 50 and with the Standard Review Plan Section 17.2.

Accordingly, the staff concludes that the applicant's description of the quality assurance program is in compliance with applicable NRC regulations and is, therefore, acceptable.

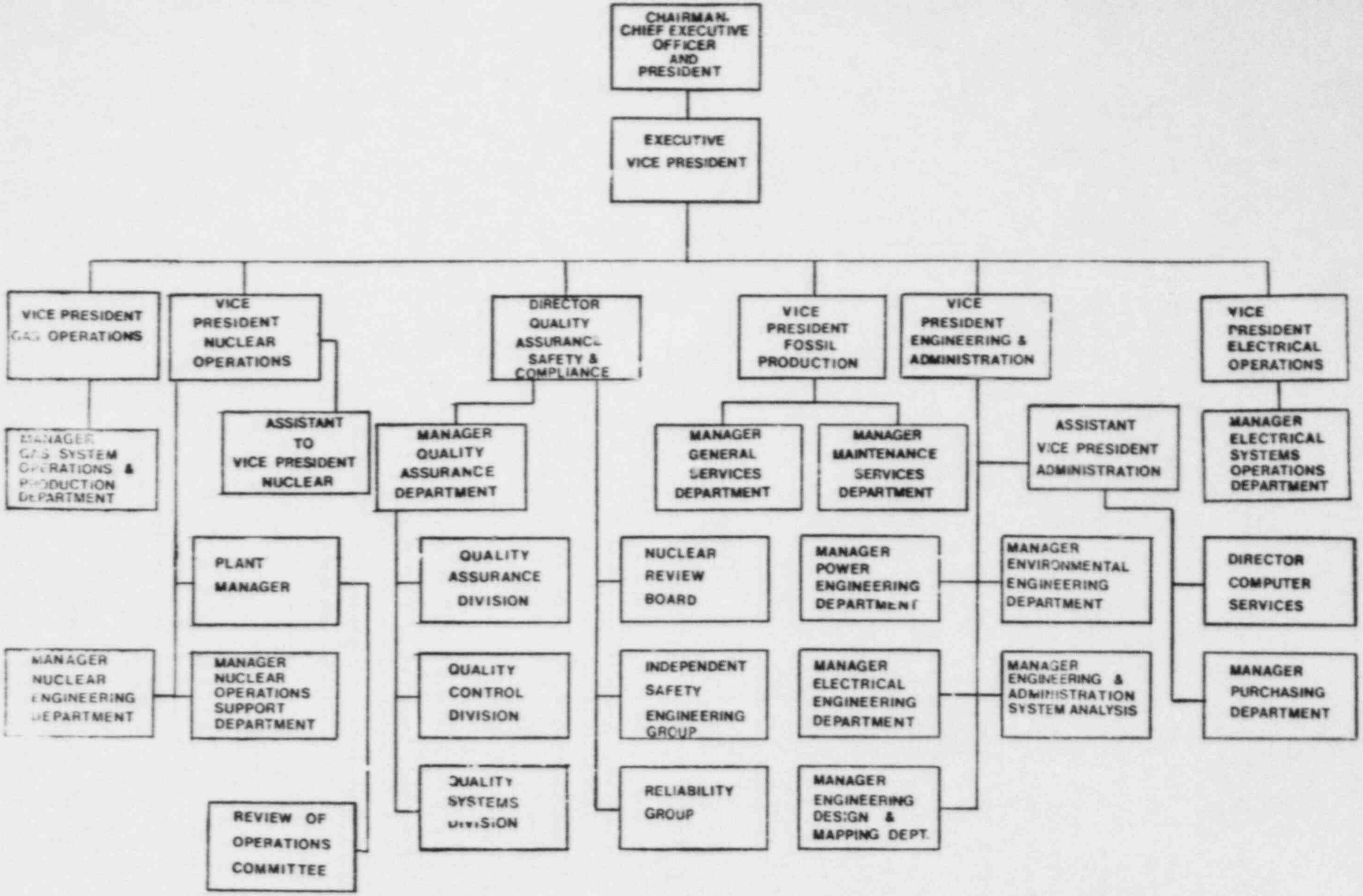

Figure 17.1 LILCO organization for quality assurance, Shoreham Unit 1

Table 17.1 Regulatory guidance applicable to quality assurance

RG 1.28 (Revision 0 - June 1972), "Quality Assurance Program Requirements (Design and Construction)"

RG 1.30 (Revision 0 - August 1972), "Quality Assurance Requirements for Installation, Inspection, and Testing of Instrumentation and Electric Equipment"

RG 1.33 (Revision 2 - February 1978), "Quality Assurance Program Requirements (Operation)"

RG 1.37 (Revision 0 - March 1973), "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"

RG 1.38 (Revision 0 - March 1973), "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

RG 1.39 (Revision 0 - March 1973), "Housekeeping Requirements for Water-Cooled Nuclear Power Plants"

RG 1.58 (Revision 0 - August 1973), "Quality Assurance Requirements for the Design of Nuclear Power Plants"

RG 1.64 (Revision 1 - February 1975), "Quality Assurance Requirements for the Design of Nuclear Power Plants"

RG 1.74 (Revision 0 - February 1974), "Quality Assurance Terms and Definitions"

RG 1.88 (Revision 0 - August 1974), "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"

RG 1.94 (Revision 0 - April 1975), "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"

RG 1.144 (Revision 0 - January 1979), "Auditing of Quality Assurance Programs for Nuclear Power Plants"

ANSI N45.2.8 - 1975, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants"

ANSI N45.2.13 - 1976, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

APPENDIX A

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS

SHOREHAM NUCLEAR POWER STATION, UNIT 1, PHASE I

EGG-HS-6374
REVISION 1
JUNE 1984

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS

SHOREHAM NUCLEAR POWER STATION (SNPS), UNIT 1

(PHASE 1)

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Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6457

EG&G Idaho

EGG-HS-6374
Revision 1

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
SHOREHAM NUCLEAR POWER STATION (SNPS), UNIT 1
(PHASE 1)
Docket No. 50-322

Author
T. C. Yen

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Revision 1
June 1984

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Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76ID01570

FIN No. A6457

ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of consistency with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This report contains EG&G's evaluation and recommendations for Long Island Lighting Company, Shoreham Nuclear Power Station.

EXECUTIVE SUMMARY

Long Island Lighting Company's response is consistent with the intent of Article 5.1.1, NUREG-0612. The applicant has supplied sufficient information for evaluation and the applicant's commitments to following the guidelines are acceptable.

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TABLES

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
SHOREHAM NUCLEAR POWER STATION, UNIT 1
(PHASE/I)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load-handling policy and procedures at Shoreham Nuclear Power Station, Unit 1. This evaluation was performed with the objective of assessing conformance to the general load-handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1.

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- o Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of the handling system

- o Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment

- o Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [3] to Long Island Lighting Company (LILCO), the applicant for Shoreham Nuclear Power Station, Unit 1 requesting that the applicant review provisions for handling and control of heavy loads at Shoreham Nuclear Power Station, Unit 1, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On July 17, 1981, LILCO provided the initial response [4] to this request. Subsequent submittal was sent on October 28, 1982.[5] LILCO confirmed information provided in conference calls by letters on November 11, 1983 [10] and February 21, 1984 [11].

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize Long Island Lighting Company's review of heavy load handling at Shoreham Nuclear Power Station, Unit 1, (SNPS) accompanied by EG&G's evaluation, conclusions, and recommendations to the applicant for making the facility more consistent with the intent of NUREG-0612. The applicant has indicated the weight of a heavy load for this facility (as defined in NUREG-0612, Article 1.2) as 1000 pounds.

2.2 Heavy Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a review of the justification for excluding overhead handling systems from the above-mentioned list.

2.2.1 Scope

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

A. Summary of Applicant's Statements

The applicant's review of overhead handling systems identified the cranes and hoists shown in Table 2.1 as those which handle heavy loads in the vicinity of irradiated fuel or safe shutdown equipment.

The applicant has also identified numerous other cranes that have been excluded from satisfying the criteria of the general guidelines of NUREG-0612.

B. <u>EG&G Evaluation</u>

Based on the drawings provided, it appears that LILCO has identified all the overhead handling systems at Shoreham Nuclear Power Plant, Unit 1, which are required to meet NUREG 0612.

C. <u>EG&G Conclusions and Recommendations</u>

Based on the information provided, EG&G concludes that the applicant has included all applicable hoists and cranes in their list of handling systems which must be consistent with the general guidelines of NUREG-0612.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of NUREG-0612, Article 5.1.1. EG&G's conclusions and recommendations are provided in summaries for each guideline.

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- o Guideline 1--Safe Load Paths
- o Guideline 2--Load-Handling Procedures
- o Guideline 3--Crane Operator Training

TABLE 2.1. NONEXEMPT HEAVY LOAD HANDLING SYSTEMS--SHOREHAM UNIT 1

Handling System	Capacity (tons)	Location
(1) Polar crane	125/3C	Reactor building
(2) RB receiving jib crane	1.5	RB Receiving area
(3) Recirculation pump mctor, Monorail	24	Reactor building
(4) Recirculation pump motor, Transfer monorail	24	Reactor building
(5) CRD pump, monorail	2	Reactor building
(6) SLC pump, bridge/monorail	2	Reactor building
(7) RBCLCW repair, bridge/monorail	1	Reactor building
(8) Personnel hatch, bridge/monorail	2	Inside containment
(9) HPCI, bridge/monorail	4	Reactor building
(10) RCIC, bridge/monorail	2	Reactor building
(11) MSIVs, monorail	2	Main steam tunnel
(12) MS SRVs monorail	2	Reactor building
(13) MS SRVs, pad eye	4	Reactor building
(14) MSIVs, monorail	2	Inside containment
(15) CDR FLT, monorail	1	Reactor building
(16) Diesel generator, monorail	2	Control building
(17) Diesel generator, monorail	15	Control b[illegible]ing

- o Guideline 4--Special Lifting Devices
- o Guideline 5--Lifting Devices (Not Specially Designed)
- o Guideline 6--Cranes (Inspection, Testing, and Maintenance)
- o Guideline 7--Crane Design.

These seven guidelines should be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool, or in other areas where a load drop may damage safe shutdown systems. The succeeding paragraphs address the guidelines individually.

2.3.1 Safe Load Paths [Guideline 1, NUREG-0612, Article 5.1.1(1)]

> "Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent-fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

A. Summary of Applicant's Statements

The applicant's initial response [4] indicates that all the safe shutdown and decay heat removal equipment is located in the reactor building, control building, and screenwell.

In the reactor building, the majority of the equipment is located below elevation 175'-0. The spent-fuel pool and the area of most heavy-load operations are located at elevation 175'-0. Heavy-load operations at this level are handled by the polar crane. The weights of heavy loads carried by the polar crane are identified in the applicant's response and load-handling operations of the spent-fuel shipping cask with the polar crane are discussed.

The applicant further states in the submittal [5] that safe load paths will be defined for every load-handling operation with a potential for an accident near RPV, spent-fuel pool, or systems required for safe shutdown. Alternatives to the safe load path will be approved by the Station Maintenance Engineer or his designee.

B. EG&G Evaluation

The drawings submitted by the applicant are sufficient to identify the locations of all the nonexempt cranes. Safe load paths, not defined at present, will be defined for every heavy-load-handling operation in the future. Load paths will be defined by the temporary use of pylons or tape.

Safe load paths for Heavy lifts have been developed and are appended to applicable procedures.[10]

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G considers that SNPS is consistent with the intent of Guideline 1.

2.3.2 Load-Handling Procedures [Guideline 2, NUREG-0612, Article 5.1.1(2)]

"Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

A. Summary of Applicant's Statements

Load-handling procedures have been developed for each heavy-load-handling system. The procedures include: inspection before movement, load-handling sequences, specific load-handling operations, and special precautions [5]. Heavy loads to be handled by the polar crane are listed in applicant's Table 3 [4].

"Twenty eight heavy load handling procedures have been written to address every heavy load handling operation with a potential for an accident near the RPV, spent fuel or systems required for safe shutdown. These procedures have been approved by the Shoreham Review of Operations Committee (ROC). Safe load paths for these heavy load lifts have been developed and are appended to the applicable procedures."[10]

B. EG&G Evaluation

LILCO has provided the necessary procedures as specified in NUREG 0612.

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that SNPS-1 is consistent with the intent of Guideline 2.

2.3.3 Crane Operator Training [Guideline 3, NUREG-0612, Article 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [6]."

A. Summary of Applicant's Statements

"The Shoreham crane operators have been trained and qualified in accordance with Chapter 2-3 of ANSI B30.2-1976. The prerequisite section of station procedure SP 35.001.01, "Handling of Heavy Loads with the Reactor Building Polar Crane (1T31-CRN-002)," requires that the crane operator be trained and qualified in accordance with ANSI B30.2-1976. The station training section is responsible for maintaining the crane operator certification records."[10]

B. EG&G Evaluation

LILCO has stated that procedures for crane operator training have been implemented and that the training is according to Chapter 2-3 of ANSI B30.2-1976.

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that SNPS-1 is consistent with the intent of Guideline 3.

2.3.4 Special Lifting Devices [Guideline 4, NUREG-0612, Article 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [7]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in

Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) or the load and of the intervening components of the special handling device."

A. Summary of Applicant's Statements

"In accordance with the commitment stated in Reference 1 (TER Reference [10]), enclosed herewith for your review are forty (40) copies of the Shoreham Nuclear Power Station, "Report on Special Lifting Devices." (Stone & Webster Report, 11600.02.) This report covers the design, fabrication and initial acceptance testing of the six (6) special lifting devices listed in Table 3 of Reference 2. (TER Reference [4].) Please note that the mark numbers and capacities of these lifting devices have been updated from the original Phase I submittal.

Special lifting devices for spent fuel shipping casks have not been addressed, since it is not known at this time which cask type will be used. However, any special lifting device used with a spent fuel cask will comply with the requirements of NUREG 0612.

As indicated in the Schedule for Special Lifting Device Modifications and Acceptance Testing, four (4) lifting devices require modification and five (5) require load testing. Preparations are currently underway to implement the report recommendations. Modification and acceptance testing will be completed prior to the beginning of the first scheduled refueling outage for all special lifting devices."

B. EG&G Evaluation

The Stone & Webster Report No. 11600.02, transmitted with Reference [11], completely analyzes the SNPS-1 special lifting devices and compares their design, inspection, and testing with the provisions of ANSI N 14.6-1978. This report also discusses the maximum hoist speeds attainable and their effect on the dynamic loading of the devices.

EG&G concurs with the conclusions of Report No. 11600 02, i.e., four (4) lifting devices require modification and five (5) require load testing. The proposed completion dates are also acceptable.

LILCO has stated that any special lifting device used with a spent fuel cask will comply with the requirements of NUREG 0612.

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that SNPS-1 is consistent with the guideline requirements of Article 5.1.1(4), NUREG-0612.

2.3.5 Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612, Article 5.1.1(5)]

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [8]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

A. Summary of Applicant's Statements

"Slings have been rated using the sum of the static and maximum dynamic loads in accordance with the guidelines of ANSI B30.9-1971. Each sling is coded to reflect the load rating. Shoreham does not have any slings that are restricted to a particular crane. The aforementioned load handling procedures specify the rating capacity of the sling(s) to be used for a particular load."[10]

B. EG&G Evaluation

The applicant's statement indicates consistency with Guideline 5.

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that SNP-1 is consistent with the intent of Guideline 5.

2.3.6 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Article 5.1.1(6)]

"The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

A. Summary of Applicant's Statements

"Crane inspection, testing, and maintenance procedures which meet the requirements of ANSI B30.2-1976, have been developed. The frequency of inspection will be determined by the service of the crane as delineated in ANSI B30.2-1976"[5].

B. EG&G Evaluation

The applicant has stated that crane inspection, testing, and maintenance procedures have been developed in accordance with the requirements of ANSI B30.2-1976.

C. EG&G Conclusions and Recommendations

Based on the statement in Reference [5], the applicant's action is consistent with the intent of Guideline 6, NUREG-0612, Article 5.1.1(6).

2.3.7 Crane Design [Guideline 7, NUREG-0612, Article 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes' [9]. An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

A. Summary of Applicant's Statements

The polar crane is the only nonexempt overhead load-handling system required to meet ANSI B30.2-1976. This crane does meet the standard as well as the guidelines of CMAA specification 70 [4].

B. EG&G Evaluation

As stated, the design of the polar crane at Shoreham Nuclear Power Station, Unit 1, satisfies the design specifications required by this guideline.

C. EG&G Conclusions

The applicant's response is consistent with the intent of Article 5.1.1(7), NUREG-0612.

2.4 Interim Protection Measures

The NRC staff has established (NUREG-0612, Article 5.3) that six measures should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612, Article 5.1, is complete. Since Shoreham Nuclear Station, Unit 1, is still under construction, these measures need not be addressed. However, if the guidelines of Article 5.1.1, NUREG-0612, are not satisfied before heavy load handling, the interim protection measures must be implemented.

3. CONCLUDING SUMMARY

3.1 Applicable Load-Handling Systems

The list of cranes and hoists supplied by the applicant as being subject to the provisions of NUREG-0612 is complete.

3.2 Guideline Recommendations

Consistency with the seven NRC guidelines for heavy load handling (Section 2.3) is satisfied at Shoreham Nuclear Power Station, Unit 1. This conclusion is represented in tabular form as Table 3.1.

Guideline	Recommendation
1. Section 2.3.1	Actions taken by the applicant are consistent with the intent of Guideline 1, NUREG 0612, Article 5.1.1 (1).
2. Section 2.3.2	Actions taken by the applicant are consistent with the intent of Guideline 2, NUREG 0612, Article 5.1.1 (2).
3. Section 2.3.3	Actions taken by the applicant are consistent with the intent of Guideline 3, NUREG 0612, Article 5.1.1 (3).
4. Section 2.3.4	Actions taken by the applicant are consistent with the intent of Guideline 4, NUREG 0612, Article 5.1.1 (4).
5. Section 2.3.5	Actions taken by the applicant are consistent with the intent of Guideline 5, NUREG 0612, Article 5.1.1 (5).
6. Section 2.5.6	Actions taken by the applicant are consistent with the intent of Guideline 6, NUREG-0612, Article 5.1.1(6).
7. Section 2.5.7	Actions taken by the applicant are consistent with the intent of Guideline 7, NUREG-0612, Article 5.1.1(7).

TABLE 3.1. GUIDELINE COMPLIANCE STATUS OF LONG ISLAND LIGHTING COMPANY, SHOREHAM NUCLEAR POWER STATION, UNIT 1, HEAVY LOAD CONTROL

Equipment Designation	Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline 7 Crane Design
IT31-CRN-002 (1) Polarcrane	34 item listed weights ranging from 1.5 to 103 tons	124/30	C	C	C	C	C	C	C
IT31-CRN-019 (2) RB Receiving Area Crane	No listing	1.5	C	C	C	C	C	C	--
IT31-CRN-037 (3) Recirc Pump Motor	Recirc. pump motor No weight given	24	C	C	C	C	C	C	--
IT31-CRN-045 (4) Recirc Pump Motor Transfer	Recirc. pump motor No weight given	24	C	C	C	C	C	C	--
[illegible]1-CRN-078 (5) CRD Pump	CRD pump No weight given	2	C	C	C	C	C	C	--
IT31-CRN-082 (6) SLC Pump	SLC pump No weight given	2	C	C	C	C	C	C	--
IT31-CRN-083 (7) RBCLCW Repair	RBCLCW No weight given	1	C	C	C	C	C	C	--
IT31-CRN-084 (8) Personnel Hatch	Personnel Hatch No weight given	2	C	C	C	C	C	C	--
IT31-CRN-085 (9) HPCI	HPCI No weight given	4	C	C	C	C	C	C	--
IT31-CRN-086 (10) RCIC	RCIC No weight given	2	C	C	C	C	C	C	--
IT31-CRN-090 (11) MSIVs	MSIV No weight given	2	C	C	C	C	C	C	--
IT31-CRN-091 (12) MS SRVs	MS SRV No weight given	2	C	C	C	C	C	C	--
IT31-CRN-092A&B (13) MS SRVs	MS SRV No weight given	4	C	C	C	C	C	C	--

TABLE 3.1. (continued)

Equipment Designation	Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline 7 Crane Design
1T31-CRN-094A&D (14) MSIV's	MSIV No weight given	2	C	C	C	C	C	C	--
1T21-CRN-099 (15) CRD Flt	CRD No weight given	1	C	C	C	C	C	C	--
1T31-CRN-104A-F (16) Diesel Generator	Diesel Generator No weight given	2	C	C	C	C	C	C	--
1T31-CRN-105 (17) Diesel Generator	Diesel Generator No weight given	15	C	C	C	C	C	C	--

C = Applicant's action consistent with NUREG-0612 Guideline.
NC = Applicant's action inconsistent with NUREG-0612 Guideline.
I = Insufficient information provided by the applicant.
-- = Not applicable

4. REFERENCES

1. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, NRC.

2. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 17 May 1978.

3. USNRC, Letter to LILCO. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 22 December 1980.

4. B. R. McCaffery (LILCO), Letter to H. R. Denton (NRC). Subject: Docket No. 500322 Shoreham Nuclear Power Station Unit 1, LILCO, July 17, 1981.

5. J. L. Smith (LILCO), Letter to H. R. Denton (NRC). Subject: Docket No. 500322 Shoreham Nuclear Power Station Unit 1, LILCO, October 28, 1982.

6. ANSI B30.2-1976, "Overhead and Gantry Cranes."

7. ANSI N14 6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials."

8. ANSI B30.9-1971, "Slings."

9. CMAA-70, "Specifications for Electric Overhead Traveling Cranes."

10. J. L. Smith (LILCO), Letter to H. R. Denton (NRC), SNRC-980, Subject: Shoreham Nuclear Power Station, Unit 1, Docket No. 50-322, November 11, 1983.

11. B. R. Mc Caffrey (LILCO), Letter to H. R. Denton (NRC), SNRC 1007, Subject: Shoreham Nuclear Power Station, Unit 1, Docket No. 50-322, February 21, 1984.

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APPENDIX B

NUCLEAR REGULATORY COMMISSION (NRC)
UNRESOLVED SAFETY ISSUES

APPENDIX B
NUCLEAR REGULATORY COMMISSION (NRC)
UNRESOLVED SAFETY ISSUES

A-44 Station Blackout

The SER stated that the applicant was required to establish emergency procedures and operator training for safe operation of the facility and restoration of alternating current power following a station blackout event. The staff has completed its review of the applicant's emergency procedures and operator training. On the basis of this review, the staff concludes that the concerns of this unresolved item have been adequately addressed; the results of this review were issued in NRC Region I Inspection Report 50-322/84-29, dated August 30, 1984. This report states:

> Station Blackout refers to the complete loss of all AC electrical power to the plant. This is considered very unlikely and beyond the plant's design basis, due to the number and diversity of AC power sources available. Nevertheless, due to the significant consequences of a station blackout, the licensee was required to establish procedures and training for this event. During a previous inspection, a confirmatory review of these procedures and training was performed. The inspector identified specific shortcomings which were subsequently addressed by the licensee. During the present inspection, the inspector verified that the following corrective actions had been taken:
>
> - A lesson plan to cover training for a Station Blackout was issued as part of the Requalification Plan;
>
> - Station Procedure SP29-015.01, "Loss of Offsite Power," was revised to include Scram, Turbine Trip, and NSSS Isolation in the Automatic Actions section of the procedure; and,
>
> - SP29.015.02, "Loss of All AC Power," was revised extensively to correct editorial errors and incorporate the recommended procedure improvements and clarifications.
>
> Based on this review, the inspector determined that the concerns addressed in this unresolved item had been adequately addressed.

Thus, the staff concludes that there is reasonable assurance that Shoreham Nuclear Power Station can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public, and this item is resolved.

A-46 Seismic Qualification of Equipment in Operating Plants

The scope of Task A-46 is limited to dealing with seismic qualification of equipment in operating plants. Shoreham is not an operating plant. Moreover, Shoreham is designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification are in accordance with the latest codes and standards. Therefore, the issue related to Task A-46 is not applicable for Shoreham.

A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern was the potential for a single failure--such as a loss of a power supply, short circuit, open circuit, or sensor failure--to cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern was that a postulated accident could cause control system failures that would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems could not safely handle, rigorous in-depth studies were not performed to verify this belief. The likelihood of an accident that would affect a particular control system--and the effects of the control system failures--may differ from plant to plant. Therefore, it is not possible to develop generic answers, but it is possible to develop generic criteria that can be used for future plant-specific reviews. The purpose of this Unresolved Safety Issue is to verify the adequacy of existing criteria for control systems and, if necessary, to develop additional generic criteria that can be used for plant-specific reviews.

The Shoreham safety systems were designed to ensure that control system failures (either single or multiple) would not prevent automatic or manual initiation and operation of any safety system equipment required (1) to trip the plant or (2) to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or accident. This has been accomplished by either providing independence between safety- and nonsafety-grade systems or providing isolation devices between safety- and nonsafety-grade systems. These devices preclude the propagation of nonsafety-grade system equipment faults so as to not impair the operation of the safety-grade system equipment.

A wide range of bounding transients and accidents have been analyzed to ensure that the postulated events could be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that the control system failures (single or multiple) will not defeat safety system action.

Also, in NRC Information Notice 79-22, "Qualification of Control System" (dated September 17, 1979), the applicant was requested to: (1) review the possibility of consequential control system failures that exacerbate the effects of high-energy line breaks (HELBs) and (2) adopt new operator procedures, where needed, to ensure that the postulated events would be adequately mitigated. As part of its review, the staff is also evaluating the qualification program to ensure that equipment that may potentially be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile enironment. The staff's evaluation of the applicant's response to Information Notice 79-22 and the adequacy of the qualification program were addressed in Section 7.7.1 of SSER 4 and Section 3.11 of SSER 7.

During the staff review, the importance of the availability of post-accident instrumentation was emphasized (Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident"). The staff evaluated the control system design to ensure that control system failures would not deprive the operator of information required to maintain the plant in a safe shutdown condition after any anticipated operational occurrence or accident. The applicant was requested to evaluate the control systems and identify any control systems whose malfunction could impact plant safety. The applicant was requested to document the degree of interdependence of these identified control systems and identify the use, if any, of common power supplies and the use of common sensors or common sensor impulse lines whose failure could have potential safety significance. The status of the review and the staff's evaluation were in Section 7.7.2 of SSER 4. This item is resolved.

In addition, in IE Bulletin 79-27 ("Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979) the applicant was asked to perform evaluations to ensure the adequacy of plant procedures for accomplishing shutdown on loss of power to any electrical bus supplying power for instruments and controls. The results of this review were in SSER 4, Section 7.5. This item is resolved.

The subtask of this issue concerning the reactor overfill transient in boiling water reactors is currently under review by the BWR Owners Group, of which the applicant is a member. Pending ultimate resolution of this item, the applicant has incorporated in the Shoreham design a commercial-grade high-level trip (Level 8) of the feedwater systems to prevent the occurence of overfill transients.

On the basis of these above considerations and on the satisfactory resolution of these items, the staff concludes that there is reasonable assurance that Shoreham can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION

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13. ABSTRACT (200 words or less)

Supplement 8 (SSET 8) to the Safety Evaluation Report on Long Island Lighting Company's application for a license to operate the Shoreham Nuclear Power Station, Unit 1, located in Suffolk County, New York, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. This supplement addresses several items that have been reviewed by the staff since the previous supplement was issued.

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