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SHOREHAM
DEFUELED SAFETY ANALYSIS REPORT (DSAR)

Insertion Instructions
for Incorporating Revision 4

(NOTE: The electronic conversion process of the DSAR from the IBM 8100 System to the DEC System has resulted in different amounts of text on certain pages. These pages, included for completeness, are marked Revision 4, but do not have revision bars since no changes to the text have occurred.)

Replace the following pages of the DSAR with the attached pages. The revised pages are identified by revision number and, as appropriate, contain vertical line(s) in the right margin indicating the area of change.

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Defueled Safety Analysis Report (DSAR) is an appendix to the Shoreham USAR and is submitted by Long Island Power Authority, hereafter known as LIPA, in support of the permanently defueled configuration of the Shoreham Nuclear Power Station as authorized by Facility Operating License NPF-82, i.e. the SNPS Possession Only License or POL, as transferred from the Long Island Lighting Company (LILCO).

The description of the plant remains essentially unchanged from the description in Section 1.1 of the SNPS USAR. However, many of the sections which described systems needed to support power operation are significantly changed or excluded from the DSAR. The DSAR format is the same as that used for the USAR (i.e. NRC Regulatory Guide 1.70, Rev. 1, 1972); however, commensurate with the level of activity of a defueled plant, the content is reduced.

The purpose of the DSAR is to provide a safety analysis for the storage and handling of Shoreham low burnup first cycle spent fuel. The DSAR confirms that fuel storage and handling systems, structures, components and programs ensure that there is no undue risk to public health and safety during normal and postulated accident conditions.

The DSAR assumes that the 560 fuel bundles comprising the Shoreham core are stored under water in the Shoreham spent fuel pool. The fuel bundles are held in Seismic Category I spent fuel racks within the stainless steel-lined spent fuel pool. The spent fuel pool is located in the secondary containment of the Shoreham reactor building. The structures are designed to withstand seismic loads.

The Shoreham spent fuel is in a low burnup condition. The Shoreham Nuclear Power Station operated during low power testing at power levels not exceeding 5% of rated power. The effective burnup of the fuel is approximately 2 full power days. This results in an estimated total core wide heat generation rate of approximately 550 watts as of June 1939. The estimated fuel heat load will reduce to approximately 250 watts by June 1991. Figure 15.1-1 depicts the fuel heat load versus time. Based on this low heat generation rate, systems for active cooling are not required, and only minimal capacity systems are required for pool water makeup to handle evaporation.

The Shoreham spent fuel contains limited quantities of radioactive materials that are available for release. As is stated in DSAR Section 12.2, approximately 176,000 curies of radioactivity reside in the 560 fuel assemblies. Gaseous activity in the fuel assemblies is primarily Krypton-85 (a noble gas with a 10.7 year half-life), and consists of approximately 1560 curies. The radioactive inventory estimation is based on a two year decay from the last burnup period (completed June 7, 1987). Other sources of radioactivity outside the core are minor, and include small amounts of contamination in the bottom of sumps, the suppression pool, inside the reactor pressure vessel, and in the radwaste systems.

Chapter 15 presents radiological analyses for those accidents identified in the USAR which are applicable to the defueled plant. In addition, no other accident mechanisms were identified for the plant's defueled condition which are not bounded by Chapter 15. The events analyzed in Chapter 15 are:

1. Fuel Handling Accident (Fuel Bundle Drop)
2. Radwaste Tank Rupture

The only design basis accident involving reactor fuel is a Fuel Handling Accident, in which no heat generation takes place. As such, the activity available for release in this design basis accident is primarily Krypton-85, and consists of approximately 2.5 curies. In addition, a worst case radiological event is postulated in which the entire gaseous activity of the core is released to the reactor building. This event was postulated to conservatively bound any possible situation involving large-scale mechanical damage of the fuel.

The results of the September 1989 spent fuel radiological analysis described in DSAR Chapter 15 indicate that integrated doses are very small in comparison with 10CFR100 limits. For the worst case scenario in which all the gaseous activity is assumed to be released from the entire core, a spectrum of cases were analyzed as follows: operation of the standby ventilation system, operation of the normal ventilation system, and no ventilation (modeled as puff release). The results of the analyses indicate that the integrated whole body and skin doses, with Reactor Building Normal Ventilation System operational, are less than approximately .03% of 10CFR100 limits. The results of the radiological analysis for the worst case fuel damage scenario are depicted graphically in Figure 15.1.36A-1. In particular, it was demonstrated that the reactor building standby ventilation system operation does not provide an important filtering or ventilation safety function and is therefore no longer required after fuel is stored in the pool.

Based on this analysis, it has been found that the spent fuel pool provides a high degree of passive safety protection for Shoreham spent fuel. Active safety systems are not required to mitigate postulated accidents; however, support systems are required to meet the intent of 10CFR50 Appendix A, General Design Criteria (see Chapter 3 for a listing) and Regulatory Guide 1.13. Supporting systems are required to provide for radiation monitoring, fuel pool makeup, fuel pool cleanup, radwaste management, and normal building services. Therefore a reclassification of safety systems is proposed based on the importance to safety associated with each plant system with the plant defueled.

The DSAR assumes that the Shoreham spent fuel from the initial core is to be stored for some interim period in the spent fuel pool contained within the SNPS reactor building.

The assumed configuration of principal plant systems is as follows:

1. All 560 fuel bundles have been removed from the reactor and are being stored in seismic Category I spent fuel racks in the spent fuel storage pool. The total decay heat power of the entire core has been determined to be approximately 550 watts as of June 1989 (reference DSAR Chapter 15).

2. As described in DSAR Chapter 9, the spent fuel storage pool water level is maintained at its normal water level. Makeup will be furnished from the condensate transfer system or the demineralized and makeup water system. The fuel pool cooling system is not in service due to the low heat load in the pool. Water quality is maintained by the fuel pool cleanup system. The spent fuel pool transfer canal gates will remain installed. Fuel pool level and temperature are alarmed in the Control Room.

3. The capability for fuel handling will be maintained as described in DSAR Chapter 9.

4. The Nuclear Boiler, Reactor Protection, Emergency Core Cooling, and Primary Containment systems are not required. This is discussed in DSAR Chapters 4, 5 and 6.

5. Two independent offsite AC power sources will be maintained to supply reliable electric power. In addition, as discussed in Chapter 8, blackstart combustion turbines exist nearby in the Shoreham west site to supply emergency power to the plant. However, as discussed in DSAR Chapter 15, onsite Emergency Diesel Electric Power is not required to mitigate design basis accidents. AC Power is required by Technical Specifications to remain operable during fuel movement (including one non-safety emergency diesel generator).

6. The normal ventilation system (RBNVS) provides a controlled and monitored release capability but secondary containment integrity is no longer required as discussed in the DSAR Chapter 15 Safety Analysis.

7. The steam and power conversion systems are not required to be operable or functional.

8. Process and area radiation monitoring are maintained consistent with fuel storage and handling requirements, and are described in DSAR Chapters 11 and 12.

9. Radwaste Systems described in DSAR Chapter 11 are maintained to provide an appropriate level of radioactive liquid and solid waste management primarily due to operation of the spent fuel pool.

10. Major systems that remain functional to provide non-safety related supporting services include:

    a) Service Water (DSAR Chapter 9 and 10)
    b) Chilled Water Systems (DSAR Chapter 9)
    c) Compressed Air (DSAR Chapter 10)
    d) HVAC Systems (DSAR Chapter 9)

The DSAR addresses the following major programs:

1. Proposed revised Technical Specifications (Appendices A and B) including the basis of the specification is provided. (DSAR Chapter 16)

2. Conduct of operations and the LIPA organizational structure is described in Chapter 13. The ISEG functions are no longer considered necessary for a defueled reactor.

3. The Quality Assurance Program is maintained as described in DSAR Chapter 17.

4. The Fire Protection Program is maintained as described in DSAR Section 9.5.1 and the FHAR. With respect to overall nuclear safety, the primary focus of the Fire Protection Program is shifted from the protection of plant safe shutdown capability to the safety of stored irradiated fuel.

5. An offsite Radiological Environmental Monitoring Program (REMP) is maintained as described in DSAR Section 11.6.

6. Changes to the LIPA Security Plan are being provided separately from the DSAR.

7. A LIPA Defueled Emergency Preparedness Plan is submitted separately to the NRC, adapted from the LILCO plan previously reviewed and approved by the NRC for the defueled configuration of SNPS.

8. A Shoreham Decommissioning Plan was submitted separately to the NRC and is incorporated by reference in this DSAR. See Section 1.2 for additional information.

1.2 GENERAL PLANT DESCRIPTION

This section of the USAR is historically descriptive but the specifics of general and design criteria and modes of operation are generally no longer applicable to the defueled plant. Design and operating information will be found in other sections of the DSAR e.g., Table 3.2-1.

Refer to the USAR for information on this subject. However, the systems which will remain operable for an extended time period in the defueled condition are listed in Table 1.2-1 of the DSAR. All other systems will be either functional or non-operable.

The following definitions apply:

1. Operable - System(s) maintained to meet Technical Specifications.

2. Functional - Essential support system(s) not required per Technical Specifications but necessary for minimal plant functions, habitability, and maintenance.

3. Nonoperable - Those systems not normally operated in the defueled mode. These systems will be in the deenergized state. All systems will be maintained consistent with the Decommissioning Rule (no action will be taken which will affect the methods or options available for decommissioning

or increase the cost of decommissioning prior to approval of a decommissioning plan). These systems may be operated as necessary to support decommissioning activities as required.

The Shoreham Decommissioning Plan submitted on December 29, 1990 as amended by letters SNRC-1832 (8/26/91), LSNRC-1855 (10/16/91), LSNRC-1859 (11/27/91) and LSNRC 1874 (12/6/91) contains a detailed description of the plan for the decommissioning (i.e. decontamination and dismantlement) of Shoreham's radioactive systems and structures. The Shoreham Decommissioning Plan as amended, is hereby incorporated by reference upon its approval by the NRC.

Where information on systems or structures appears in both the DSAR and the DP, the information in the DP must be considered governing. For example, the DP states that the following systems and structures are contaminated or activated and must be decommissioned:

<u>Systems</u>

- Control Rod Drive
- Process Sampling
- Core Spray
- Residual Heat Removal
- Reactor Water Cleanup
- Liquid Radwaste
- Fuel Pool Cooling and Cleanup
- Condensate Demineralizer
- Reactor Recirculation

<u>Structures</u>

- Primary Containment
- Equipment/Floor Drains and Sumps
- Dryer and Separator Storage Pool
- Reactor Head Cavity
- Spent Fuel Storage Racks
- Spent Fuel Storage Pool
- Radwaste Laydown Area
- Reactor Pressure Vessel and Internals

The DSAR, however, also contains descriptions of these systems and structures which do not address how they will be affected by decommissioning. The descriptions of the above systems and structures in the DSAR are, therefore, historical only and are superseded by the information in the DP.

1.3 COMPARISON TABLES

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject. However, the status of systems which will remain operable for an extended time period in the defueled condition is described in Table 1.2-1 of the DSAR. The systems described in this section are not required for the defueled condition.

1.6 MATERIAL INCORPORATED BY REFERENCE

The information contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

1.7 SYMBOLS USED IN ENGINEERING DRAWINGS

The information contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2.3 METEOROLOGY

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged except that the 33 ft. tower south of the plant will not be used. Additionally, the following information regarding the Operational Program applies to DSAR. Refer to USAR for other information on this subject.

2.3.3.2 Operational Program

The operational meteorological monitoring program uses instrumentation to determine wind-speed and -direction at 33- and 150-ft. ambient air temperature at 33-ft and temperature differential (Temp @ 150-ft minus Temp @ 33-ft). These instruments are located on SNPS' 400 ft. meteorological tower which is located approximately 5100-ft WSW of the reactor building (Figure 2.1.1.1). The MET tower was positioned sufficiently close to SNPS to provide representative observations of released gaseous effluents, but far enough away to minimize atmospheric disturbances caused by SNPS' structures.

Wind-speed and -direction at the 33-ft level, along with the temperature differential are transmitted to the Technical Support Center. In addition to these parameters, wind-speed and -direction at 150-ft., and temperature at 33-ft. are transmitted to the Main Control Room and entered into the RMS computer.

All instrumentation was either manufactured or supplied by Climatronics Corporation, Hauppauge, New York. The specifications outlined in Regulatory Guide 1.23 were used in the selection of these instruments. Wind instrumentation includes F460 wind sets (three cup anemometers and direction vanes) at the

33 and 150 ft. levels. Temperature sensors in shielded aspirators are oriented in a northerly direction to limit the influence of solar insolation. A motor and fan draw a constant flow of air at ambient conditions over the sensor to ensure accurate measurements.

Observations from 33 ft. are used to model the dispersion of ground level release of activity, while data from 150 ft. are used for elevated releases. The data obtained are used to project the dispersion of plant gaseous effluents based on Gaussian model and are included in required periodic reports.

To ensure the operability of the system, semi-annual calibrations are performed by a qualified vendor, and channel checks are performed by the operators on shift using qualitative assessment of the channel's behavior during operation. Operators do this by checking the chart recorders in the control room. This instrumentation includes:

1) Wind speed monitors at the 33-ft. and 150-ft. elevations;
2) Wind direction monitors at the 33-ft. and 150-ft. elevations;
3) Ambient temperature monitor at the 33-ft elevation; and
4) Differential air temperature monitor which uses the temperature data recorded at 33-ft. and 150-ft. elevations.

Meteorological sensors are replaced on a semi-annual basis with replacement sensors which have been calibrated in the laboratory of a qualified vendor. Vendor personnel perform the sensor substitutions under the direction of LIPA personnel. LIPA/LILCO technicians perform normal maintenance and inspection on instrumentation at the tower. Calibration and maintenance procedures have been developed for field testing and maintenance of each meteorological channel at the Shoreham site.

Spare sensors and auxiliary equipment are available for replacement of any malfunctioning components of the system. In the event that a Technical Specification meteorological tower instrument is damaged, causing one or more monitoring instrumentation channels to be inoperable for more than seven (7) days, refer to the Technical Specifications for the required action.

2.4 HYDROLOGIC ENGINEERING

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged with the exception of Subsections 2.4.8.1 and 2.4.11.5:

2.4.8.1 Canals

The USAR requires that the Intake Canal bottom be monitored on a yearly basis, and dredging carried out when the results of the annual monitoring indicate cumulative sediment deposition has exceeded one (1) foot. This one (1) foot maximum sediment depth requirement is based upon anticipated sediment deposition of 3.2feet during a low water Probably Maximum Hurricane (PMH) event. For the defueled condition, design for the PMH is not required since the decay heat load of the fuel is negligible. Annual monitoring and dredging will not be required during the time that the plant is expected to be in the defueled condition. This is based on the May 1990 Intake Canal soundings and the current rate of sediment deposition. However, the intake canal will continue to be used as a source of cooling water for normal plant needs (refer to DSAR Section 9.2.1).

2.4.11.5 Plant Requirements

The USAR states that the required minimum safety related cooling water flow is 12,800 gpm supplied by two service water pumps. This minimum safety related flow is no longer required for the defueled condition since the RBSW system is considered non-safety related because it does not provide cooling water to any plant equipment required to perform a safety function. One RBSW pump will be used to supply cooling water for normal plant needs (see DSAR Section 9.2.1).

2.5 GEOLOGY AND SEISMOLOGY

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2A BORING LOGS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2B SEISMICITY INVESTIGATIONS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2C A REEVALUATION OF THE INTENSITY OF THE EAST HADDAM, CONNECTICUT EARTHQUAKE OF MAY 16, 1971

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2D REEVALUATION OF THE REPORTED EARTHQUAKE AT PORT JEFFERSON, LONG ISLAND, NEW YORK

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2E REEVALUATION OF THE EARTHQUAKE OF OCTOBER 26, 1845

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2F REEVALUATION OF THE EARTHQUAKE OF JANUARY 17, 1855

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2G EARTHQUAKES WHICH HAVE AFFECTED THE SITE AREA WITH A MODIFIED MERCALLI INTENSITY OF IV OR GREATER

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2H REPORT ON SEISMIC SURVEY-PROPOSED SHOREHAM POWER STATION LONG ISLAND LIGHTING COMPANY

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2I LABORATORY SOILS TESTS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2J SUMMARY REPORT OF GEOTECHNICAL STUDIES OF REACTOR BUILDING FOUNDATION

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2K AIRCRAFT CRASH PROBABILITY STUDY

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2L REPORT ON SERVICE WATER SYSTEM SOILS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2M REPORT ON DENSIFICATION OF SERVICE WATER SYSTEM SOILS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

2N HURRICANE STUDY

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIMENT, AND SYSTEMS

3.1 CONFORMANCE TO GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS (10CFR Part 50, Appendix A)

The General Design Criteria (GDC), contained in the Shoreham USAR Section 3.1, were reviewed to establish those criteria that may be applicable to the storage of SNPS low burnup cycle spent fuel in the spent fuel pool. The following GDC are addressed:

I. Overall Requirements

GDC1	Quality Standards and Records
GDC2	Design Bases for Protection Against Natural Phenomena
GDC3	Fire Protection
GDC4	Environmental and Dynamic Effects Design Bases

II. Protection by Multiple Fission Product Barriers

GDC13	Instrumentation and Control
GDC17	Electric Power Systems
GDC18	Inspection and Testing of Electric Power Systems
GDC19	Control Room

IV. Fluid Systems

GDC44	Cooling Water
GDC45	Inspection of Cooling Water System
GDC46	Testing of Cooling Water System

VI. Fuel and Radioactivity Control

GDC60	Control of releases of radioactive material to the environment
GDC61	Fuel storage and handling and radioactivity control
GDC62	Prevention of criticality in fuel storage and handling
GDC63	Monitoring fuel and waste storage
GDC64	Monitoring radioactivity releases

The following GDC were found <u>not</u> to be applicable to a defueled reactor:

I <u>Overall Requirements</u>

GDC5 Sharing of structures, systems, and components

Shoreham is a single unit, thus the above criterion does not apply.

II <u>Protection By Multiple Fission Product Barriers</u>

GDC10	Reactor Design
GDC11	Reactor Inherent Protection
GDC12	Suppression of reactor power oscillations
GDC14	Reactor Coolant Pressure Boundary
GDC15	Reactor Coolant System
GDC16	Containment Design

The above criteria do not apply because the reactor and primary containment are not operable.

III <u>Protection And Reactivity Control Systems</u>

GDC20 - 29 requirements apply only to an operating reactor protection and reactivity control systems

IV <u>Fluid Systems</u>

GDC 30-43 address reactor and containment systems required for power operation only.

V <u>Reactor Containment</u>

GDC 50- 57 address the primary containment design which is no longer required for a defueled reactor.

<u>Applicable Criterion Conformance</u>

<u>Quality Standards and Records (Criterion 1)</u>

<u>Criterion</u>

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Design Conformance

Structures, systems, and components are classified in Section 3.2. The LIPA QA program described in DSAR Chapter 17 assures that quality practices and documentation are maintained commensurate with the classification that is identified in this Defueled Safety Analysis Report (DSAR).

Design Basis for Protection Against Natural Phenomena (Criterion 2)

Criterion

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

Design Conformance

The spent fuel racks, fuel pool, and reactor building which are required to maintain the SNPS fuel in a safe condition are designed to withstand natural phenomena as described in the USAR. Because of the low burnup condition of the SNPS Cycle 1 spent fuel, the need for support systems is limited (see Chapters 9, 15). Natural phenomena are described in Chapter 3 of the Shoreham USAR.

Fire Protection (Criterion 3)

Criterion

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Design Conformance

This criterion is satisfied by the SNPS fire protection program which is described in Section 9.5.1 of this report and the USAR.

Environmental and Missile Design bases (Criterion 4)

Criterion

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Design Conformance

Chapter 15 of this report defines accidents that are applicable to spent fuel storage and fuel handling. The spent fuel is stored in the spent fuel storage pool. The pool structure, Reactor Building, and spent fuel racks provide passive safety protection from missiles or other conditions that could cause fuel mechanical damage. The structural design basis of the fuel storage racks is discussed in Chapter 9 of the USAR. Additional information on the design of structures, systems, and components can be found in Chapter 3 of the Shoreham USAR.

Instrumentation and Control (Criterion 13)

Criterion

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Design Conformance

Instrumentation is provided to monitor spent fuel pool level and temperature as well as fuel pool cleanup. Instrumentation is provided for process and effluent radiation monitoring, area and airborne radiation monitoring, and accident monitoring. Radiation monitoring is maintained as described in DSAR Chapters 11 and 12.

Electric Power Systems (Criterion 17)

Criterion

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss of coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Design Conformance

The criterion applies principally to the design of an operating reactor. As demonstrated in DSAR Chapter 15, active system are not required to provide cooling or makeup functions in the event of postulated accidents including a seismic event. However, operability of the electric power system will be required by Technical Specifications during fuel movement to provide for a controlled and monitored release capability in the event of a fuel drop accident. One offsite power transmission system will be maintained to provide power for support system operation. In addition, blackstart combustion turbines exist nearby at Shoreham-West to provide reliable power in the unlikely event of a loss-of-offsite power occurs. One non-safety Emergency Diesel Generator will be provided during fuel handling operations. A further discussion of electric power requirements can be found in Chapter 8.

Inspection and Testing of Electric Power Systems (Criterion 18)

Criterion

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the conditions of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Design Conformance

Electric Power Systems will be tested and inspected in accordance with SNPS operating procedures and Technical Specifications. See Criteria 17 response.

Control Room (Criterion 19)

Criterion

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and

(2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Design Conformance

A control room is provided and equipped to operate the plant safely under normal and accident conditions.

Based on the results of radiological analyses provided in DSAR Chapter 15 control room shielding and ventilation functions are not required for the mitigation of postulated accidents. Instrumentation available in the control room for accident monitoring and support system control are described in DSAR Chapter 7.

Cooling Water (Criterion 44)

Criterion

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink, shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power operation (assuming onsite power is not available) the system's safety function can be accomplished, assuming a single failure.

Design Conformance

As demonstrated in Chapter 15 of this report, active cooling of the spent fuel pool is not required based on the low heat generation rate of the low burnup spent fuel. Service water and other support systems are expected to be normally available to provide plant building services; however, these systems do not fulfill a safety function.

Inspection of Cooling Water System (Criterion 45)

Criterion

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Design Conformance

The service water system which will be maintained functional is designed to permit appropriate visual inspection in order to assure the integrity of system components. See Criterion 44 response.

Testing of Cooling Water System (Criterion 46)

Criterion

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of full operational sequence that brings the system into operation for reactor shutdown and for loss of coolant accidents, including operation of applicable portions of the protection systems and the transfer between normal and emergency power sources.

Design Conformance

See Criterion 44 response.

Control of Releases of Radioactive Materials to the Environment (Criterion 60)

Criterion

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Design Conformance

Because SNPS is not in normal operation, effluent releases are due primarily to maintenance of the spent fuel pool water quality. Means are provided to control and/or hold up the release of liquid and gaseous effluents as required. Fuel pool cleanup and appropriate radwaste systems are provided and are described in Chapters 9 and 11. See also Criterion 61.

Fuel Storage and Handling and Radioactivity Control (Criterion 61)

Criterion

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed, (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Design Conformance

Fuel Storage and Handling

The low burnup SNPS spent fuel is to be stored in the spent fuel storage pool located in the reactor building. The fuel racks and fuel pool structure are Seismic Category I. Systems required for safe fuel storage will be subject to appropriate inspection and testing requirements.

Adequate shielding is provided by maintaining a minimum water depth over the active fuel. Dose rates at the refueling level without the effects of shielding were calculated to be approximately 1R/HR.

The SNPS Secondary Containment is a Seismic Category I controlled leakage building surrounding the fuel pool facility. The Reactor Building Normal Ventilation System (RBNVS) will be used to provide ventilation and a monitored release pathway. Because the gas activity present in the fuel and available for release is primarily noble gas (Kr-85), the filtering role of the Reactor Building Standby Ventilation System (RBSVS) is not required. Certain components of the RBSVS are needed to support operation of the RBNVS. These components will remain functional to provide these services. As discussed in Chapter 15, credible potential releases from accidents are small in comparison to 10CFR100 limits, and neither the Reactor Building Standby Ventilation System nor secondary containment integrity is required to reduce offsite doses due to postulated accidents.

Radiation monitoring is provided as described in Chapter 11 and 12 to detect radiological releases.

Because of the extremely low residual heat load (approximately 550 watts) associated with the SNPS spent fuel, active fuel pool cooling is not required. Reliable fuel pool makeup sources including condensate storage, demineralized water, and fire protection water, are capable of maintaining pool water inventory to compensate for evaporation. Chapter 9 contains a complete discussion of makeup requirements.

The fuel pool is a Seismic Category I structure. Systems that connect to the pool (fuel pool cooling, fuel pool cleanup, etc.) have been designed to minimize the potential for draining of the pool inventory. High and low level alarms indicate pool water level changes in the main control room.

Radioactive Waste Systems

The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal of all radioactive liquids and solid waste produced as a result of spent fuel storage. The off-gas system is not needed. Any Krypton 85 will be retained within the fuel cladding. Should pin-hole leaks develop, the gases will be handled by the ventilation systems. They will be discharged to atmosphere via the main plant vent. The radiological consequences of this type of release are negligible. This accident is bounded by the analysis of the Fuel Handling Accident (Section 15.1.36).

Liquid radwastes are collected, classified, and treated as high conductivity, low conductivity, chemical or laundry wastes. Processing includes filtration, ion exchange, analysis, and dilution. Wet solid wastes are packaged in steel containers or polyethylene high integrity containers. Dry solid radwastes are compressed and/or packed in steel drums or boxes.

Accessible portions of the spent fuel pool area and radwaste building have sufficient shielding to maintain dose rates within the limits set forth in 10CFR20 and 10CFR100. The radwaste building is designed to preclude accidental release of radioactive materials to the environs above those allowed by the applicable regulations.

The fuel storage and handling and radioactive waste systems are designed to assure adequate safety under normal and postulated accident conditions. The design of these systems meets the requirements of Criterion 61.

Radwaste systems are designed to meet the limits for effluents set forth in 10CFR20 and 10CFR50.

Prevention of Criticality in Fuel Storage Handling (Criterion 62)

Criterion

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Design Conformance

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in spent fuel storage is prevented by the geometrically safe configuration of the storage rack. There is sufficient spacing between the assemblies to assure that the array, when fully loaded, is substantially subcritical. Fuel elements are limited by rack design to only top loading and designated fuel assembly positions.

Spent fuel is stored under water in the spent fuel storage pool. The racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. Spent fuel is maintained at a subcritical multiplication factor k-eff of less than 0.95 for both normal and abnormal storage conditions.

The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel and to minimize the possibility of mishandling or misoperation.

The use of geometrically safe configurations for new and spent fuel storage and the design of fuel handling systems precludes accidental criticality in accordance with Criterion 62.

For further discussion, see the following section:

Section 9A Criticality Analysis

Monitoring Fuel and Waste Storage (Criterion 63)

Criterion

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas, (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and (2) to initiate appropriate safety actions.

Design Conformance

Appropriate systems have been provided to meet the requirements of this criterion. A malfunction of the fuel pool cleanup system is alarmed in the main control room. It is also alarmed in the radwaste control room on high pressure differential. Alarmed conditions include high/low fuel pool level. The refueling level ventilation exhaust radiation monitoring system detects abnormal amounts of radioactivity. As demonstrated in Section 9A and Chapter 15 active cooling of the spent fuel pool is not required because of the low heat generation rate.

Area radiation and sump levels are monitored and alarmed to give indication of conditions that may result in excessive radiation levels in the fuel storage and radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

Monitoring Radioactivity Releases (Criterion 64)

Criterion

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Design Conformance

Means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences. The following station release pathways are monitored:

1. Gaseous releases from the station ventilation exhaust
2. Liquid discharge to the discharge tunnel

Radioactivity levels in the normal plant effluent discharge paths and in the environment are continually monitored during normal conditions by the various radiation monitoring systems and by the offsite radiological environmental monitoring programs.

The semiannual Effluent Release Report is submitted to the NRC. This report includes specific information on the quantities of the principal radionuclides released to the environment.

Additional discussion of radiation monitoring is contained in Chapters 11 and 12.

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Seismic Category I structures, systems, and components are those necessary to ensure:

1. The integrity of the reactor coolant pressure boundary

2. The capability to shut down the reactor and maintain it in a safe shutdown condition

3. The capability to prevent or mitigate the consequences ofaccidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

   Criteria 1 and 2 do not apply to a defueled reactor with respect to the storage and handling of low burnup Shoreham spent fuel. A set of postulated accidents has been identified and analyzed in Chapter 15 of this report that defines the potential for a radiological release. Based on this analysis it has been concluded that potential radiological releases are far below the exposure limits of 10CFR100. The analysis in Chapter 15 of this report assumes that the structural integrity of the filled fuel pool, fuel pool liner, reactor building structure and fuel racks together form a passive safety system that requires a seismic Category I designation. The Category I designation has been maintained for fuel handling equipment as well.

   A reclassification of structures, systems, and components is provided in DSAR Table 3.2-1. Table 3.2-1 supplements the information provided in USAR Table 3.2.1-1. The quality group classification in USAR Table 3.2.1-1 reflects the original design basis. As analyzed in Chapter 15, active cooling of the spent fuel pool is not required and pool makeup requirements are minimal. Supporting systems are required to maintain building habitability, provide radiation monitoring capability, and normal operating service functions.

Design Basis Earthquakes (DBE) and Operating Basis Earthquakes (OBE) are described in the Shoreham USAR Section 2.5.

Structures, systems, and components whose safety functions require conformance to the quality assurance requirements of 10CFR50, Appendix B, are summarized in Table 3.2-1 under the heading, LIPA Quality Assurance Category, with the notation I.

Modifications to QA Category II equipment and components at and above the 175' elevation in the reactor building shall be designed to withstand the DBE without failing in a manner that would result in an unacceptable impact to the spent fuel pool integrity or unacceptably damage the spent fuel whereby a public health and safety concern could be created.

A key of definitions is provided at the end of Table 3.2-1. Chapter 17 discusses the graded level of Q.A. requirements for this equipment.

3.3 WIND AND TORNADO LOADING

The information contained in the USAR remains the same although the requirements to protect safe-shutdown equipment no longer exists.

3.4 WATER LEVEL (FLOOD) DESIGN

The design of flood-protected structures remains the same although the requirements to protect safe-shutdown equipment no longer exist.

3.5 MISSILE PROTECTION

The design information contained in this section is unchanged. However the spent fuel pool is the only area of the plant requiring missile protection.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED RUPTURE OF PIPING

In the defueled state high energy piping systems inside primary containment listed in USAR Table 3.6.1A-1 are no longer pressurized and thus piping rupture need not be postulated.

3.7 SEISMIC DESIGN

Seismic design methods remain the same; however, hydrodynamic load effects resulting from safety relief valve discharge and loss-of-coolant-accidents are no longer applicable for a defueled reactor.

3.8 DESIGN OF SEISMIC CATEGORY I STRUCTURES

The design methods for seismic Category I structures such as the reactor building will remain as described in USAR Section 3.8 except that Safety Relief Valve (SRV) and LOCA hydrodynamic loads are no longer applicable to a defueled reactor.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section addresses methods and procedures used to qualify mechanical equipment. The information contained in this section is relevant only to reactor operating conditions and is, therefore, not applicable to the DSAR.

In the future, mechanical equipment will be accorded the safety significance demonstrated by the classification in Table 3.2-1 of the DSAR.

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Seismic Category I equipment is identified in Table 3.2-1 and is limited to structures and equipment required to maintain the integrity of the fuel in the spent fuel pool. As discussed in Section 3.2, only the Reactor Building, fuel pool, fuel racks, and fuel handling equipment are required to be Seismic Category I. The instrumentation described in USAR Section 3.10 is no longer required to be seismically qualified.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Electrical Equipment Environmental Qualification

Purpose

The purpose of the Electrical Equipment Environmental Qualification Program for Shoreham is to provide assurance that electrical equipment important to safety as defined by 10CFR50.49 located in potentially harsh environments maintains functional operability when required to mitigate the consequences of a postulated accident or to bring the plant to a cold shutdown condition afterward. Since the fuel has been removed and stored in the fuel pool, LOCA or HELB cannot occur (see Chapter 15), and there is no potential for creation of harsh environment (i.e., the remaining design basis accidents discussed in Chapter 15 do not result in harsh environments). Based on these conditions, 10CFR 50.49 is not applicable, therefore the environmental qualification program is not required. Environmentally qualified electrical equipment will be designated Q.A. Category II.

3.12 SEPARATION CRITERION FOR SAFETY RELATED MECHANICAL AND ELECTRICAL EQUIPMENT

The systems described in this section are no longer required to fulfill a safety related function regarding the storage of spent fuel. Thus, there no longer exists a need to maintain separation criteria for these systems. Q.A. Category I equipment will be designated Q.A. Category II.

3A Computer Programs for the Stress Analysis of Category I Structures, Dynamic and Static Analysis, and Dynamic and Stress Analysis of Seismic Category I Piping Systems

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

3B NRC Regulatory Guides

This section is described in the USAR. Specific topics are covered elsewhere in this DSAR.

3C Pipe Failure Outside Primary Containment

In the defueled state, piping systems outside primary containment which were considered high energy systems are no longer pressurized. Pipe rupture need no longer be postulated.

TABLE 3.2-1

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

SYSTEM/ COMPONENT	LIPA QUALITY ASSURANCE CATEGORY	SEISMIC CATEGORY	COMMENTS
I. Reactor System	II	N/A	NR
II Nuclear Boiler	II	N/A	NR
III Recirculation System	II	N/A	NR
IV Control Rod Drive Hydraulic System	II	N/A	NR
V Standby Liquid Control System	II	N/A	NR
VI Neutron Monitoring	II	N/A	NR
VII Reactor Protection	II	N/A	NR
VIII Fixed Process, Airborne, and Effluent Radiation Monitors	II	N/A	(1)
IX RHR	II	N/A	NR
X Core Spray	II	N/A	NR
XI HPCI	II	N/A	NR
XII RCIC	II	N/A	NR
XIII Fuel Service Equipment			
1. Fuel preparation machine	I	I	
2. General purpose grapple	I	I	
XIV Reactor Vessel Service Equipment			
1. System Line Plugs	II	N/A	NR
2. Dryer & Separator sling and RPV head strongback	I	I	
3. Drywell head lifting rig	I	I	

TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

	SYSTEM/ COMPONENT	LIPA QUALITY ASSURANCE CATEGORY	SEISMIC CATEGORY	COMMENTS
XV	In-vessel Service Equipment			
1.	Control rod grapple	I	I	
XVI	Refueling Equipment			
1.	Refueling platform	I	I	(4)
2.	Refueling bellows, drywell	II	N/A	
3.	Refueling bellows, cavity reactor	II	N/A	
4.	New Fuel Inspection Stand	II	N/A	NR
XVII	Storage Equipment			
1.	New Fuel Storage Racks	II	N/A	NR
2.	Defective fuel storage container	I	I	
3.	Spent fuel pool, dryer/sep. pool, reactor cavity liners	I	I	
4.	Spent fuel storage racks	I	I	
XVIII	Radwaste System	II	N/A	
XIX	Reactor Water Cleanup System	II	N/A	NR
XX	Fuel Pool Cleanup Subsystem			
1.	Demineralizer vessel	II	N/A	
2.	Filters	II	N/A	
3.	Pumps, purification & transfer	II	N/A	
4.	Piping	II	N/A	
5.	Valves	II	N/A	
6.	Tanks, backwash storage and air accumulator	II	N/A	

TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

SYSTEM/ COMPONENT	LIPA QUALITY ASSURANCE CATEGORY	SEISMIC CATEGORY	COMMENTS
XXI Fuel Pool Cooling Subsystem			
1. Piping	II	N/A	
2. Valves	II	N/A	
XXII Control Room Panels			
1. Electrical modules	II	N/A	
2. Cable	II	N/A	
XXIII Local Panels			
1. Electrical modules	II	N/A	
2. Cable	II	N/A	
XXIV Offgas System	II	N/A	NR
XXV Service Water System	II	N/A	
XXVI Compressed Air System	II	N/A	
XXVII Onsite Power Systems (USAR safety related)			
a. Diesel Emergency Power Systems	II	N/A	(2)
b. AC Power Systems	II	N/A	
c. Containment Electrical Penetrations	II	N/A	NR
d. Fire Stops	II	N/A	
e. DC Power Systems	II	N/A	
XXVIII Primary Containment Atmosphere Control	II	N/A	NR
XXIX a) Reactor Building Normal Ventilation	II	N/A	
b) Reactor Building Standby Ventilation	II	N/A	NR*

* Certain components such as fans and valves wil[illegible]ain functional to support RBNVS operations.

TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

SYSTEM/ COMPONENT		LIPA QUALITY ASSURANCE CATEGORY	SEISMIC CATEGORY	COMMENTS
XXX	Primary Containment Purge	II	N/A	NR
XXXI	Power Conversion	II	N/A	NR
XXXII	Condensate Storage and Transfer	II	N/A	
XXXIII	Emergency Support Facilities			
	1. TSC Bldg.	II	I	
	2. EOF	II	N/A	NR(3)
	3. OSC	II	N/A	
XXIV	MSIV Leakage Control	II	N/A	NR
XXXV	Miscellaneous			
	1. RB Polar Crane	I	I	(4)
	2. ECCS Loop Level	II	N/A	NR
XXXVI	Reactor Building Closed Loop Cooling	II	N/A	NR
XXXVII	Equipment and Floor Drains	II	N/A	
XXXVIII	Miscellaneous Ventilation Systems			
	1. 125 Volt DC Battery room H & V	II	N/A	
	2. Screenwell pumphouse H&V	II	N/A	
	3. Relay and emergency switchgear H&V	II	N/A	
	4. Control room air conditioning, including filter trains	II	N/A	
	5. Diesel generator room ventilation	II	N/A	

TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

SYSTEM/ COMPONENT	LIPA QUALITY ASSURANCE CATEGORY	SEISMIC CATEGORY	COMMENTS
XXXIX Area Radiation Monitoring System			
1. All components	II	N/A	
2. High Range Area	II	N/A	NR
XL Leak Detection System	II	N/A	NR
XLI Fire Protection System			
1. Water spray deluge systems	II	N/A	
2. Sprinklers, carbon dioxide systems	II	N/A	
3. Portable and wheeled extinguishers	II	N/A	
XLII Civil Structures			
1. Reactor building	I	I	
2. Office and service building	II	N/A	
3. Screenwell	II	N/A	
4. Control building	II	N/A	(5)
5. Turbine building	II	N/A	(5)
6. Intake Canal	II	N/A	
7. Discharge tunnel	II	N/A	
8. Discharge pipe and diffuser	II	N/A	
9. Radwaste Building	II	N/A	(5)
10. Auxiliary boiler and MG set building	II	N/A	
11. Biological shielding	II	N/A	(5)
12. Missile barriers	II	N/A	
13. Waterproof doors	II	N/A	
14. Site grading	II	N/A	
15. Masonry walls (RB)	II	N/A	(5)
16. Masonry walls (non-RB)	II	N/A	

TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

SYSTEM/ COMPONENT	LIPA QUALITY ASSURANCE CATEGORY	SEISMIC CATEGORY	COMMENTS
XLIII Primary Containment Structure	II	N/A	5
XLIV Safety Parameter Display System	II	N/A	NR
XLV Post Accident Sample System	II	N/A	NR
XLVI Containment Isolation Valve Position Indicator	II	N/A	NR
XCVII Accident Monitoring Instrumentation (NUREG 0578)	II	N/A	NR

TABLE 3.2-1
(Continued)

KEY

Quality Assurance Category:

I - Meets 10CFR50 Appendix B requirements (same as USAR).

II - Meet requirements of industrial and engineering standards (commercial grade quality).

Seismic Category

I - Equipment is designed in accordance with the seismic requirements for the DBE/OBE.

N/A - Seismic requirements for DBE/OBE earthquake are not applicable to the equipment.

Comments:

NR - Not required (System secured from service or not required to support safe storage or handling of spent fuel).

(1) - Seismic events will not create a radiological release due to passive protection provided by the spent fuel pool.

(2) - Loss-of-offsite power will not create the potential for a radiological release as discussed in Chapter 15.

One emergency diesel generator will be maintained non-safety related operable, as required by Technical Specifications during fuel movement.

(3) - Based on LIPA Defueled Emergency Preparedness Plan, the EOF is not required.

(4) - Only structurally safety related.

(5) - Originally constructed as Seismic Category I; modifications will be analyzed for DBE to ensure integrity of Reactor Building.

CHAPTER 4

REACTOR

This Chapter includes reactor description, mechanical design, nuclear design, thermal and hydraulic design, reactor materials and control rod drive housing supports. In the plant's defueled condition, the fuel is not in the core and the reactor is depressurized. All sections of this Chapter are, therefore, not applicable to the DSAR. Fuel storage is addressed in DSAR Chapter 9. In particular, Section 9A addresses criticality and Section 9B addresses fuel pool make-up requirements.

4.1 REACTOR SUMMARY DESCRIPTION

The NSS system is no longer needed for the defueled condition and hence is depressurized.

4.1.1 Reactor Vessel

The reactor vessel design and description are covered in USAR Section 5.4.

4.1.2 Reactor Internal Components

The reactor internal components are as described in the USAR. The fuel rods and control rods are removed from the reactor.

4.1.3 Reactivity Control System

This system is no longer needed as there is no fuel in the reactor vessel.

4.1.4 Analysis Techniques

The description contained under this heading in the latest revision of the USAR is no longer relevant in the plant's defueled condition.

4.4 THERMAL AND HYDRAULIC DESIGN

The linear heat generation rate (LHGR) limit of 13.4 kw/ft will not be exceeded by the decaying fuel in the spent fuel pool. Justification for this limit can be found in Appendix A, of General Electric Standard Application for Reactor Fuel (GESTAR II).

4.5 REACTOR MATERIALS

Neither the Control Rod System or Reactor Internal materials are of importance to the defueled plant conditions.

4.6 CONTROL ROD DRIVE HOUSING SUPPORTS

There is no fuel in the vessel in the defueled state and hence this system is not of concern.

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 GENERAL

Because of the Defueled Plant Configuration, there is no longer a need for engineered safety features (ESF) systems at Shoreham. This is substantiated by a review of the Design Basis Accidents and Postulated Transients. These are covered in Chapter 15.

This chapter discusses the effect of radiological accidents in the Secondary Containment. The Secondary Containment is utilized for maintaining a controlled and monitored release point for the design basis accident, the Fuel Bundle Drop accident. In addition, a worst case release of the entire gaseous inventory of the fuel is postulated in Chapter 15 that bounds any possible large scale mechanical-damage event.

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Design Basis

6.2.1.1.1 Safety Criteria

The primary containment system is not required and will not be maintained functional as there will be no fuel within the primary containment structure. The secondary containment will maintain a subatmospheric pressure for postulated radiological accidents to assure radiological monitoring of building releases. It is not needed to mitigate the consequences of an accident.

6.2.1.1.2 Design Basis Accidents

The major design basis accident identified which will affect the secondary containment is the Fuel Handling Accident (Fuel Bundle Drop). The results of this accident from a radiological standpoint are presented in Chapter 15. There are no pressure and temperature effects of this accident and the RBNVS would continue to maintain a subatmospheric condition.

The other event which would have an effect on the secondary containment is the loss of normal AC.

A loss of normal AC power may result in loss of the subatmospheric conditions within the secondary containment and a loss of spent fuel pool water makeup capability. However, as explained in Chapter 15, should the loss of AC power occur as part of any event which results in fuel damage, while the radioactive release to the atmosphere would not be monitored, the offsite dose consequences to the public would be insignificant. With regard to loss of spent fuel pool water makeup capability, evaporative loss would be so slow that corrective action would be taken before loss of shielding is significant. There are no radiological consequences associated with the loss of normal AC power.

6.2.1.2 System Design

The reactor building, which completely encloses the primary containment and acts as the secondary containment, is maintained at subatmospheric pressure by the RBNVS.

6.2.1.3 Design Evaluation

This entire subsection is not applicable as it deals with the primary containment which is no longer maintained.

6.2.2 Containment Heat Removal System

This subsection is not applicable as it deals with the primary containment which is no longer maintained.

6.2.3 Containment Air Purification and Cleanup Systems

This subsection is not applicable as it deals with the filtration portion of the RBSVS which is no longer required.

6.2.4 Containment Isolation System

This subsection is no longer applicable as it deals with the primary containment isolation system. The primary containment is no longer maintained.

6.2.5 Combustible Gas Control in Containment

This subsection is no longer applicable as it is concerned with hydrogen combustion inside the primary containment.

6.3 EMERGENCY CORE COOLING SYSTEMS

The emergency core cooling systems protect the core against hypothetical pipe breaks of various sizes. In the plant's present state, the fuel is not in the core and the reactor is depressurized. Therefore, pipe breaks are not postulated and the emergency core cooling systems are not required and this section is not applicable to DSAR.

6.3.2.2.3 <u>Core Spray System</u>

The Core Spray (CS) System is described in the USAR. In the defueled status of the Shoreham Nuclear Power Station the CS System serves no function and is no longer maintained.

6.4 HABITABILITY SYSTEMS

The systems, aside from the control room air conditioning portion, are no longer maintained because they are not needed since the fuel is stored in the spent fuel pool. The control room air conditioning system is described in Section 9.4.1.

6.5 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

The main steam isolation valve-leakage control system (MSIV-LCS) is not required in the defueled state and is, therefore, not included in the DSAR.

6.6 OVERPRESSURIZATION PROTECTION

The overpressurization protection system is not required in the defueled state and is, therefore, not included in the DSAR (See Chapter 5. of DSAR).

6.7 MAIN STEAM LINE ISOLATION VALVES

The main steam isolation valves (MSIVs) are not required in the defueled state and are, therefore, not included in the DSAR (See Chapter 5. of DSAR).

6.8 CONTROL ROD DRIVE SUPPORT SYSTEM

The control rod drive support system is not required in the defueled state and is, therefore, not included in the DSAR (See Chapter 4 of DSAR).

6.9 CONTROL ROD VELOCITY LIMITERS

The control rod velocity limiters are not required in the defueled state and this Section is, therefore, not included in the DSAR (See Chapter 5 of DSAR).

6.10 MAIN STEAM LINE FLOW RESTRICTORS

The main steam line flow restrictors are not required in the defueled state and this Section is, therefore not included in the DSAR, (See Chapter 5. of DSAR).

6.11 REACTOR CORE ISOLATION COOLING SYSTEM

The RCIC system is not required in the defueled state and is, therefore, not included in the DSAR (See Chapter 5. of DSAR).

6.12 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system is not required in the defueled state and is, therefore, not included in the DSAR (See Chapter 4 of DSAR).

7.1.1.1.6 Reactor Manual Control System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.7 Reactor Vessel Instrumentation

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.8 Reactor Recirculation System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.9 Feedwater Control System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.10 Pressure Regulator and Turbine-Generator Controls

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.11 Remote Shutdown System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.12 Screenwell Pumphouse Ventilation System

The screenwell pumphouse ventilation system instrumentation and controls remain functional and are designed to ventilate each of the two rooms of the building using separate, 100 percent outside air ventilation systems.

7.1.1.1.13 Process Computer System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.14 Reactor Core Isolation Cooling System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.15 Standby Liquid Control System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.16 Reactor Water Cleanup System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.17 Leakage Detection System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.18 Reactor Shutdown Cooling Mode-RHR System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.19 Radwaste System

Radwaste system instrumentation and controls support manual processing and disposing of the radioactive process wastes.

7.1.1.1.20 Emergency Diesel Generators

This system is utilized to provide backup emergency power. One emergency diesel generator will be operable when fuel is being handled in the secondary containment.

7.1.1.1.21 Turbine Building Closed Loop Cooling Water System

The turbine building closed loop cooling water (TBCLCW) system instrumentation and controls remain functional to maintain the turbine building cooling water system at design temperature and monitor system performance. The TBCLCW system also cools the equipment in the radwaste building and supports the station air compressors.

7.1.1.1.22 Service Water System

The service water system provides cooling for the plant components. Instrumentation and controls for this system are provided to operate the system in accordance with Section 9.2.

7.1.1.1.23 Recirculation Pump Trip System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.24 Reactor Building Standby Ventilation System

The filtration portion of the system is not needed to support the storage of the fuel in the fuel pool. Certain fans and air operated valves will remain functional to support RBNVS operation. See DSAR section 9.4 for additional information.

7.1.1.1.25 Reactor Building Closed Loop Cooling Water System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.26 Primary Containment Atmospheric Control System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.27 Fuel Pool Cooling and Cleanup Systems

Fuel pool cooling and cleanup systems instrumentation and controls remain unchanged except that the cooling portion is not required because evaporative cooling is sufficient to remove the small amount of decay heat.

7.1.1.1.28 Control Room Air Conditioning System

The control room air conditioning (CRAC) system instrumentation and controls for one of the two redundant subsystems are functional to maintain the main control room at design temperature during normal and emergency conditions, monitor system performance, and permit manual as well as automatic initiation of an air supply fan.

7.1.1.1.29 Chiller Equipment Room Ventilation System

This system remains operable to service the chiller equipment room located on the 63' elevation of the control building.

7.1.1.1.30 Diesel Generator Room Emergency Ventilation Systems

This system is needed to support the operation of the emergency diesel generator during movement of fuel in the secondary containment.

7.1.1.1.31 Relay Room, Emergency Switchgear Rooms, And Computer Room Air Conditioning System

The relay room, emergency switchgear rooms, and computer room air conditioning system instrumentation and controls for one of the two redundant subsystems are maintained functional to automatically control the ventilation system to maintain these rooms at their design temperature and system performance.

7.1.1.1.32 Battery Room Ventilation System

The battery room ventilation system instrumentation and controls automatically control and monitor the ventilation system to maintain the battery room at its design temperature and monitor system performance. Each of the three battery rooms has its own ventilation system which will remove any generated hydrogen.

7.1.1.1.33 Containment Spray and Suppression Pool Cooling

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.34 Rod Sequence Control System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.35 Motor Control Center Room Ventilation System

The motor control center (MCC) room ventilation system instrumentation and controls are maintained functional to provide automatic control of the ventilation system to maintain the room at design temperature for habitability. Each of the two MCC rooms in the reactor building has its own ventilation system.

7.1.1.1.36 Motor Generator Room Ventilation System

The motor generator (MG) room ventilation system instrumentation and controls remain functional to maintain the room at design temperatures for habitability. Each of the four MG rooms in the reactor building has its own ventilation system.

7.1.1.1.37 Compressed Air System (SRV Accumulators)

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.1.38 Main Steam Isolation Valve Leakage Control System

This system is not needed to support the storage of the fuel in the fuel pool, therefore it is not included in the DSAR.

7.1.1.2 Classification

Section 3.2 provides a reclassification of systems based on their importance to safety.

CHAPTER 8

ELECTRIC POWER

8.1 INTRODUCTION

This chapter describes the details of the plant auxiliary power distribution system which is designed to provide adequate electrical power to all plant equipment. The defueled condition of the plant does not require the operation of any Class 1E power system. However, as stated in Section 8.3.1 item 2, a diesel generator and associated equipment shall remain operable while fuel handling is taking place.

8.1.1 Utility Grid

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged in the defueled condition. For further information on this subject refer to the USAR.

8.1.2 Interconnection To Other Grids

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged in the defueled condition. For further information on this subject refer to the USAR.

8.1.3 Offsite Power System

While in the defueled condition the offsite power system provides power to all operating plant equipment. Power to the Shoreham Nuclear Power Station is provided from the LILCO system through 138KV or 69KV circuits. The 138KV switchyard is arranged in a two bus configuration with circuit breakers and switches arranged to permit isolation and/or repair of either bus section. Four 138KV circuits enter into the switchyard (two per bus) each containing a circuit breaker at the connection to its respective bus. Two separate rights-of-way are provided, each containing two of the 138KV circuits. The 69KV circuit from the Wildwood substation enters the site sharing one of the aforementioned rights-of-way for a distance of one mile. This circuit, however, is mounted on separate towers and is separated from the 138KV circuits. The detailed description of the remaining offsite system remains as described in the USAR except as follows:

Three Brookhaven 80MW (each) Combustion Turbine units are located on LILCO SNPS property approximately 3600 feet fr... the 138KV switchyard. These units are connected into one of the 138KV Holbrook transmission lines and are available to provide an additional source of onsite power to the SNPS. (see figure 8.2.1-2)

The spare Reserve Station Service and Normal Station Service transformers will no longer be required.

8.1.4 On Site AC Power System

The station electrical power system includes electrical equipment and connections required to provide power to and control the operation of electrically driven station equipment in the defueled condition. A non-safety emergency diesel generator will provide backup AC power during fuel handling in the secondary containment.

8.1.5 On Site DC Power System

During the defueled condition, the 125V DC distribution systems do not have a safety function. However, a DC distribution system will be maintained operable during fuel handling operations. It will remain functional at other times.

The 24V DC power source will no longer be required. This system provides power to the Nuclear Source and Intermediate Range Instrumentation which is no longer in service in the defueled condition.

8.1.6 Identification of Safety Related Systems

The description contained under this heading in the latest revision of the Shoreham USAR will not be applicable in the defueled state.

Table: 8.1.6-1 Identification of Safety Loads

The basis for these tabulations no longer exists. The electrical distribution system will remain in service to maintain power to plant equipment on the site in the defueled condition.

8.1.7 Identification of Safety Criteria

The description contained under this heading in the latest revision of the Shoreham USAR is not applicable in the defueled state.

Table 8.1.7-1 Regulatory Design Criteria For Electric Power

The basis for these tabulations no longer exists. The electrical distribution system will remain in service to maintain power to plant equipment on the site in the defueled condition.

8.2 OFFSITE POWER SYSTEM

8.2.1 Description

The description contained under this heading in the latest revision of the Shoreham USAR remain unchanged except as follows:

Service buses 101, 102 and 103 are not required to be maintained as safety related while in the defueled condition. They are reclassified as Category II.

8.2.1.1 One Line Diagrams and Physical Drawings

The information contained under this heading in the latest revision of the Shoreham USAR remains unchanged in the defueled condition.

8.2.1.2 Transmission Line

The description contained under this heading in the latest revision of the USAR remains unchanged in the defueled condition except that the safety related function of the busses (1R22-SWG-101, 102, and 103) no longer exists. They are reclassified as Q.A. Category II systems.

8.2.1.3 Station Switchyard

The description contained under this heading in the latest revision of the USAR remains unchanged in the defueled condition. For further information on this subject refer to the USAR.

8.2.1.4 Transmission Line Exits

The description contained under this heading in the latest revision of the USAR remains unchanged in the defueled condition except for the following:

The new Brookhaven Combustion Turbines are added to the existing transmission line configuration. (Figure 8.2.1-2)

8.2.2 Analysis

The basis of the analysis no longer exists. The analysis as described in the USAR is not required in the defueled condition.

8.3 Onsite Power Systems

The plant power system is designed to provide an adequate source of electrical power to all systems required to be operational in the defueled condition.

C.3.1 AC Power Systems

The general description of the plant electrical power (AC) systems is as provided in this section of the USAR. However, the safety related design criteria are no longer applicable. The following does apply:

1 - Equipment, switchgear, or buses built and designed to safety standards are not maintained as safety related but will be inspected in the defueled condition since they are required for the diesel to be classified as operable.

2 - One diesel generator set shall remain operable during fuel handling in the secondary containment.

3 - Required surveillances and tests will be performed in accordance with the Technical Specifications.

4 - Adequate equipment protection and emergency measures are available for the required plant electrical systems in the defueled condition.

The equipment, switchgear, and buses have been reclassified to Q.A. Category II. Therefore, safety functions such as auto-start, redundancy, etc., are no longer required.

8.3.2 DC Power Systems

8.3.2.1 Description

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged in the defueled condition except as follows:

1 - The 24V DC system, providing power to source and intermediate range nuclear instrumentation, is no longer used.

2 - All class 1E/safety related functions of the DC system are no longer classified as such.

The batteries are being maintained for those systems remaining functional or operable in the defueled condition. Q.A. Category I equipment is now Q.A. Category II.

9.1.2.5 Radiological Considerations

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

9.1.3 Fuel Pool Cooling and Cleanup System

All of the equipment in this system will be retained for operation, but in a modified manner. Since the fuel pool cooling subsystem is designed to remove the decay heat produced by spent fuel assemblies, as described in the USAR, and only a negligible amount of heat is expected to be generated from the slightly irradiated spent fuel bundles stored there, the cooling mode is not required. Thus reactor building closed loop cooling water is not required.

Appendix 9B provides an evaluation of spent fuel pool makeup requirements.

However, the spent fuel pool cooling subsystem will be used in the makeup mode in order to provide normal makeup water to the fuel pool from the condensate storage tank using the condensate transfer and storage system. Alternate makeup sources for the spent fuel pool are Demineralized and Makeup Water System, and Fire Protection Water System. The makeup mode is described at the end of USAR paragraph 9.1.3.2.1.

The fuel pool cleanup subsystem will be used as designed.

The fuel pool cannot be inadvertently drained because the pump suctions for the fuel pool cooling and cleanup system are taken above elevation 168, or about 7 feet below the normal water level. If a break occurred in these lines, about 18 feet of water would remain above the fuel in the pool. This is more than enough to provide adequate shielding. Pump returns to the pool are equipped with siphon breakers to prevent inadvertant pool drainage.

9.1.4 Fuel Handling System

9.1.4.1 Design Basis

See USAR. This section is identical to the USAR.

9.1.4.2 Equipment Description

See USAR. This section is identical to the USAR.

9.1.4.3 Description of Fuel Transfer

The fuel handling system provides a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. The preceding subsection describes the equipment and methods used in fuel handling. The following paragraphs describe the integrated fuel transfer system, which ensures that the design bases of the fuel handling system and the requirements of Regulatory Guide 1.13 are satisfied.

9.1.4.3.1 Arrival of Fuel On Site

No new fuel is expected to arrive on site. Therefore this section of the USAR is not required.

9.1.4.3.2 Refueling Procedure

No refueling is planned. Therefore this section of the USAR is not required.

9.1.4.3.3 Departure of Fuel from Site

This section applies as written in the USAR.

In addition:

1. The spent fuel will be removed from the site in certified fuel shipping casks.

2. The casks will be leak tested prior to shipment.

The remainder of USAR Section 9.1.4 is applicable.

9.2 WATER SYSTEMS

9.2.1 Service Water System

The Service Water (SW) System is as described in USAR Sections 9.2.1.1 thru 9.2.1.5 with the following changes because of the reduced heat removal requirements with the plant in the de-fueled state.

a) The RBSW system is considered non-safety related because it does not provide cooling water to any plant equipment required to perform a safety function.

b) One RBSW pump will supply cooling water to one RBSVS/CRAC chiller condenser, an emergency diesel generator, and to all Turbine Building Service Water (TBSW) cooling loads. (See item e below.) No service water is required for RHR, RBCLCW, drywell cooling, and makeup water to the reactor vessel ultimate cooling connection (UCC). The testable check valve in the UCC will not require testing to verify forward flow. Emergency service water to the spent fuel pool is not required (per DSAR Chapter 15) because of the very low heat generation by the fuel.

c) Automatic start/initiation due to accident signals are not required.

d) The double isolation valves which split the RBSW from the TBSW subsystems will be opened to intertie the subsystems.

e) Normal operation will now consist of only one RBSW pump in use because of the minimal heat load imposed by the TBCLCW system to support the station air compressors. It will supply cooling water to one TBCLCW heat exchanger, and the circulating water pump bearing. Cooling water for the vacuum priming pump seal cooler is not required. The second RBSW pump will remain in standby.

f) The TBSW pumps are out of service since they are no longer required.

g) Table 9.2.1-1 has been revised.

9.2.1.5 <u>Instrumentation Application</u>

This section remains unchanged except that only the instrumentation needed for the Service Water System as described in 9.2.1 a) through g) is required.

9.2.2 <u>Reactor Building Closed Loop Cooling Water (RBCLCW) System</u>

This system is not needed to support the storage of fuel in the spent fuel pool.

9.2.3 <u>Makeup Water Demineralizer System</u>

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged except as follows:

1. SBLC, RBCLCW, seal water injection, and vacuum priming are no longer users of demineralized water in the defueled condition.

2. The HPCI suction line from the condensate storage tank is not required to be maintained as safety related in the defueled condition.

9.2.9 Reactor Building Standby Ventilation And Control Room Air Conditioning Chilled Water System

Redundancy in this system is not needed since neither the RBSVS nor CRAC systems are safety related in the defueled condition. The heat loads generated by the electrical equipment in the control room, relay room and the emergency switchgear room are greatly reduced, such that only one chiller is required to maintain the control room, relay room and switchgear room at design conditions. The operating chiller and associated pumps will be manually controlled from the control room. This system has been reclassified QA Category II. Aside from the above, the system design remains unchanged and further information can be found under the above heading in the Shoreham USAR.

9.3 PROCESS AUXILIARIES

9.3.1 Compressed Air Systems

The description contained under this heading in the latest revision of the USAR remains unchanged in the defueled condition except for the following:

1. Piping that has been installed as ASME III code class 2 is no longer considered safety related and is reclassified QA Category II.

2. Nitrogen will no longer be used for inerting the primary containment or for equipment within the primary containment.

3. Safety related functions of the compressed air system no longer exist. No pneumatically operated valves are required for safe shutdown.

For further information on the compressed air system, refer to the USAR.

9.3.2 Process Sampling System

[illegible] Process Sampling System provides monitoring of certain [illegible] operations while fuel is in the spent fuel pool for [illegible] short or long term storage. The process monitoring is [illegible]lished as necessary by means of measuring, analyzing and/or recording for conductivity, pH, and silica concentration, as shown on DSAR Table 9.3.2-1.

9.3.3 Equipment and Floor Drainage System

With the Reactor defueled and the fuel assemblies stored in the Fuel Pool, large portions of the Equipment and Floor Drainage System are not required.

System Description

This system is described in the USAR. Changes in status are addressed below.

Reactor Building

The only source of radioactive waste to the Equipment and Floor Drainage System in the Reactor Building is the Fuel Pool and associated service equipment leakage. Sources in the USAR that are no longer applicable are the Drywell Equipment Drain System and the Reactor Recirculation Pumps Drainage System. The Drywell Equipment Drain Tank is no longer required. One or more floor drain sumps are no longer required, as applicable.

Turbine Building

The Turbine Building Floor Drain and Equipment Drain Systems are no longer required, as applicable, except for the Decontamination Sump drains and associated equipment. There is no steam and the turbine is no longer required, so that the only source of radioactive waste is the Chemical Laboratory.

Radwaste Building

The Radwaste Building Equipment and Floor Drainage System is maintained operational. The Dirty Waste Sump and Pumps (1N52-TK 114 and 1N52-P-187A/B) and Regenerant Recovery Sump and Pumps (1N52-TK-115 and 1N52-P-181A/B) are no longer required.

9.3.4 Chemical, Volume Control, and Liquid Poison Systems

The Standby Liquid Control System is no longer required in the defueled condition. The RWCU System is also no longer required unless the Reactor is layed up wet.

9.3.5 Failed Fuel Detection System

With the fuel in the pool, the description in the USAR Section is no longer applicable.

In the event of gross fuel rod failure in the fuel pool (see "Worst Case Fuel Damage Accident" in DSAR Chapter 15), the refueling floor process radiation monitors will detect this radioactivity if it becomes airborne.

9.3.6 Suppression Pool Pumpback System

This system not required to support storage of fuel in the fuel pool.

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 Control Room Air Conditioning System

The Control Room AC system remains unchanged in design and operating functions. However, the system is reclassified to QA Category II, the filter portion of the system will no longer be required and one of each of the redundant fans and ACUs will no longer be required. The AC system will only function to provide an OSHA environment for the operators during the fuel storage period. This requires the operation of only one RBSVS/CRAC chiller. Automatic initiation systems and interlocks for the habitability portion of the system will be non-operable and the AC system will be manually controlled from the control room. For further discussion on this system refer to the Shoreham USAR.

9.4.2 Reactor Building Normal Ventilation System

9.4.2.1 Design Basis

The RBNVS remains unchanged in design and operating function except that the system will only:

1. Provide ventilation by introducing filtered outside air into the reactor building at a rate of approximately 2.7 air changes per hour

2. Remove heat generated by solar and external heat transmission, lighting and, the fuel pool.

3. Induce slight negative pressure in the reactor building to prevent potentially contaminated air from escaping from the building without being monitored.

The RBNVS may be operated in a recirculation mode in order to control Reactor Building humidity. This helps to protect equipment from damage due to corrosion. While operating in the recirculation mode, the operating functions, as discussed above, are maintained except that supply air is limited to infiltration caused by the negative pressure inside the Reactor Building.

For further discussion on this system refer to the USAR.

9.4.3 Radwaste Building Ventilation

The description contained under this heading in the latest Shoreham USAR remains unchanged, except that the charcoal exhaust filtration system is no longer required and one of the two redundant supply and exhaust fans, mechanical refrigeration units and circulating pumps are also no longer required. Refer to the USAR for information on this subject.

9.4.4 Turbine Building Ventilation System And Station Exhaust System

A) Turbine Building Ventilation System

This system is not required to support the storage of fuel in the spent fuel pool.

B) Station Exhaust System

This system will expel the exhaust air from the radwaste building and the reactor building. However, only two fans will be required for this purpose, one fan operating and one fan on standby. This will ensure that the Isokinetic nozzles located in the upper level of the exhaust duct will see a sufficiently high velocity to be operational. For further discussion regarding this system refer to the Shoreham USAR.

9.4.5 Battery Room Heating And Ventilation

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject. This system is reclassified to Q.A. Category II.

9.4.6 Drywell Air Cooling System

This system is not needed while the fuel is stored in the spent fuel pool.

9.4.7 Screenwell Pump House Heating And Ventilation

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject. This sytem is reclassified to Q.A. Category II.

9.4.8 Plant Heating

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

9.4.9 Primary Containment Purge System

This system is not needed while the fuel is stored in the spent fuel pool.

9.4.10 Diesel Generator Room Ventilation

The description contained under this heading in the latest revision of the Shoreham USAR is revised. This system is reclassified to Q.A. Category II and nonseismic. The system is no longer safety related and the design bases for tornado missile protection and room temperature control are no longer applicable.

9.4.11 Relay Room, Emergency Switchgear Room And Computer Room Air Conditioning System

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged with the exception that only one train of equipment will remain functional. Refer to USAR for information on this subject. This system is reclassified to QA Category II.

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 Fire Protection System

9.5.1.1 Design Basis

The design basis section applies with the following addition:

The basic premise of the fire protection discussions in the USAR and FHAR is protection from fire for safety related areas including areas containing equipment or circuits that are (1) required for safe shutdown, or (2) required to prevent or mitigate radiological releases comparable to 10CFR 100 limits. Since safe shutdown is assured by non-operation of the plant, and all of the nuclear fuel is in the fuel storage pool, the only remaining safety related area is the Reactor Building. Structures, systems components and administrative controls in place to protect areas, equipment or circuits previously identified as safety related will be maintained as required for property loss prevention purposes and should be considered the same as those fire protection features described in the USAR for protection of non-safety related areas.

Three documents which were used in the design of the plant's fire protection features and continue to be part of the fire protection prc jram are:

1. Evaluation of the SNPS Fire Protection Program as compared to 10CFR50 Appendix R criteria submitted via SNRC 572 dated May 21, 1981.

2. Fire Hazards Analysis Report.

3. Cable Separation Analysis Report:
SNRC 532 dated February 10, 1981
SNRC 811 dated April 13, 1983

However, the term "safety related", as used in those documents and in USAR section 9.5.1, applies only to the Reactor Building.

Section 6 of the Fire Hazards Analysis Report (FHAR) contains technical requirements that formerly were fire protection technical specifications.

FHAR Chapter 6 reflects reductions in the technical requirements that are consistent with the text of this DSAR Section).5.1.

Types of Fires

The "types of fires" section applies with no changes.

Design Criteria

The "design criteria" section applies with the following addition:

As discussed above, this design will be maintained for property loss prevention purposes. However the "safety related" application of the listed documents, particularly NRC's Branch Technical Position APCSB 9.5-1 and Appendix A thereto, is limited to the Reactor Building.

Locations of Fires

The "locations of fires" section applies with the following changes:

The rooms listed parenthetically as examples of safety related areas having a concentration of cables are reclassified to Q.A. Category II. The rooms listed as examples of where oil fires could occur near safety related equipment no longer fit that description because these areas are reclassified to QA Category II. Furthermore, the fire hazard associated with this equipment is significantly reduced while the equipment is not being used because the ignition sources associated with the operating equipment have been eliminated.

Intensity of fires

This section applies without change.

Fire Characteristics

This section applies without change.

Building Arrangement and Structural Features

The "building arrangement and structural features" section applies with the following changes:

In the response to NRC question 3, as shown in FHAR revision 3, SNPS has stated our intention to replace existing motorized fire dampers with newly designed fire dampers. All of the areas where these new dampers were to be installed are in the Control Building and are reclassified to Q.A. Category II. Therefore, this proposed modification will not be implemented. The CO_2 systems for those rooms are in electric lockout. When a fire is detected, the CO_2 system controls would cause the dampers to close on an electrical signal. As a backup, the fusible link of each of the existing fire dampers is sufficient to cause closure of a damper in the event of a fire, thus assuring integrity of the fire barriers.

In contrast with this USAR section, an unprotected HVAC opening exists in the east wall of each of the three diesel generator rooms within 50 feet of an oil-filled (Reserve Station Service) transformer. This deviation was reported to the NRC on Licensee Event Report 87-021. The proposed corrective action was to install a deluge water curtain system below the existing missile shield wall between the transformer and the wall openings. Since the diesel generator rooms are reclassified to QA Category II, this modification will not be implemented. The partial protection provided by the missile barrier is considered sufficient for non-safety related areas.

Seismic Design

This section applies without change.

Water Requirements

The "water requirements" section applies with the following additional statement:

Although some areas previously identified as safety related are reclassified to QA Category II, the water supply is not being reduced.

Codes and Standards

This section applies without changes. SNPS will continue to meet the requirements of the applicable NFPA codes for fire protection systems that remain functional.

9.5.1.2 System Description

The "System Description" section applies with the following changes:

As discussed earlier, all fire protection features remain in place. Several rooms/areas listed in this section as safety related are reclassified to Q.A. Category II. Essential circuitry installed for safe shutdown of the plant is no longer needed for that purpose. No removal of such cable or change in its physical separation is contemplated. Similarly, the service water line inside the Reactor Building, where a spare connection exists for manual hookup to the fire protection water system, is reclassified to Q.A. Category II. Modifications that would degrade its seismic design are not contemplated at this time.

9.5.1.3 Safety Evaluation

Electrical Insulation Fires

This section applies without change.

Charcoal Fires

This section applies without change.

Oil Fires

The "oil fires" section of the safety evaluation applies with the following change:

As discussed earlier, the fire hazards associated with non-operating equipment are significantly reduced because the primary ignition sources - electrical energy and hot surfaces - are eliminated.

Severity, Intensity and Duration of Fires

This section applies without changes.

Time Estimates

This section applies without changes.

Failure Mode and Effects Analysis

This section applies without changes.

Accidental Initiation of Fire Protection System

The "accidental initiation of fire protection system" section applies with the following change:

Areas protected by CO_2 systems are among those that are no longer considered safety related.

Single Failure in Fire Protection Systems

This section applies without change.

Pipe Breaks in Fire Protection Systems

This section applies without changes.

Failure of Fire Protection System Affecting Safety Related Equipment

This section applies with the following change:

Of the areas listed, only the Reactor Building is still considered safety related.

9.5.1.4 Tests and Inspections

This section applies without changes.

9.5.1.5 Personnel Qualification and Training

This section applies without changes.

9.5.2 Communications System

9.5.2.1 Design Bases

This section of the USAR remains unchanged.

9.5.2.2 System Description

This section of the USAR remains unchanged except for the following:

1. For the very low frequency (VLF) portable radio systems, one low-powered VLF radio base station will be used in conjunction with two mobile car units to provide offsite radio communications (instead of two VLF base stations and four mobile car units).

2. The Emergency Operations Facility (EOF) is not required, since no emergency requiring EOF activation can occur with the fuel in the Spent Fuel Pool.

9.5.2.3 Tests and Inspections

This section of the USAR remains unchanged.

9.5.3 Lighting Systems

While in the defueled condition this system will provide all the necessary required lighting to the plant and the site. The description of this system in the USAR remains unchanged except for the following:

1. Section 9.5.3.2, item #2 - the standby AC lighting system will receive power from plant service buses which are powered from offsite.

2. Same section, item #5 - the fifth lighting subsystem will receive power from DC battery sources while the plant remains in the defueled condition.

3. The last paragraph of the same section, the independent power sources for lighting, remains unchanged but the source of power will be from plant service buses and DC battery sources if needed.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

An emergency diesel generator is required to be operational when irradiated fuel is being handled in the reactor building. Sections 9.5.4-9.5.7 in the USAR remain descriptive of the EDG auxiliary systems except that these systems and their components are classified as QA Category II. Also, the requirements of redundancy to prevent malfunction or failure of these systems and their components, i.e., fuel storage tanks, fuel pumps, air start tanks, etc. and 7-Day operability Post-LOCA are no longer applicable.

Statements in Sections 9.5.4-9.5.7 indicating that portions of these systems are designed to ASME Boiler Pressure Vessel Code, Section III, Code Class 3, that they meet Seismic Category I requirements, and that the concrete block house and the two-foot thick concrete slab above the fuel storage tanks are Seismic Category I and provide missile protection are also no longer applicable. The USAR description of equipment design with respect to applicable codes is representative of the original design of these systems but these designs which were applicable to safety related equipment in an operating nuclear power plant will no longer be maintained as safety related equipment, based on the DSAR Chapter 15 safety analysis.

9.5.5 Diesel Generator Cooling Water System

9.5.6 Diesel Generator Starting System

9.5.7 Diesel Generator Lubrication System

9.5.8 Primary Containment Leakage Monitoring System

With the fuel in the Spent Fuel Pool, the Primary Containment Leakage Monitoring System is not required.

9.5.9 Storage of Gases Under Pressure

The quantities and type of gases stored in pressurized containers in the defueled condition is reduced from that previously on hand. The design bases remain unchanged. Storage facilities are provided for the following gases as shown in Table 9.5.9-1:

1. Carbon Dioxide for fire protection.
2. Halon 1301 for fire protection.
3. Air for instrument, control, breathing and service.
4. Nitrogen for glycol and HW heating.
5. Propane for auxiliary boiler ignition.

The following gases are no longer used or required to be stored in the defueled condition:

1. Hydrogen for main generator.
2. Hydrogen and oxygen for gas analyzers.
3. Nitrogen for containment inerting.
4. Nitrogen for drywell floor seals.
5. Nitrogen for electrohydraulic control.
6. Air for MSIV accumulators (inboard and outboard).
7. Air for long term accumulators.

The statement in the USAR relative to maintenance and laboratory gases remain unchanged. The safety evaluation discussed in section 9.5.9.3 of the USAR is only applicable for air for instruments, service breathing, and control and for carbon dioxide and halon. Statements relative to the pressure relief valves and gas release hazards remain as discussed in the USAR. Gas use for safe shutdown is no longer necessary in the defueled condition.

Appendix

9A FUEL CRITICALITY ANALYSIS

The Shoreham Spent Fuel Rack (SFR) is of a stainless steel and water neutron flux trap design which uses no additional poison. A description of the storage racks is provided in 9.1.2. The criticality analysis of this rack design is described in detail in Appendix 9A of the Shoreham USAR. The reactivity results which are summarized in USAR Table 9A-4 remain valid for the conditions existing at Shoreham after defueling. Furthermore, due to the differences in U-235 enrichment between the SFR designed and the current Shoreham fuel, a large negative reactivity credit should be taken into account. This is explained as follows:

The Shoreham SFR design is based on a maximum U-235 enrichment of 3.1 wt. %. The resulting basic cell k is calculated to be 0.9129 without uncertainty and model adjustments (Table 9A-4, Appendix 9A, Shoreham USAR). The Shoreham Cycle 1 fuel loading uses three (3) enrichments. Of the 560 fuel assemblies in the core, 340 bundles have the highest bundle average U-235 enrichment of 2.19 wt. %, 144 bundles of 1.76 wt. % and 76 remaining bundles uses natural uranium.

If the six inch natural uranium segments at the top and bottom of the fuel are excluded, the average enrichment of a 2.19 wt. % bundle becomes 2.33 wt. %. Using this enrichment and linearly extrapolating the reactivity vs. U-235 enrichment results given in Figure 9A-5 of Appendix 9A, Shoreham USAR, the reactivity difference between the SFR design enrichment of 3.1 wt. % and the current maxiumum loading enrichment of 2.33 wt. % is found to be about -6.0% in k_{eff} (k_{eff} -0.060). This brings the basic cell k_{eff} under nominal storage conditions for the current fuel down to 0.85, which is well below the regulatory acceptance criterion of k_{eff} 0.95. All the corrective and uncertainty adjustments listed in Table 9A-4 of the Shoreham USAR remain applicable.

During the period from July, 1985 to June, 1987, Shoreham went through three separate stages of low power testing (less than 5% of rated power), which resulted in a total core exposure of approximately 48 MWd/MT as determined by a series of core-follow analyses. The net effect of the core exposure is a slight decrease in reactivity (-0.002 in k_{eff}) mainly due to the offsetting contributions from the formation of Sm-149 and the slight depletion of the burnable Gd poison in the fuel bundles. In light of the large reactivity margin described previously (k_{eff} 0.85), no additional credit will be claimed here.

9B EVALUATION OF SPENT FUEL POOL MAKEUP REQUIREMENTS

An analysis was performed to determine the rate of water loss from the spent fuel pool through evaporation under the worst case scenario described below. The following conservative assumptions are used in the analysis to maximize the calculated pool evaporation rate:

1) The spent fuel pool temperature is 110°F.

2) The ambient temperature above the spent fuel pool is conservatively assumed to have zero relative humidity.

3) The reactor building air flow exists due to normal ventilation system operation to maximize evaporation.

The result of the calculation shows that the maximum evaporation rate from the pool is approximately 0.6 gpm which translates to a pool level depletion rate of one foot per eleven days. Based on this very low maximum depletion rate, external cooling of the spent fuel pool is not required. Technical Specifications require that the water level above the spent fuel be a least twenty-one feet. In addition, it should be noted that pool water level is alarmed in the control room and alarm response procedures exist to provide appropriate operator action.

This section is no longer applicable since most of the waste streams would no longer exist.

11.2.2.2 Low Conductivity Waste Subsystem

Waste Collector Subsystem

This system will receive all the influents as stated in the USAR except that no inputs will be received from the Condensate Demineralizer System, Drywell Equipment Drain System and the Phase Separator Tanks.

11.2.2.3 High Conductivity Waste Subsystem

Floor Drain Subsystem

This system will not receive any influents from the Drywell Floor Drain System, the Turbine Building Floor Drain Sumps and the Condensate Demineralizer System. The Waste Evaporator will not be utilized to process this waste. Floor drain influents will be processed through the Floor Drain Filters.

11.2.2.4 Regenerant Chemical Subsystem

In this system the only equipment still required are the Chemical Waste Sump the Regenerant Liquid Evaporator Feed Tanks and their associated pumps. The regenerant evaporator is not required.

11.2.2.5 System Operational Analysis

The analysis described under this heading in the latest version of the USAR is not applicable in the defueled plant condition.

11.2.3 System Design

11.2.3.1 Equipment Description

This Section remains as presented in the USAR.

11.2.3.2 Applicable Codes and Standards

This Section remains as presented in the USAR.

11.2.3.3 Radwaste Building

This Section remains as presented in the USAR except that the Radwaste Building is now designated a Quality Assurance Category II, Non-Seismic Structure.

11.2.3.4 Liquid Radwaste Equipment Quality Group Classification

This Section remains as presented in the USAR.

11.2.3.4.1 Conditions and Assumptions

This accident (raised in USAR Section 11.2.3.4) postulates the simultaneous failure of the liquid radwaste system tanks in or associated with the radwaste building. These tanks hold the radioactivity and potentially radioactive liquid waste from the floor drains, equipment drains, nonradioactive chemical wastes, and processed liquid effluents. The tanks (and their capacities) that are assumed to fail are:

1. Waste collector tanks: Two at 25,000 gal each (Contents are insignificantly radioactive).

2. Floor drain tanks: Two at 25,000 gal each (Contents are insignificantly radioactive).

3. Regenerant liquid and evaporator feed tanks: Two at 25,000 gal each (contents are insignificantly radioactive).

4. Recovery sample tanks: Two at 25,000 gal each (located outside the radwaste building contents are insignificantly radioactive)

5. Discharge waste sample tanks: Two at 25,000 gal each (located outside the radwaste building)

6. Spent resin tank: One at 4,700 gal (Section 11.5)

The source concentrations in the above are described in DSAR Table 11.1-1.

11.2.3.4.2 Accident Description

The accident description can be considered as described in Section 11.2.3.4.2 of the USAR except the structure is now classed QA Category II.

11.2.3.4.3 Accident Analysis

This section remains as presented in the USAR except that:

1. A conservative airborne partition factor of 1.0E-03 is assumed for all isotopic activities listed in DSAR Table 11.1-1, with the exception of Tritium (H-3), for which it is assumed that all the activity evolves.

2. Ground release atmospheric dispersion factors are assumed, as given in USAR Table 15.1-3, for the EAB.

3. The breathing rate of persons offsite is assumed to be 3.47E-04 cubic meters per second, consistent with Regulatory Guides 1.3 and 1.25. For other age groups the breathing rate was obtained from the ratio of the maximum age group rates given in Regulatory Guide 1.109 (Reference J).

11.2.3.4.4 Results and Consequences

The doses resulting from the analysis described above are as follows:

	Dose, millirem		
Source	Whole body Gamma*	Beta Skin	Maximum Organ**
Spent Resin Tank	1.8E-05	2.7E-06	1.3E-03
Radwaste Filters	1.2E-07	1.7E-08	8.3E-06
Discharge Sample Tanks	3.1E-08	1.4E-08	7.7E-06
Totals	1.8E-05	2.8E-06	1.3E-03

* External & internal pathways; child is the limiting age group

** Teen is the limiting age group, and lung is the limiting organ

The consequences of the above postulated accident are clearly very low. These projected doses are far below those which justify Quality Group D, non-seismic qualification of radwaste equipment (i.e., 500 mrem whole body, or its equivalent to parts of the body), in Reg. Guide 1.26, Rev. 1, and Reg. Guide 1.29, Rev. 1.

11.2.3.5 Instrumentation & Control

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

11.2.3.6 Shielding Field Routed Pipe

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

11.2.4 Operating Procedures

Operating procedures including administrative control of liquid radwaste releases are as described in the USAR except the Radwaste Building is now classified as QA Category II.

11.2.5 Performance Tests

Performance tests of equipment are as described in the USAR, except for activity reduction factors (DF), which are no longer applicable. Only equipment that remains in operation will be periodically tested.

11.2.6 Estimated Releases

Liquid effluent releases are expected to be minimal with the fuel in the spent fuel pool. This is based on the fact that during the period from June 1988 through May 1989, only one release had an isotopic concentration greater than LLD.

The quantity of the annual release of contaminated liquids is conservatively estimated by noting that the discharge volume from SNPS is approximately 5,000,000 gallons per year. Assuming the effluent concentration is consistently equal to that found in the one sample above LLD (7.83E-08 uCi/cc of Co-60, from DSAR Table 11.1-1), the estimated release is:

1.5E-03 Ci/yr of Co-60

11.2.7 Release Points

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

11.2.8 Dilution Factors

Under the plant's present condition, service water or circulating water will be used, if necessary, for dilution so that the discharged effluent concentration in the Long Island Sound will not exceed that prescribed in 10CFR20, Appendix B, Table II, Column 2.

Treated radioactive effluents are collected in the discharge sample tanks. The filled tank is sampled, and then discharged at a maximum rate of 150 gpm for a period of approximately 2.5 hours. If necessary, the treated effluent is diluted with about 8000 gpm of service water prior to discharge into the sound. Thus, if necessary a dilution factor of approximately 50 may be obtained during actual discharge.

No credit is taken for the external dilution factor, i.e. the mixing ratio in the Sound, for service water.

11.2.9 Estimated Doses

Offsite doses due to liquid releases are expected to be minimal, as discussed in DSAR Section 11.2.6. An estimate of the yearly dose is conservatively obtained by assuming each batch liquid release contains the maximum batch activity concentration of 7.83E-08 uCi/cc of Co-60, and the release volume is approximately 5,000,000 gallons per year. Assuming no dilution, the resulting doses are as follows:

Whole Body	0.166 mrem	(adult)
GI-LLI	1.43 mrem	(adult)
Liver	0.074 mrem	(adult)

As noted in Section 11.2.8, service water dilution remains available as necessary.

11.3 GASEOUS WASTE SYSTEM

11.3.1 Design Objectives

With the fuel in the Spent Fuel Pool, the radioactive gaseous waste system is no longer required to meet either 10CFR20 or 10CFR50 Appendix I limits.

11.3.2 System Descriptions

With the fuel in the Spent Fuel Pool, and negligible amounts of radioactive halogens in the fuel, the radioactive waste sources described no longer apply, and the systems necessary to process them are not required.

Normal ventilation will be maintained in the Radwaste and Reactor Buildings with discharge through the station ventilation exhaust duct.

11.3.3 System Design

The process offgas system, which is the system described in USAR Sections 11.3.3, 11.3.4 and 11.3.5, is not required with the fuel in the Spent Fuel Pool.

11.3.4 Operating Procedures

11.3.5 Performance Tests

11.3.6 Estimated Releases

In the plant's present state, no releases of radioactive gaseous effluents are anticipated. This is evidenced by the fact that since the plant achieved initial criticality in 1985, there have been no recorded releases documented in the Semi-Annual Radiological Effluents Reports.

11.3.7 Release Points

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

11.3.8 Dispersion Factors

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

11.3.9 Estimated Doses

There will be no expected offsite doses because no releases of radioactive gaseous effluents are anticipated under the plant's present defueled state.

11.3.10 Unmonitored Release Points

The unmonitored gaseous release paths as described in the USAR would be expected to occur during normal plant operation. In the defueled condition some pathways do exist on loss of ventilation systems. Doses in such an event would be insignificant due to the low radionuclide inventory in the plant. (Ref: NED Safety Analysis No. 4170024)

11.4 PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM

The description contained under this heading in USAR only apply to those monitoring systems described in DSAR Section 12.3.4. Refer to the USAR for further information. The changes to the USAR relating to the Radiation Monitoring System for the defueled condition are described in DSAR Section 12.3.4.

Sampling for halogens is not needed in the defueled condition.

11.5 SOLID WASTE SYSTEM

11.5.1 Design Objectives

The description contained under this heading in the latest revision of the USAR remains unchanged as it is used to develop the basic, design criteria of the plant.

However, in the present plant configuration this system is no longer required except for the retractable fill pipes and the transfer carts in the cubicles (since no solidification of waste, per se, is needed). High Integrity containers (HICs) will continue to be used since some wastes will continue to be generated, and must be shipped. Also Dry Active Waste (DAW) will continue to be generated, and must be shipped. The volume of both will be significantly less than that given in the USAR.

It should be noted that waste will be generated from the Spent Resin Tank, Radwaste Filter and Floor Drain Filter, as described in Section 11.2, to be transferred directly into HICs or to a mobile solidification or dewatering vendor. The HICs are then transported by the transfer carts out of their cubicles to be handled by the overhead crane.

Tables 11.5.1-1D and 11.5.1.-2 thru 5 of the USAR are superseded by DSAR Table 11.1-1.

11.5.2 System Input: Source Terms

The actual radwaste source terms in the plant's defueled condition are as follows:

The combined activity concentration in the spent resin tank, radwaste filters, and the floor drain filter is assumed to equal the maximum in the most recent solid waste shipments during the period November-December 1988. DSAR Table 11.1-1 lists the activity concentrations of radionuclides.

Figure 11.5.2-1 no longer applies.

11.5.3 Equipment Description

11.5.3.1 General

The only equipment remaining in use in this system is as follows:

4,700 Gallon Spent Resin Tank (SRT)

For the defueled condition, this tank receives backwashed resin and filter media from the Radwaste Demineralizer and the Fuel Pool Cleanup Demineralizer and Filters. (This tank is also discussed in Section 11.2. It is included here since it is a direct feed to the Solidification system.)

The spent resin pump transfers the spent resin to HICs which are set on the Radwaste floor or in the pits in the floor. The HICs are then dewatered by portable air-operated diaphragm pumps which draw suction from specially designed piping internals in the HICs. When convenient, HICs may be dewatered while in the fill cubicles.

Baler

This equipment is furnished to compress miscellaneous dry active waste (DAW) into 55 gallon drums.

Transfer Carts and Fill Pipes

These carts position the HICs at various stations within the fill cubicle during filling and dewatering operations. These are filled from the Radwaste Filters and Floor Drain Filters through fill pipes.

A connection is provided to allow for solidification dewatering of resins by a mobile vendor.

No other equipment in this Section of the USAR is required.

11.5.3.2 Wet Wastes

The first paragraph of this Section of the USAR no longer applies. The second paragraph remains applicable.

11.5.3.3 Dry Wastes

This Section of the USAR is applicable, as some DAW will continue to be generated.

11.5.3.4 Irradiated Reactor Components

This Section of the USAR still applies.

11.5.3.5 Operating Procedures

This section of the USAR no longer applies except that:

1. SRT waste can be transferred into a high integrity container (HIC) where it can be dewatered by the in-house dewatering system to Federal and burial site limits. Ultimately, this waste will be shipped to burial sites.

2. The shipping container is located under the retractable fill pipe by first placing the container on the waste container transfer vehicle within its locating guides and then running the transfer vehicle to a preset position directly beneath the fill pipe. The fill pipe is lowered over the container and the fill pipe splatter shield entirely covers the container opening. The remotely operated fill pipe is powered in the vertical direction by pneumatic cylinders.

11.5.3.6 Instrumentation

All instrumentation in this section is no longer needed except for the radiation monitors.

11.5.4 Expected Volumes

This Section of the USAR is superseded by the following:

A conservative expected estimated volume of waste in HICs and carbon steel liners is 1,000 cubic feet per year buried volume. See DSAR Table 11.1-1 for activities.

This statement and Table together supersede Table 11.1-1A of the USAR.

DAW volume is conservatively estimated to be 1,000 cubic feet per year, buried volume. The DAW activity is negligible.

11.5.5 Packaging

The description contained under this heading in the latest revision of the USAR remains unchanged. Refer to the USAR for information on this subject.

11.5.6 Storage

The description contained under this heading in the latest revision of the USAR remains unchanged. Refer to the USAR for information on this subject.

11.5.7 Shipment

The description contained under this heading in the latest revision of the USAR remains unchanged. Refer to the USAR for information on this subject.

11.6 OFFSITE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

DISCUSSION

The objectives of SNPS' Offsite Radiological Environmental Monitoring Program [REMP] are to identify and measure plant generated radioactivity in the environment and to calculate the potential dose to the surrounding population. SNPS' REMP is designed to comply with the Plant's Offsite Dose Calculation Manual (ODCM) and NRC Regulatory Guide 4.15. REMP data is acquired by sampling various media in the environment and then analyzing these samples for radioisotopes; Tables 11.6.3-1 and 11.6.3-2 detail the REMP sampling/analyzing program. Since REMP results vary for each sample and location, several sampling locations were selected for each medium using available meteorological, land, and water use data. The range of analyses performed on a sample depend on the type of sample taken.

Sampling locations are designated as either indicator or control. Indicator locations provide representative measurements of radiation and radioactive materials for those exposure pathways and radionuclides (from SNPS) that lead to the highest potential radiation exposures. Control locations are placed sufficiently far from SNPS so that they will be beyond the measurable influence of SNPS or any other nuclear facility. This monitoring program implements Section IV.B.2 of Appendix I to 10CFR Part 50, by verifying that measured concentrations of radioactive materials and direct radiation are representative of the actual contamination levels and doses to the public.

SNPS' REMP has been subdivided over three distinct time intervals: Preoperational REMP (prior to SNPS' initially achieving criticality), Operational REMP (from initial criticality until removal of the fuel from the core), and Post-Defuel REMP (after the core was transferred to the spent fuel pool).

Preoperational REMP was performed to identify and determine background levels of environmental activity around SNPS. Preoperational REMP also served to verify that indeed the media being sampled and analyzed is sensitive to radiological fluctuations in SNPS' environs (indicator locations) and to provide effective monitoring of potential critical pathways.

Preoperational and Operational REMP samples within the aquatic environment included surface water, algae, fish, invertebrates (clams, lobsters, etc.) and sediment. The atmospheric environment was sampled for airborne particulates, iodine, and noble gases. Milk, potable water, precipitation, game and food products were obtained from the terrestrial environment. Direct radiation was measured using thermoluminescent dosimeters (TLDs). The range of analyses for each sample could include: gamma spectrometry, Sr-89 and Sr-90; I-131; H-3, gross beta, direct radiation and noble gases. Under Post-Defuel REMP, several of the above sample types, sampling locations and/or analyses are discontinued. The current Post-Defuel REMP program is outlined in Tabl s 11.6.3-1 & 11.6.3-2.

Preoperational REMP began in February 1977 and continued through 1984, although the official Preoperational REMP period; i.e. the time frame against which the data base from Operational REMP was compared, occurred during 1983 and 1984. The Operational REMP began on February 15, 1985 when initial criticality was achieved. Except for reactor operator training programs which required the reactor to operate at '0.0% power' (during January 1989), SNPS has not generated radioisotopes since the last 5.0% power test, completed on June 6, 1987. Comparisons between the above two phases of REMP were documented in each annual REMP report.

As of August 9, 1989, SNPS' core was transferred to the spent fuel pool as part of the agreement between LILCO, state and local governments not to operate Shoreham. This transfer prevents criticality from being reestablished. In addition, since SNPS' last 5.0% power test was completed during June 1987, per Ref. 9, with the exception of I-129 and Kr-85, all iodines and gaseous effluents have decayed away. Consequently, the surveillance requirements for SNPS' Post-Defuel REMP were reduced to below the operational level.

Justification for Reducing REMP to Post-Defuel Surveillance Levels

Pursuant to Reg Guide 4.1, once the initial core of the licensee has reached the point (in time),of maximum burnup, and the licensee has demonstrated (using results from environmental media or calculations) that the doses and concentrations associated with a particular pathway are sufficiently small (comparable to preoperational levels), then the number of media sampled in the pathway and the frequency of sampling may be reduced to operational Tech Spec requirements. Since (as of August 9, 1989) the core has been in the spent fuel pool, the initial core has "exceeded" the point of maximum burnup.

It should be noted that the concept of "normal" Tech Spec requirements as referred to in Reg. Guide 4.1, refers to a fully operational station with normal surveillance requirements. Reg. Guide 4.1 does not account for the unique condition at SNPS. Consequently, the justification for reducing REMP will be performed in two steps. Step one reduces Operational REMP to the level mandated when SNPS was to become operational. Step two reduces the surveillance program further, to the revised requirements corresponding to the defueled condition.

Dose calculations to SNPS' environs (1983 - 1988) were performed by analyzing positive concentrations of radioactivity detected in collected samples. Table 11.6.1-4 compares the radiological impact from each major pathway to the public during SNPS' preoperational and operational REMPs. Specifically, the radiological impact during SNPS' 5.0% power testing program (1985 - 1987) was compared to preoperational REMP.

In all cases, the calculated doses during both the operational and preoperational phases of REMP were comparable. Therefore, no environmental radioactivity was identified (during any of the 5.0% power tests) as having originated at SNPS. These results satisfy the criteria established in Reg. Guide 4.1 for reducing post-defuel REMP to the level originally mandated by SNPS' license. The sampling points not required by the license are:

1) Game;
2) Aquatic Plants;
3) Aquatic Sediment;
4) Rain Water; and
5) Noble Gases.

Justification for reducing REMP to the revised requirements (after the core was defueled) is given based on the above information; i.e., the measured environmental impact due to 5.0% power testing was comparable to that of preoperational REMP, and as of August 9, 1989, the core was removed from the reactor pressure vessel. SNPS' last 5.0% power test was completed on June 6, 1987, and per Ref. 9, with the exception of I-129 and Kr-85, all iodines and gaseous effluents have since decayed away. In addition, radwaste system activities are quite low (listed in DSAR Sections 11.1 & 12.2). As a result, the only remaining radioisotopes (and their release pathways) are:

	Isotope(s)	Source	Effluent Pathway
1)	Kr-85	Spent Fuel	Gaseous
2)	Solubles and Particulates	Radwaste	Gaseous and Liquid

SNPS' Post-Defuel REMP Surveillance Program Outline (Steps 1 & 2)

1) DIRECT RADIATION: Reduce from 36 to 18 locations
Quarterly Surveillance Frequency

2) AQUATIC

a. Aquatic Plants and Beach Sediments	- Delete, not required
b. Fish, Surface Water and Invertebrates	- Retain, may be impacted from liquid release path to L.I. Sound

Perform Semiannual surveillances as available

3) AIRBORNE

a. Iodine	- Delete, insignificant quantity
b. Particulates and Gross Beta	- Retain, particulates and and solubles still exist.
c. Noble Gas	- Delete Noble Gas, not required.

Quarterly Surveillance Frequency

4) TERRESTRIAL

a. Precipitation, Soil, and Game	- Delete, not Tech Spec required
b. Potable Water	- Delete, well water not impacted by discharges to L.I. Sound
c. Milk, Food products	- Retain, long lived particulates

Quarterly Surveillance for Milk, Annually for Food

SUMMARY/CONCLUSION

1) Examination of the radiological impact to REMP locations which are to be eliminated -- From 1983 (preoperational REMP) through 1988 (which encompasses SNPS' 5.0% power testing program) -- indicates no measured increase in environmental contamination; refer to Table 11.6.1-4.

2) As of August 9, 1989, the SNPS core was transferred to the spent fuel pool; thus, the initial core has reached maximum burnup.

3) Per Regulatory Guide 4.1, if the above two conditions are met, then the operational phase of REMP may be reduced to the requirements that were written when SNPS was to be operated as designed.

4) The post-operational REMP surveillance program may be reduced from Step 1 to the requirements as delineated in DSAR Table 11.6.3-1 (Step 2), developed after the SNPS core was transferred to the spent fuel pool, because:

   a) Criticality will not be reestablished at SNPS. As of August 9, 1989, no additional fission/activation products will be generated;

   b) SNPS' last 5.0% power test was completed on June 6, 1987, which means that with the exception of I-129 and Kr-85, all remaining gaseous effluents have decayed away; and

   c) the only possible release paths for the remaining soluble or particulate effluents is through either the spent fuel pool cleanup or makeup water systems (independent systems with no direct release path to the general public), or the radwaste treatment systems (liquid and gaseous pathways) through which effluents are being or could be processed.

11.6.1 Objectives of REMP

11.6.1.1 Preoperational REMP

The objectives of the Preoperational REMP were:

1. To identify and determine baseline radiological characteristics in the environment around SNPS (these background levels were then compared with data collected during actual plant operation);

2. To assure that the media being sampled and analyzed are sensitive to fluctuations in the radiological characteristics of the environs at SNPS, and to assure that REMP will be responsive to radioactive discharges from SNPS (i.e., to identify indicator locations and critical pathways);

3. To provide effective monitoring of critical pathways of radiological effluents to unrestricted areas; and

4. To train personnel and evaluate procedures, equipment and techniques which are utilized in the Operational and Post-Defuel phase of REMP, including emergency response capabilities.

The years 1983 and 1984 served as the official preoperational period, as stipulated in Reference 8. All data collected during this period were used in developing a baseline for ultimate comparison with operational data. From the levels and fluctuations of radioactivity analyzed in environmental samples it was concluded that sensitive indicators of radioactivity for the environment around SNPS had been selected. Sensitive indicators revealed minute quantities of radioactive fallout from the October 1980 atmospheric nuclear weapons test by the People's Republic of China during 1980 and 1981, in addition to radioactivity remaining from two decades of atmospheric testing. Airborne particulate samples registered an increase in gross beta levels, along with identifying the gamma emitting isotopes Zr-95, Nb-95, Ru-103 and Ce-141. Also in 1983 and 1984, REMP sampling identified low levels of iodine-131 in Port Jefferson Harbor area aquatic samples. This was attributed to local hospitals treating patients for thyroid carcinoma.

Along with these anomalies in the environment, expected normal background radioactivity was measured in REMP samples. Aquatic samples consisting of surface water, fish, invertebrates, aquatic plants and sediment were chosen and reflected the normal background radiation found in this environment. The atmospheric environment was sampled for airborne particulate matter, iodine, and noble gases. All airborne radioiodine analyses were below detectable levels. In addition, milk, potable water, game, food products, beach sediments and rain water were sampled. The esults obtained from the analyses of these samples were typical of the radioactivity values usually associated with samples of these types. All radioiodine analyses of milk were below etectable levels. Direct radiation levels were relatively low, and approximately the same at all locations. No unusual radiological characteristics were observed in the environs of SNPS during 1983 and 1984. A summary of the annual program results for 1983 and 1984 is given in USAR Tables 11.6.1-1 and 2.

11.6.1.2 Operational REMP

The objectives of Operational REMP were:

1) Identify and measure plant-related radioactivity in the environment for the calculation of potential doses to the public.

2) Identify excessive radionuclide concentrations of limited duration, so that appropriate action may be taken.

3) Determine the long-term variation in radionuclide concentration, or

4) determine the effects of plant effluents on the environment.

5) Comply with regulatory requirements and provide records to document compliance.

6) Comply with the REMP requirements as outlined previously.

Operational REMP used the Preoperational data base to identify plant-contributed radiation, and to evaluate the possible effects of radioactive effluents on the environment. The Preoperational and Operational phases of REMP were designed to comply with Regulatory Guide 4.15 (5) and the associated Branch Technical Position (4).

Analyses of the environmental samples show results (8) consistent with those found during the preoperational years (1983 - 1984). Sensitive indicators revealed minute quantities of radioacti.. fallout remaining from the October, 1980 atmospheric nuclear weapons test by the Peoples Republic of China. Radioactivity traces from the previous two decades of international above ground atomic bomb testing were also recorded. Radioactivity increases from the accident at the Soviet Union's Chernobyl Nuclear Power Plant (during April, 1986) were also measured. Along with these environmental anomalies, expected normal background radioactivity was measured in REMP samples between 1985 and 1988. USAR Table 11.6.1-3 summarizes results from REMP during 1985, and DSAR Table 11.6.1-4 presents a comparison of preoperatonal and operational REMP data from 1983 through 1988.

11.6.1.3 Post-Operational REMP

The objectives of Post-Defuel and Operational REMP are identical. Differences in the execution of Post-Defuel REMP account for both the permanent defueling of SNPS, and experience gained during the preoperational and operational REMP phases.

11.6.2 Potential Pathways

11.6.2.1 Liquid Effluent Pathways

The exposure pathways for liquid effluents are:

1. External exposure from radionuclides in water; and
2. Ingestion of fish and shellfish containing radionuclides.

The concentrations of radionuclides expected to be released to the service water are listed in Section 11.2. Dilution of these concentrations in Long Island Sound is discussed in Section 11.2.8.

USAR Section 11.6.2.1 contains detailed discussions about the projected doses from various liquid pathways. With the updated source terms as described in the DSAR (Sections 11.1 and 12.2), future doses from liquid pathways are expected to be a small fraction of the doses presented in the USAR. See DSAR Section 11.2.9 for dose calculations.

11.6.2.2 Gaseous Effluent Pathways

The exposure pathways for gaseous effluents are:

1) Submersion in a cloud of noble gas;
2) Drinking milk from a milking animal pastured in an areas of long-lived particulates;
3) Eating leafy vegetables on which particulates have deposited.

The calculated air dose (using REMP when SNPS was to operate as designed) at the north-northeast site boundary is 1.1 mrad/yr from gamma radiation and 1.2 mrad/yr from beta radiation. Doses from gaseous effluent pathways are summarized in USAR Table 11.6.2-3. Computational methods are discussed in Section 11.6.2.3.

A dairy survey is performed annually to determine the location of any milking animal within a 5-mile radius of SNPS. When a milking cow or goat is found, annual doses are calculated using either current meteorological or activity release data, in accordance with the methods specified in the Shoreham Offsite Dose Calculation Manual.

11.6.2.3 Dose Computational Methods

11.6.2.3.1 Liquid Effluent Pathways

The discussion contained in the latest version of the Shoreham USAR (Section 11.6.2.3.1) continues to apply.

11.6.2.3.2 Gaseous Effluent Pathways

The discussion contained in the latest version of the Shoreham USAR (Section 11.6.2.3.2) continues to apply.

11.6.3 Sampling Media, Locations, and Frequency

Typical Post-Operational REMP sampling locations and frequency are given in Table 11.6.3-1. These locations are described in Table 11.6.3-2 and are shown in Figures 11.6.3-1 and -2. By virtue of the liquid and gaseous effluents from the plant, REMP is divided up into four distinct categories: atmospheric, terrestrial, aquatic and direct radiation. Sampling media, locations, and frequencies are discussed in the following sections.

11.6.3.1 Sampling Media

11.6.3.1.1 Aquatic Environment

The aquatic environment is examined by analyzing samples of: 1) Surface water; 2) Fish; and Invertebrates. Surface water samples are taken in May and October using a Niskin Bottle. The samples are placed in new polyethylene bottles following three rinses with the sample medium prior to collection. When available samples of Winter Flounder, *Pseudopleuronectes americanus*, Windowpane, *Scophthalmus aquosus*, Sea Robin, Prionotus spp, Little Skate, *Raja erinacea*, Blackfish, *Tautog onitis* and Summer Flounder, *Paralichthys dentatus* are taken by trawl, sealed in plastic bags, frozen, and shipped to the analytical laboratory for analysis.

When available, invertebrate samples of American Lobster, *Homarus americanus*, Squid, *Loligo pealeii* and Channeled Whelk, *Busycon canaliculata* are collected by trawl. Channeled whelk are also collected using pots. These invertebrate samples are then sealed in plastic bags, frozen and shipped to the laboratory for analysis. Blue Mussels *Mytilus edulis* are collected by hand along jetties and soft-shell clams, *Mya arenaria* from Wading River are shelled and sealed in plastic bags, frozen and shipped to the analytical laboratory.

11.6.3.1.2 Atmospheric Environment

The atmospheric environment is examined by analyzing airborne particulates collected on Gelman Type A/E filters using low volume air samplers (approximately 1 cfm). The samplers used are equipped with vacuum recorders for sample volume correction and to indicate sample validity and maintenance problems when they occur. Should the sampler lose vacuum due to a leak the vacuum level reading will drop to zero. Since this may occur without a corresponding loss of electric supply the exact time of the maintenance problem will be evident on the recorder chart.

Sample volumes are measured using dry gas meters and corrected for differences between the actual pressure that the volume meter sees and the average atmospheric pressure. Sample volumes are corrected to standard pressure using average weekly barometric pressure (measured at Environmental Engineering Department, Melville) and air sampler vacuum readings. Time totalizers indicate the duration of tir the sample is taken.

11.6.3.1.3 Terrestrial Environment

The terrestrial environment is examined by analyzing samples of milk and food products. When available, milk samples are collected quarterly, except during the pasture season (May through October) when the sampling is increased to monthly. Milk samples are prepared for shipment in accordance with the instruction of the laboratory performing the analysis. Food products consisting of vegetables and fruit are collected from area farm stands and shipped fresh to the laboratory.

11.6.3.1.4 Direct Radiation

Direct radiation levels in the environs are measured with energy compensated calcium sulfate (CaS04:Dy) TLDs, each containing four separate readout areas. The TLDs are annealed prior to placement in the field. One TLD is placed at each of the 18 locations, and exchanged on a quarterly bases; these locations correspond to the 16 meteorological sectors in the general areas of the site boundary, plus two control locations (actual locations are listed in Table 11.6.3-1). The units are then packaged and shipped to the laboratory for analysis.

11.6.3.2 Sampling Locations and Frequency

Typical REMP sampling locations and frequency are given in Table 11.6.3-1. These locations are described in Table 11.6.5-2 and shown in Figures 11.6.3-1 and 11.6.3-2.

11.6.4 NOT USED IN THE DSAR (Data Incorporated Into Section 11.6.1)

11.6.5 Data Analysis, Presentation and Interpretation

The discussion contained in the latest version of the Shoreham USAR (Section 11.6.5, 11.6.5.1, and 11.6.5.2) continues to apply.

11.6.6 Program Statistical Sensitivity

The discussion contained in the latest version of the Shoreham USAR (Section 11.6.6) continues to apply.

REFERENCES In Section 11.6

1) Regulatory Guide 4.1 "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants"

2) Not Used

3) Not Used

4) Radiological Branch Technical Position, Rev. 1, Nov. 1979

5) Reg. Guide 4.15, Rev. 1, February 1979, "Quality Assurance For Radiological Monitoring Program (Normal Operation) Effluent Streams and the Environment"

6) SNPS Offsite Dose Calculation Manual
3/4.12 Radiological Environmental Monitoring
3/4.12.1 Monitoring Program Table 3.12.1-1 "REMP"

7) Not Used

8) SNPS' Operational REMP Annual Reports: January 1, to December 31, 1983, 1984, 1985, 1986, 1987, & 1988 issued by Nuclear Engineering and Environmental Engineering Departments of LILCO.

9) C-RPD-476, Rev. 0, 10/21/88, "SNPS Core Thermal Power After Shutdown"

TABLE 11.1-1
Radwaste Sources Greater than LLD

Spent Resin Tank, Radwaste Filter, & Floor Drain Filter

The activity concentration is assumed to equal the maximum in the most recent HIC shipment (Nov-Dec 1988) and is (From Reference 2):

Isotope	Activity Concentration, uCi/cc	% of Activity
*Cr-51	9.84E-04	58.46%
Mn-54	2.17E-05	1.29%
*Fe-55	4.19E-04	24.88%
*Co-57	7.92E-07	0.05%
Co-58	6.43E-06	0.38%
Co-60	1.09E-04	6.51%
*Fe-59	4.57E-05	2.71%
*Ni-63	6.41E-06	0.38%
*Sb-124	3.25E-06	0.19%
*Zn-65	1.89E-05	1.12%
H-3	6.21E-06	0.37%
*C-14	3.94E-07	0.02%
*Sr-90	1.69E-07	0.01%
*Zr-95	1.52E-05	0.91%
*Nb-95	2.55E-05	1.51%
*Tc-99	4.79E-09	0.00%
*I-129	7.32E-10	0.00%
*Cs-137	1.34E-06	0.08%
*Ce-144	2.95E-06	0.18%
*Pu-241	1.59E-05	0.95%

Discharge Waste Sample Tanks

The activity concentration in these tanks is assumed to equal the maximum concentration measured in the past 12 months preceding May 1989 (from Ref. 3):

Isotope	Activity Concentration, uCi/cc	% of Activity
Co-60	7.83E-08	100.0%

Note: The remaining radwaste tanks (floor drain collector tanks, waste collector tanks, and recovery sample tanks) were all determined in Reference 4 to have isotopic concentrations less than LLD.

* Calculated based on generic scaling factor.

TABLE 11.6.1.-4

Comparison Of Operational - Preoperational REMP Data

SAMPLE TYPE	Unit/Isotope	1988	1987	1986	1985	1984	1983
		(------ Operational REMP			------)	(-- Preoperational REMP	--)
Potable Water	pCi/l (H-3)	240 - 410	140 - 450	130 - 420	150 - 290	120 - 540	70 - 22[illegible]
Game	pCi/Kg(Cs-137)	76.7 - [illegible]270	35.1 - 6490	54 - 3230	992 - 4330	641 - 5340	34.0 - 63[illegible]
Direct (gamma)	mrem Monthly	2.3 - 5.2	2.8 - 6.9	1.9 - 5.7	3.0 - 6.2	2.7 - 6.9	2.3 - [illegible].7
Radiation	Quarterly	2.7 - 4.8	2.9 - 5.0	2.9 - 4.9	2.8 - 5.5	3.1 - 6.2	2.8 - [illegible]
Air:Gross Beta	[x1.0E-3]	5.0 - 44.0	4.0 - 32.0	5.0 - 360	6 - 47	4.2 - [illegible]1.	5 - 54
Particulate Sr-90	pCi/m3 x 1.E-3	LT 0.8	LT 0.8	0.11 - 0.27	LT 0.8	LT 0.07	1.3 - 1.4
Iodine-131	pCi/m3 x 1.E-3	LT 10.0	LT 10.0	35 - 1230**	LT 10.0	LT 10.0	LT 30.0
Aquatic	pCi/Kg (Sr-90)	LT 1.0	LT 1.0	LT 1.0	6.8 - 27.	* 33.	LT 20.0
Plants	pCi/Kg (Cs-[illegible]7)	LT 6.0	* 85.5	* 47.9	* 45.	69.7 - 140.	36 - 55
	pCi/l (Sr-90)	0.76 - 6.00	0.61 - 5.70	[illegible].98 - 13.0	0.86 - 4.60	0.69 - 5.3	0.9 - 7.[illegible]
Milk	pCi/l (Cs-137)	6.00 - 14.8	5.90 - 11.5	7.0 - 8.9	* 4.4	9.6 - 14	12.9 - 14.1
	pCi/l (I-131)	LT 0.20	LT 0.20	2.1 - 4.8	LT 0.20	LT 0.20	[illegible]A
Food	pCi/Kg (I-131)	[illegible].0	LT 4.0	LT 4.0	LT 4.0	LT 3.0	NA
Products	(wet) (Cs-137)	LT 5.0	LT 5.0	* 12.2	LT 5.0	LT 5.0	* 24.7

* Ranges are not given since only one data point contained an identified isotope.

** Evidence of Chernobyl accident.

TABLE 11.6.1.-4 (Cont'd)

Comparison Of Operational - Preoperational REMP Data

SAMPLE TYPE	Unit/Isotope	(------------- Operational REMP			--------------------)	(-- Preoperational REMP --)	
		1988	1987	1986	1985	1984	1983
Aquatic	pCi/Kg (Sr-90)	LT 1.0	LT 1.0	* 5.6	LT 1.0	LT 0.9	* 86
Invertebrate	(wet) (Cs-137)	LT 5.0	34.8 - 36.2	NA	NA	NA	NA
Beach	pCi/Kg (Sr-90)	LT 1.0	LT 1.0	LT 1.0	LT 1.0	* 3.3	LT 2.0
Sediment	(dry) (Cs-137)	LT 8.0	LT 8.0	LT 8.0	LT 8.0	LT 9.0	NA
Aquatic	pCi/Kg (Sr-90)	LT 2.0	LT 2.0	LT 2.0	LT 2.0	* 1.7	LT 3.0
Sediment	(dry) (Cs-137)	LT 10.0	* 21.7	LT 10.0	* 30.4	44.2 - 49.4	NA
Surface Water	pCi/l (H-3)	* 190	170 - 430	180 - 280	180 - 220	50 - 270	90 - 280
Fish	pCi/Kg (Sr-90)	LT 0.5	LT 0.5	LT 0.5	LT 0.5	LT 0.6	LT 0.7
	pCi/Kg (Cs-137)	7.11 - 17.5	11.0 - 25.8	10.2 - 13.8	7.70 - 17.4	8.4 - 21.4	8.8 - 19.1
Rain Water	pCi/l (H-3)	130 - 490	130 - 410	120 - 190	140 - 320	80 - 970	90 - 270
	pCi/l (Cs-137)	NA	NA	1.40 - 12.4	NA	NA	NA
Noble Gases	pCi/m3 (Kr-85)	28 - 44	24 - 45	21 - 48	24 - 40	30	18 - 49
	pCi/m3 (Xe-133)	LT 11.0	LT 11.0	LT 11.0	LT 11.0	LT 34.0	LT 40.0

* Ranges are not given since only one data point contained an identified isotope.

SHOREHAM DSAR TABLE 11.6.3-1

Second Step Post-Operational Radiological Environmental Monitoring Program (REMP)

Media	Sampling Locations	Sampling Frequency	Analysis
Direct Radiation (1)	1S1,2A2,3S1,4S1,5S2, 6S2,7A2,8A3,9S1,10A1, 11A1,12A1,13S3,14S2, 15S1,16S2,*5E2,*6E1	Quarterly	Gamma Exposure
Fish and Invertebrates (2)	3C1, 14C1, *13G2	Semi-annually or when in season	Gamma-isotopic
Fruit and Vegetables (3)	8B1, 6B21, *12H1	At time of Annual Harvest	Gamma-isotopic
Airborne Particulates (4)	6S2,2A2,3S1,7B1,*11G1	Quarterly	Gross-Beta and Gamma-isotopic
Milk (5)	[illegible]B1,*10F1, or *8G2	Mthly during Grazing Season, Qrtly. at all other times.	Gamma-isotopic
Surface Water	3C1 or 14C1, and *13G2	Semiannual Grab Sample	Gamma-isotopic H-3

(*) Designates Control Locations

(1) Eighteen monitoring stations, DR1 through DR18, (16 indicator and 2 control) are used. One indicator location is positioned in each meteorological sector near the site boundary. One dosimeter or continuously measuring dose rate instrument is placed at each location.

(2) At each Indicator location, one sample of each commercially and recreationally important species. One sample of same species in control location.

(3) Sample three different kinds of broad leafy vegetables grown nearest to two indicator locations -- having highest predicted average ground level D/Q (when milk samples not available). Also take one sample of same leafy vegetation grown nearest to Control Location.

(4) Three samples (near SNPS), one from each of the three Meteorological sectors having the largest annually averaged ground-level D/Q, are taken. One sample (near a community) also having the highest calculated annually averaged ground-level D/Q is taken. Establish one Control Location.

(5) Indicator samples from milking animals having highest potential dose. Sample within 5 km distance (preferably), within 5 to 8 km where doses are calculated to exceed 1 mrem/yr (second choice) or from 8 to 17 km. Control location is 15 to 30 km from SNPS and in the least prevalent wind direction.

SHOREHAM DSAR TABLE 11.6.3-2
REMP SAMPLING LOCATIONS

DESIGNATION	LOCATION
1S1	Beach east of intake, 0.3 mile [N]
2A2	West end of Creek Road, 0.2 mile [NNE]
3C1	Fish and Invertebrates, Outfall Area, 2.9 miles [NE]
3S1	Site Boundary, 0.1 mile [NE]
4S1	Site Boundary, 0.1 mile [ENE]
*5E2	Calverton, 4.5 miles [E]
5S2	Site Boundary, 0.1 mile [E]
6B21	Condezella's Farm Stand, 1.8 miles [ESE]
*6E1	LILCO ROW, 4.8 miles [ESE]
6S2	Site Boundary, 0.1 mile [ESE]
7A2	North Country Road, 0.7 mile [SE]
7B1	Overhill Road, 1.4 miles [SE]
8A3	North Country Road, 0.6 mile [SSE]
8B1	Local Farm, 1.2 miles [SSE]
*8G2	Dairy (Cow), 10.8 miles [SSE]
9S1	Service Road SNPS, 0.2 mile [S]
10A1	North Country Road, 0.3 mile [SSW]
*10F1	Goat Farm, 9.2 miles [SSW]
11A1	Site Boundary, 0.3 mile [SW]
*11G1	MacArthur Substation, 16.6 miles [SW]
12A1	Meteorological Tower, 0.9 mile [WSW]
*12H1	Background Farm, 26 miles [WSW]
13B1	Goat Farm, 1.9 miles [W]
*13G2	Fish and Invertebrates, Background, 13.2 miles [W]
13S3	Site Boundary, 0.2 mile [W]
14C1	Fish and Invertebrates, Outfall Area, 2.1 miles [WNW]
14S2	St. Joseph's Villa, 0.4 miles [WNW]
15S1	Beach west of intake, 0.3 mile [NW]
16S2	Site Boundary, 0.3 mile [NNW]

* Designates Control Locations

Reactor Building

There is no significant source of airborne activity assumed to exist in the reactor building in the plant's present defueled condition.

Turbine Building

There is no source of airborne activity assumed to exist in the turbine building.

Radwaste Building

There is no significant source of airborne activity assumed to exist in the radwaste building.

Further discussion regarding airborne activity is provided in sections 11.1 and 12.4.

REFERENCES

General

Updated Safety Analysis Report (USAR) Shoreham Nuclear Power Station Revision 1, December 1987.

1. ORIGEN2, Isotope Generation and Depletion Code, ORNL CCC-371, 7/80.
2. LILCO calculation C-RPD-476, rev. 0, 10/21/88.
3. LILCO calculation C-RPD-530, rev. 0, 05/19/89.
4. LILCO calculation C-RPD-529, rev. 0, 06/07/89.
5. QADMOD-G, Point Kernel Shielding Code, ORNL CCC-396, 12/79.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged as it is used to develop the basic design criteria of the plant. Refer to the USAR for information on this subject. However, the defueled condition, with low activity levels, some design features are not necessarily utilized as described in the USAR. For example, liquid filters in the radwaste system do not usually require portable shielding or remote backwashing. Also, the radiation zone designations shown on USAR Figures 12.3.1-1 through -35 are not applicable for the plant's present condition.

12.3.2 Shielding

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged as it is used to develop the basic design criteria of the plant. Refer to the USAR for information on this subject.

12.3.3 Ventilation

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to the USAR for information on this subject.

12.3.4 Radiation Monitoring Instrumentation

In order to support the storage of the fuel in the fuel pool, SNPS will need process and effluent radiation monitoring instrumentation, and area and airborne radiation monitoring instrumentation.

Process and Effluent Radiation Monitoring System

The process and effluent radiation monitoring system is designed in accordance with General Design Criterion 64. All normal paths for release of radioactive materials are monitored to ensure compliance with the requirements of 10CFR20, 10CFR50, and Regulatory Guide 1.21.

Table 12.3.4A lists the monitors in service, and Table 12.3.4B provides data for each monitor.

Normally, nonradioactive systems that may become significantly contaminated by leaks from radioactive systems are monitored continually to ensure that no condition hazardous to the operating personnel or to the general public develops. For effluent streams that discharge to the environs, sample points are located downstream of the last point of possible radioactive fluid addition to the effluent being monitored.

All monitors in the process and effluent radiation monitoring system detect gross activity levels and readout and alarm in the main control room. Alarms in the main control room are by annunciators and cathode ray tube (CRT) display.

There are three normal effluent release points from the station that require radiation monitors: the station ventilation exhaust, the liquid radwaste effluent, and the reactor building salt water drain tank.

Area Radiation and Airborne Radioactivity Monitoring Instrumentation

This section contains a description of the area and airborne radiation monitoring systems. All channels have local readout by means of a log-ratemeter and local audible and visual alarms. Each channel has high radiation and fail alarms which are annunciated locally and in the main control room. The area monitors are provided with an audio and visual alert and high radiation alarms. Monitors are placed in areas where personnel normally have access and where there is a possibly that radiation levels could become significant.

All airborne monitors are offline monitors and are designed in accordance with ANSI N 13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities." Sample lines are kept as short as possible to minimize plate out while allowing the monitor to be located in an accessible area.

Airborne radiation monitoring is provided where potentially radioactive sources exists. Each of these monitors is provided with an isokinetic nozzle which is sized to obtain a representative air sample at the normal flow in the ventilation duct from which the sample is taken.

Table 12.3.4B lists the airborne monitors, and Table 12.3.4C lists the area monitors.

Radiation Monitoring System Computers

The RMS is equipped with redundant computers powered from U.P.S. 1Z97-INV-005/TSC Black Battery/69 kV primary feed. These units provide continual surveillance for all airborne, area, process, and effluent radiation monitors. Communication with the computer is through keyboard equipped CRT displays in the main control room, the health physics office, the process computer room, and the technical support center.

Inservice Inspection, Calibration, and Maintenance

The operability of each channel of the area and airborne RMS is checked periodically from the main control room or manually at the monitor. Both systems are checked periodically or as specified by the plant technical specifications.

Calibration of all monitors is normally conducted at an interval of 18 months unless mandated sooner by Technical Specification. This calibration will allow indication in a low, mid, and high response range of each monitor.

12.4 DOSE ASSESSMENT

12.4.1 Design Objectives

The design of the shielding was originally based on conservative estimates of the occupancy time required in each area of the plant, under operating conditions. An effort has been made to keep the dose to plant personnel as low as is reasonably achievable (ALARA) under all conditions, including the defueled condition. Table 12.4-1 lists the six zone designations that were originally established, along with the maximum allowable dose rates and estimated occupancy times for each area. With the plant in its present condition, with spent fuel stored underwater in the pool, there are no occupiable areas which are Zone III or higher.

12.4.2 Airborne Activity

An area within the Shoreham facility is described as an "airborne radioactivity area" if the sum of the concentrations of all airborne radionuclides divided by their respective Maximum Permissible Concentrations (MPCs) (from 10CFR20, Appendix B, Table 1, Column 1) exceeds 0.25. At Shoreham, there are no "airborne radioactivity areas" in the defueled condition. With the fuel in the spent fuel pool, and insignificant quantities of radioactive material elsewhere (see Sections 11.1 and 12.2), it is not expected that airborne radioactivity areas will exist in the future, unless systems which are currently anticipated to remain closed are opened to the atmosphere. In this instance, the radiation work permit procedure (see Section 12.5) will be applied to assure there is no release of contamination into the air.

With exposures reasonably expected to be much less than 2 MPCa-hrs per day and/or 10 MPCa-hrs per week, paragraph 103(a)(3) of 10CFR20 indicates that exposure, and the resulting internal doses, need not be included in the dose assessment to individuals. With no "airborne radioactivity areas" postulated, doses are thus taken to be <u>essentially zero</u> for the defueled condition.

It should be noted that the above conclusion will be confirmed in actual practice by the whole body counting program (see Section 12.5). Procedures are in place for taking appropriate actions, including investigation, when any positive whole body count occurs in excess of 1% of the maximum permissible organ burden (MPOB), or 1% of the maximum permissible body burden (MPBB).

12.4.3 Occupational Dose Assessment

Occupational dose at Shoreham is expected to be essentially zero for the defueled condition. This conclusion has three bases:

1) At present, the dose rates in occupiable areas are virtually all less than 0.5 mrem/hr, as described in Section 12.3. There are no sources of radiation present which would cause the present dose rates to increase to any significant extent.

2) In the defueled condition, occupancy in measurable dose rate areas is expected to be less than or equal to that in the recent past at Shoreham. However, physical decontamination activities (beginning in 1991) will result in occupancy levels substantially exceeding the levels between 1988 and 1990.

3) The recent collective station dose history at Shoreham is as follows (TLD data collected in response to the requirements of 10CFR20.407):

Time Period	Dose, man-rem
1/1/86 - 6/30/86	0.562
7/1/86 - 12/31/86	3.123
1/1/87 - 6/30/87	0.341
7/1/87 - 12/31/87	0.065
1/1/88 - 6/30/88	0.050
7/1/88 - 12/31/88	0.000
1/1/89 - 6/30/89	0.020
7/1/89 - 12/31/89	0.075

Since February of 1987, when a change was made from R. S. Landauer to Panasonic TLDs, doses have been insignificant, and due almost entirely to small statistical fluctuations rather than actual doses.

Based on the above statements, it is anticipated that occupational dose at Shoreham will be essentially zero until physical decontamination activities commence. As such, the 1991 corporate ALARA goal was established at 2.5 man-rem. Doses will be measured as indicated in the Health Physics Program, Section 12.5.

12.4.4 Offsite Dose Assessment

There are no sources (eg, N-16) in the defueled condition which, under normal (non-accident) conditions, could lead to offsite direct doses, either by direct radiation or "skyshine", based on the source terms presented in Sections 11.1 and 12.2. As such, offsite doses to the population are projected to be zero in the defueled condition. This conclusion will be confirmed by the REMP, as described in Section 11.6.

12.5 HEALTH PHYSICS PROGRAM

The Shoreham Health Physics Program, the intent of which is to provide for the protection of all permanent and temporary personnel and all visitors from radiation and radioactive materials in a manner consistent with Federal and State regulations during all phases of operation, is described in Section 12.5 of the USAR. The program is applicable in its entirety to the defueled condition at Shoreham, with the following exceptions:

A) Handling of new fuel is no longer applicable to Shoreham.

(Reference USAR Section 12.5.1.2, Personnel Experience and Qualifications. The basis of this change is that with the Settlement Agreement with New York State, no new fuel will be brought onsite.)

B) The laundry facility does not contain an automated respirator washer, unloading table for same, or a respirator dryer. Cleaning of respirators is done by hand methods when necessary. Respirator fitting may at some time in the future be moved from the Annex Building to another onsite location. Protective clothing is to be cleaned either onsite or offsite, as conditions warrant.

(Reference USAR Section 12.5.2.1, Location of Equipment, Instrumentation and Facilities. The basis of this change is the fact that with no airborne areas currently identified, and none expected in the defueled condition, requirements for respirator use are infrequent. Also, the need to clean protective clothing is significantly reduced.)

C) Deleted

D) The numbers of detectors and monitoring instruments will not necessarily be maintained as indicated in USAR Section 12.5.2.2. Rather, the number maintained will be as required by the defueled plant's activities and number of personnel.

(Reference USAR Section 12.5.2.2, Types of Detectors and Monitoring Instruments. Justification of this change is due to the near total decay of radioiodines at the site.

E) Radiation Work Permits are required for work under any of the following conditions:

1. Work in a posted radiation area.

2. Entry into a posted high radiation area.

3. Work in a posted contaminated area (see Item F below).

4. Entry into airborne radioactivity areas.

5. Breach of a radioactively contaminated system boundary.

(Reference USAR Section 12.5.3.2, Radiation Work Permits. The basis of this change is a change to station procedures.

F) Under the discussion of access control, add the definition of a contaminated area:

Contaminated Area

Any area having removable beta/gamma-emitting radioactive material in excess of 1000 dpm/100 sq cm, or alpha-emitting radioactive material in excess of 20 dpm/100 sq cm.

(Reference USAR Section 12.5.3.3.1, Access Control. The basis for this change is a modification to the station health physics procedures, as recommended by the Institute of Nuclear Power Operations, in their document INPO 85-001, rev.1.)

G) Under the discussion of access control, the "secondary access facility" no longer exists.

(Reference USAR Section 12.5.3.3.1, Access Control. The basis for this change is that as of September 1, 1989, the secondary access facility was taken out of service.)

H) The ALARA Review Committee (ARC) now administratively reports to the Resident Manager.

(Reference USAR Section 12.5.3.3.4, Post-Operations Review. The basis for this change is an organizational change. See Chapter 13 of the DSAR for further details.)

I) As stated in DSAR Section 12.1D, there is no longer a need to provide dosimetry to personnel entering the RCA, unless they are required by an RWP.

(Reference USAR Section 12.5.3.5, Health Physics Training Program.)

It should be noted that some of the procedural requirements or commitments indicated under the USAR Health Physics Program will not apply in the defueled condition. For example, no areas requiring reevaluation for extra shielding are anticipated, due to the low current source terms (Reference USAR Section 12.5.3.3). However, potential sources of radioactivity (during physical decontamination activities) warrant that the procedures and commitments remain in place.

CHAPTER 13

CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE AND RESPONSIBILITIES

The description contained under this heading in the latest revision of the Shoreham USAR is superseded in its entirety by the following.

The Long Island Power Authority (LIPA), a non profit public entity, as the sole owner of the Shoreham Nuclear Power Station (SNPS) has assumed full responsibility for its maintenance and decommissioning. LIPA has contracted the New York Power Authority (NYPA) to manage the day-to-day maintenance and decommissioning of the SNPS. NYPA, also a non-profit public entity, is the sole owner and operator of the Indian Point 3 (IP3) and James A. Fitzpatrick (JAF) Nuclear Power Plants. LIPA and NYPA have entered into an agreement whereby several key corporate and site upper management personnel are coemployees of both LIPA and NYPA. This arrangement allows LIPA to establish and maintain technical cognizance through onsite residency. Coemployed personnel shall have management responsibility for the safe conduct of operations at the Shoreham site as a whole, as well as individual management responsibilities in the areas of operations and maintenance, decommissioning, radiological controls, quality assurance, and licensing/regulatory compliance. This coemployment status shall be maintained in order to provide assurance of a safe and efficient maintenance and decommissioning process in conformance with Nuclear Regulatory Commission (NRC) requirements and the facility licensing commitments. Additional NYPA resources are available to LIPA to support SNPS maintenance and decommissioning on a non-coemployed basis.

LIPA has also entered into separate agreements with the Long Island Lighting Company (LILCO) to secure the support of selected incumbent LILCO technical, administrative and management staff personnel, as well as offsite support services in areas such as training, emergency preparedness, environmental engineering, and other areas. LILCO is an investor-owned public utility and was responsible for the original design, construction and licensing of SNPS.

Figure 13.1-1 depicts the corporate and plant organization of LIPA for the maintenance and decommissioning of the SNPS.

13.1.1 Corporate Organization

Figure 13.1-1 depicts the LIPA Corporate Organization for the management of the SNPS.

13.1.1.1 Corporate Organizational Arrangement

The LIPA Chairman has overall responsibility for the administration of LIPA, including all financial, legal, and public relations aspects. The Chairman is appointed by the Governor of the State of New York.

In meeting and supporting these responsibilities, the LIPA Chairman has a President of Shoreham Project reporting directly to him on matters relating to operations, engineering, decommissioning, quality assurance, and security. The President of Shoreham Project shall have an Executive Vice President of Shoreham Project (EVPSP) reporting directly to him.

The EVPSP shall be a corporate project intermediary addressing administrative, budgetary, engineering, quality assurance, security, and decommissioning activities at the SNPS. The EVPSP shall provide overall guidance and direction to the Shoreham Decommissioning Project, shall be the corporate executive responsible for the overall nuclear safety of the plant and shall have the authority to take such measures as may be needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. These operations are discharged by the Shoreham Plant Resident Manager and the managers of the Operations and Maintenance, Decommissioning, Nuclear Operations Support, Nuclear Quality Assurance, Finance and Administration, and Licensing/Regulatory Compliance departments. Supplementary technical support is provided to these organizations under the direction of LILCO executive management by various offsite LILCO departments and divisions through appropriately defined LIPA Nuclear Operations Corporate Policies.

The EVPSP shall be a coemployee of both LIPA and NYPA. As a minimum, the EVPSP shall have a Bachelor's degree in science or an engineering field associated with power production. The EVPSP shall also have 10 years of experience associated with plant design and operation, at least 5 years of which shall be nuclear power plant experience.

The Manager of the NQA Department has overall responsibility for nuclear quality assurance (QA) activities directing the activities of the Quality Control (QC) Manager and Quality Systems (QS) Manager. The Manager, NQA reports to the EVPSP for policy matters, and to the Resident Manager for personnel administration, budgetary control, and functional day to day assignments. The Manager, NQA has direct access to the EVPSP and to the President of Shoreham Project for nuclear safety matters, as he deems necessary.

13.1.1.2 Technical Support

Technical support for the activities associated with the maintenance and decommissioning of Shoreham is provided b, several organizations. Onsite technical support is provided by the Nuclear Engineering Division of the Operations and Maintenance Department, and by the Decommissioning Engineering Division of the Decommissioning Department. Refer to sections 13.1.2.1.1 and 13.1.2.1.3, respectively, for a description of the technical support functions of these organizations.

Additional offsite technical support is also available as needed through LILCO and NYPA corporate engineering organizations, as well as through qualified outside contractors.

13.1.2 Operating Organization

The SNPS organization chart is shown in Figure 13.1-1. This chart depicts the titles and line of authority of the plant personnel in charge of the various plant departments. The station organization shall include all the technically trained personnel necessary to support all aspects of the maintenance and decommissioning of the plant.

13.1.2.1 Station Organization

The Resident Manager, reporting to the EVPSP, has complete responsibility for the safe, efficient, and dependable maintenance and decommissioning of the plant. The Resident Manager administers an organization of LIPA management employees skilled in the various disciplines required for nuclear plant maintenance and decommissioning. Management employees in turn direct the actions and supervise the performance of station personnel at the plant, which are a composite of NYPA, LILCO and contractor employees under LIPA's umbrella organization. All plant personnel shown in Figure 13.1-1 shall meet or exceed the minimum qualifications of ANSI N18.1 1971 for comparable positions, as appropriate for the permanently defueled status of SNPS.

13.1.2.1.1 Operations & Maintenance Department

The Operations & Maintenance Department (O&M) provide onsite technical and administrative support for operations, maintenance, radiological controls, instrumentation and controls, and engineering services.

The O&M department is organized along the following division lines:

Operations Division

The Operations Division (OD) shall be responsible for the operation and monitoring of station systems and equipment required for daily operations and maintenance, and for complying with the station license and regulations of governing agencies. The OD will assist the Decommissioning Department, as necessary, and shall review decommissioning activities related to operations. The OD shall provide station administrative support and assurance that the station is in compliance with the requirements of the License.

The OD shall be responsible for field engineering and providing the criteria for post-modification/return to service testing of station modifications. The OD shall prepare and issue procedures and coordinate the implementation, testing and startup activities associated with station modifications. The OD shall assure that the plant and systems modifications are properly installed, tested and demonstrated functional.

The OD will implement their portion of the station surveillance programs.

The OD will interface with the Decommissioning Department and other support organizations, as appropriate. This includes planning and scheduling support associated with plant O&M activities.

During off-shifts and in the absence of the Resident Manager or his designee, the OMD in the person of the Watch Engineer shall be the person-in-charge of station activities.

The OD will direct the Maintenance Division in fuel handling operations.

Maintenance Division

The Maintenance Division (MD) shall be responsible for maintaining the station's systems and equipment (i.e. - mechanical, electrical, instrumentation, computer, etc.) and for implementation of approved modifications. This responsibility includes administering the local area computer network and providing general computer support. The MD shall have a staff experienced in mechanical and electrical maintenance of large steam-electric generating stations. As necessary, the division's staff may be supplemented with competent maintenance personnel from outside contractors.

The MD shall be responsible for ensuring that plant systems, applicable to the maintenance of SNPS are maintained in accordance with the station license requirements and provide the Decommissioning Department with maintenance support throughout the planning and execution of SNPS decommissioning.

In addition, the MD shall be responsible for ensuring the calibration, maintenance, and testing of all instruments and control systems during daily maintenance and decommissioning activities except for those instruments which are calibrated and maintained by the Radiological Controls Division. The MD will repair, test, and maintain all hardware, software, and firmware associated with security, and radiological monitoring systems and selected portions of the process radiation monitoring system.

The MD shall provide refueling bridge operators to work under the direction of the Operations Division during fuel handling operations.

Radiological Controls Division

The Radiological Controls Division (RCD) shall be responsible for establishing programs and procedures for protecting the public, station personnel, and the environment from the effects of radiation associated with normal maintenance of the plant and associated decommissioning activities. It shall provide mechanisms for ensuring radiation doses of station personnel and the public are maintained as low as reasonably achievable (ALARA) and assure proper handling, processing, and disposal of radioactive materials in compliance with applicable regulatory requirements.

The RCD shall be responsible for activities such as analysis and monitoring of potentially radioactive effluents to the environment associated with maintenance and decommissioning of the plant. The RCD shall work with NED in assessing radiation doses to the public, and shall be responsible for station chemical and radiochemical activites.

The RCD shall be responsible for assisting the Emergency Director in evaluating an emergency condition. Continuing assessment actions will be taken for the purposes of identification and characterization of the incident, prediction of offsite doses, if any, resulting from the incident, notification to offsite authorities, determination of appropriate corrective measures, and determination of escalation, reduction, or termination of the emergency.

The RCD shall be responsible for maintaining an effective waste reduction program and assuring regulatory compliance, in handling, packaging, storing, and shipping of all radioactive waste generated during daily maintenance and decommissioning activities at SNPS. The RCD shall maintain an adequate inventory of protective clothing and contamination control equipment, to support daily maintenance and decommissioning activities; and shall maintain the plant as radiologically clean as possible through implementation of non-specialized decontamination processes.

The RCD shall interface with all departments and with offsite agencies and parties regarding procedures, techniques, and resources necessary to adequately maintain preparedness for any radiological emergency condition which could arise with the plant in the defueled condition or as a result of decommissioning activities.

The RCD will prepare Radiation Work Permits, perform radiological surveillances, maintain personnel exposure records, calibrate and maintain all fixed and portable radiation detection instrumentation, and dispose of radioactive material properly.

The RCD shall be responsible for providing environmental support to the SNPS organizations, as necessary.

The RCD shall be responsible for developing conducting and evaluating the final site radiation survey.

Nuclear Engineering Division

The Nuclear Engineering Division (NED) shall be responsible for providing design and engineering expertise in station systems, structures, and equipment. This includes identifying problems and recommending corrective action or design changes, development of plant improvements, monitoring of plant modification implementation and performance, and issuance of approved engineering procedures and specifications for use, as required by plant procedures.

The NED shall provide the design for station modifications.

The NED shall perform, manage, direct, and provide design verification for engineering, design and safety analyses and perform/review engineering and safety evaluations.

The NED shall perform project engineering for engineering studies and plant modifications, and prepare and monitor engineering schedules and cost estimates for all engineering work related to the maintenance of SNPS and shall provide engineering and technical support to the Decommissioning Department. The NED shall provide administrative and technical direction to outside engineering consultants and LILCO's Office of Engineering that are performing activities related to chartered responsibilities.

The NED shall provide engineering support for technical review of spare and replacement parts, procurement and dedication of commercial grade items to nuclear application as required.

The NED will prepare and review licensing correspondence and submittals with regard to impact on design and safety analyses and licensing documents. The NED shall review and evaluate the technical adequacy of design, licensing and operating aspects of new or proposed regulatory requirements and industry experience.

The NED shall also prepare and assure accuracy of the content of plant technical basis, design control and licensing documents (i.e., drawings, specifications, calculations, procurements, technical manuals, equiment and other controlled lists, and program descriptions) as required.

The NED shall be responsible for technical interface with the Decommissioning Department engineering personnel to assure that decommissioning engineering plans, activities and station modifications are compatible with the existing Shoreham plant design.

The NED shall be responsible for developing and maintaining the Radiological Environmental Monitoring, Process Control, Offsite Dose Calculation Manual, LIPA/LILCO Corporate Fire Protection Program (Nuclear) and ALARA programs.

13.1.2.1.2 Nuclear Operations Support Department

The Nuclear Operations Support Department (NOSD) provides LIPA with security, fire protection and safety, emergency preparedness coordination, plant administration, records management, document control, and training support for the planning and execution of SNPS decommissioning.

The NOSD is organized along the following division lines:

Nuclear Security & Training (NST) Division

NST shall consist of the Nuclear Security, Emergency Preparedness, and Nuclear Training Sections.

Nuclear Security Section

The Nuclear Security (NS) section shall be responsible for establishing and implementing security plans, procedures, contingency plans, guard training and qualification plans and programs necessary to comply with the rules and regulations of the governing regulatory agencies. NS ensures that strict security is established and maintained to keep the station buildings, equipment, materials, and personnel safe from injury, unauthorized used, or destruction. NS will review appropriate plant modifications and decommissioning activities to ensure compliance with current and projected security requirements and commitments.

NS will ensure security intrusion detection and access control systems meet the requirements of the security plan. These systems protect the facility against sabotage or attack and provide and enforce a system of prevention of theft or loss of LIPA property through a protection of assets program. In addition, the NS will maintain security records concerning systems, equipment, and personnel.

The NS will maintain liaison with local, state, and federal law enforcement agencies to assure coordination and support of the security plan.

Nuclear Training Section

Nuclear Training (NT) shall be responsible for implementing and coordinating training required to establish qualifications for personnel assigned at SNPS. Individual Departments will establish position descriptions and qualification requirements within the Training and Qualification (T&Q) Program. Training to meet these requirements will be conducted, coordinated, and recorded by the Nuclear Training Coordinator (NTC).

Training will be established through contracts, the LILCO Training Center, or directly conducted through the NTC. Review and approval of qualifications will be the responsibility of the appropriate department. T&Q records will be developed, tracked within the LILCO T&Q Program and Controlled as QA records.

The NTC shall be responsible for the development of training policy and procedures; and for the development, implementation, and evaluation of training programs for permanent, temporary, and contractor personnel. The NTC will ensure that training programs meet regulatory compliance, radiation dose minimization, worker safety, cost effectiveness, and plant staff qualification requirements.

The NTC shall coordinate personnel training and prepare appropriate training materials for the following subjects and for other specialized training, as necessary: General Employee Training; Fitness for Duty; ALARA; security; fire protection and safety; emergency preparedness; decommissioning; quality assurance indoctrination; etc.

Although the applicable training will be directed through the NTC, the individual SNPS departments shall have the responsibility for ensuring that personnel under their direction are qualified to assume the responsibilities of their positions.

The NTC shall make recommendations to the Resident Manager on training policies and procedures which impact on other departments or on maintenance and decommissioning activities of the plant.

Emergency Preparedness Section

The Emergency Preparedness Coordinator (EPC) shall be responsible for reporting on the status of the development and implementation of the emergency plan to protect the public and station personnel from the effects of radiation exposure in the event of a radiological emergency during daily maintenance activities at the plant and during the planning and execution of SNPS decommissioning.

The EPC is the primary interface with LILCO's Office of Corporate Services for matters pertaining to the development, maintenance, and revision of LIPA's Emergency Plan.

Fire Safety & Administration (FSA) Division

The FSA shall consist of the Fire Protection & Site Safety, Plant Administration, and Records Management and Document Control Sections.

Fire Protection & Site Safety Section

The Fire Protection & Site Safety Section (FPSS) shall be responsible for implementing the fire protection program, including drills, surveillance activities, and maintenance of fire equipment; and for providing effective health and safety programs for the employees, contractors, and visitors to SNPS, in compliance with local, state, and federal laws regarding health, industrial safety, and fire protection. The FPSS shall review and audit the installation and maintenance of fire protection and prevention equipment throughout decommissioning activities and shall maintain fire protection and safety records and files.

The FPSS shall manage an on-site medical/first-aid facility to provide competent emergency care and first-aid to minimize medical complications from injury/illness. The FPSS shall develop medical unit policies and procedures. The FPSS shall analyze plant first-aid and medical equipment needs and establish a network of available emergency equipment to optimize emergency response in conventional and radiation areas. The FPSS shall maintain records of all health and safety related documentation.

The FPSS shall maintain information regarding quantities and types of hazardous materials stored and used on-site; and develop appropriate strategies and resources for protecting station personnel from unacceptable exposures to hazardous materials. The FPSS will coordinate hazardous material information with the Radiological Engineeing Section of the Radiological Controls Division of the O&M Department, as appropriate, in accordance with applicable regulatory requirements.

The FPSS shall be responsible for development and review of all Fire Protection, Hazardous Material, and Safety training material including the training of a fire brigade and Hazardous Material Response Team. The FPSS shall maintain a liaison between local, state, and federal agencies to ensure efficient response in any emergency. The FPSS shall be responsible for operation and maintenance of intoxilyzer equipment as required to support the Fitness for Duty Program.

Plant Administrative Section (O&M)

The Plant Administrative Section - O&M is supervised by the Plant Administrative Coordinator (PAC) and shall be responsible for providing direction to the office organization, including plant personnel records, plant filing system, office procedures, miscellaneous office equipment and supplies, and reproduction equipment for the Operations & Maintenance Department. The PAC administers the flow of correspondence, specifications and drawings into and out of the plant. The PAC maintains, updates, and distributes plant procedures.

The PAC shall be responsible for the supervision of the secretarial, clerical, and other administrative office personnel required for the Operations & Maintenance Department.

The PAC will interface with the Operations & Maintenance Department, as appropriate.

Records Management and Document Control Section

The Records Management and Document Control Section (RMDC) shall be responsible for administration support and control for procedures, records management, and document control. RMDC shall establish, implement, and maintain the SNPS Records Management and Document Control Programs consistent with applicable requirements.

13.1.2.1.3 <u>Decommissioning Department</u>

The Decommissioning Department (DD) provides LIPA with engineering, construction, and special process support for the planning and execution of SNPS decommissioning.

The Decommissioning Department is organized into the following divisions:

° Decommissioning Engineering Division

° Construction Division

° Special Processes Division

Decommissioning Engineering Division

The Decommissioning Engineering Division (DED) shall be responsible for providing engineering support for the implementation of the SNPS Decommissioning Plan. This includes development of engineering packages and safety evaluations required to accomplish the decommissioning of SNPS.

The Decommissioning Engineering Division shall be responsible for the day-to-day engineering support of decommissioning activities, direction and performance of the principal Architect/Engineer, and resolution of technical issues related to decommissioning. The Decommissioning Engineering Division shall review all decommissioning activities performed by the Decommissioning Department and assure that the activities are in compliance with the requirements of project specifications and the License.

The Decommissioning Engineering Division shall coordinate and interface with other station departments and divisions as necessary.

The Decommissioning Engineering Division shall be responsible for cost and schedule control of all activities under its cognizance.

Construction Division

The Construction Division (CD) shall be responsible for the implementation and performance of dismantlement and construction support activities necessary for the decommissioning of the station's systems, equipment and structures in accordance with project specifications, station policies and procedures, applicable regulatory criteria, and the Decommissioning Plan. The activities conducted by this division shall encompass those dismantlement techniques and construction support activities that are considered standard industry techniques requiring little or no plant-specific development, demonstration or qualification prior to use.

The CD shall be the focal point for the acquisition and direction of decommissioning craft labor and shall be responsible for the performance of the decommissioning General Contractor and any other construction subcontractors not under the direction of the General Contractor. The CD shall also assist with plant maintenance as needed and requested.

The CD shall coordinate and interface with other station departments and divisions as necessary.

The CD shall be responsible for cost and schedule control of all activities under its cognizance.

The CD staff may be supplemented with competent construction personnel from outside contractors as necessary.

Special Processes Division

The Special Processes (SP) Division shall be responsible for the specification, selection, implementation, and performance of specialized decontamination and dismantlement methods and processes to be used during plant decommissioning activities, including project management of the disposition of the spent fuel.

Special methods and processes are those which require some level of plant specific development, qualification, and/or demonstration prior to use at Shoreham. (They should not be confused with the term special processes used in conjunction with quality assurance processes.) Examples of special processes include wire rope and underwater plasma arc cutting to be used on the reactor pressure vessel, and ultra high pressure water or chemical decontamination.

The SP Division shall be responsible for the acquisition, direction and performance of the specialty contractors selected to develop and implement the required decommissioning special processes.

The SP Division shall be responsible for ensuring that plant systems, components, and structures on which special processes are performed are decommissioned in accordance with project specifications, station policies and procedures, applicable regulatory criteria and the Decommissioning Plan.

The SP Division shall be responsible for project management of the implementation of the option selected for disposition of the spent fuel. This includes responsibility to ensure that the option selected is implemented by the appropriate specialty contractor(s) in a safe and efficient manner in accordance with applicable regulatory criteria and the Decommissioning Plan, within schedule and budget. Further, the SP Division is responsible to ensure all spent fuel disposition activities are properly integrated with other maintenance and decommissioning activities.

The SP Division shall be responsible to coordinate and interface with other station departments and divisions as necessary.

The SP Division shall be responsible for cost and schedule control of all activities under its cognizance.

13.1.2.1.4 Licensing/Regulatory Compliance Department

The Licensing/Regulatory Compliance Department (LRCD) provides LIPA with guidance regarding regulatory, licensing, nuclear safety, and environmental compliance, licensing commitment identification and tracking, and nuclear regulatory and licensing information support for the planning and execution of SNPS decommissioning.

The LRCD monitors the status of all regulatory, licensing, safety, and environmental compliance activities, licensing commitment status, and generic and plant-specific information regarding developments in nuclear regulation and licensing.

The LRCD shall perform the following:

- Provide the justification for LIPA's exemptions, exceptions, or proposed alternatives for compliance with NRC regulatory criteria.

- Prepare amendments to SNPS Defueled Technical Specifications and licensing basis documents, and related justifications, as required.

- Advise LIPA personnel regarding the proper scope and content of safety evaluations required under 10CFR50.59.

- Prepare No Significant Hazards Consideration evaluations for submittal to the NRC for Defueled Technical Specifications changes and other license amendments.

- Review and concur with and/or prepare changes to SNPS Defueled Safety Analysis Report (DSAR) pursuant with LIPA policy, and bring the DSAR up-to-date in accordance with the requirements of 10CFR50.71(e), as required.

- Maintain overall schedule for meeting regulatory requirements and commitments and, as appropriate, obtain schedule extensions from the NRC for completion of decommissioning commitments.

- Develop licensing strategies for maintenance and decommissioning activities.

- Review and concur with criteria selected for plant and decommissioning modifications and activities.

- Review surveillance documentation and maintain the Master Surveillance Schedule.

- Maintain required licenses and permits and coordinate renewals as necessary.

- Coordinate LIPA interface with nuclear industry organizations such as the Nuclear Utility Management and Resources Council (NUMARC) and other industry forums.

- Interface with agencies of the State of New York for matters pertaining to compliance with state regulations.

- Provide support in determinations for events potentially requiring NRC notification under 10CFR50.9 and 10CFR50.72 and prepare the Licensee Event Reports required under 10CFR50.73.

- Perform the LRCD functions specified in NOC Policy 24, "Corporate Evaluation and Reporting Responsibilities Pursuant to 10 CFR Part 21."
- Provide support for and coordinate activities of the Site Review Committee (SRC) and Independent Review Panel (IRP).

The LRCD shall be responsible for the receipt of incoming nuclear licensing correspondence and regulatory documents (both plant-specific and generic) and for reviewing such information to determine if a corporate position or response is required. The LRCD shall distribute such information to appropriate site and corporate organizations, and shall establish a strategy and schedule for the development of input for any required corporate positions or responses. The LRCD shall be responsible for assigning input development responsibilities or other required actions, and for the coordination of input development, corporate review and comment resolution. The LRCD shall review draft input for responsiveness, compliance with regulations and consistency with corporate policy, and shall assemble a final document package for signature by the Resident Manager or Executive Vice President-Shoreham Project, as appropriate.

The LRCD shall be responsible for the overall management, staffing, coordination, strategy and conduct of Atomic Safety and Licensing Board (ASLB) litigation, as well as any other litigation pertaining to nuclear licensing or safety issues. The LRCD shall be the primary interface with LIPA's legal counsel, and shall work closely with other LIPA, NYPA and LILCO organizations to assign technical resources, select witnesses and develop the LIPA strategy for a given issue.

The LRCD shall interface with the licensing organizations of other utilities; and coordinate licensing positions with other utilities, particularly those who have decommissioned or are decommissioning a nuclear power plant. The LRCD shall be responsible for recommending appropriate licensing actions in concert with other utilities upon obtaining concurrences from appropriate LIPA personnel.

13.1.2.1.5 <u>Nuclear Quality Assurance Department</u>

The Nuclear Quality Assurance (NQA) Department provides LIPA with quality assurance and control related licensing commitments, a Quality Assurance (QA) Program, and other LIPA administrative policy support for the planning and execution of SNPS decommissioning.

NQA is responsible for establishing and maintaining a quality assurance program, documented by written policies, procedures or instructions, that meets the requirements of Appendix B to 10 CFR 50 and other NRC requirements for the SNPS. This program sets forth the requirements for quality related activities performed by the various departments at the plant and all applicable LIPA contractors.

Refer to Chapter 17, Quality Assurance, for a detailed explanation of the Nuclear Quality Assurance Department.

13.1.2.1.6 Finance & Administration Department

Onsite support to all the technical departments of the SNPS are supplied by the Finance & Administration (FA) Department.

This includes providing administrative support to the plant departments in budgetary, accounting, procurement, and material control matters during operations and throughout decommissioning of the plant.

The FA Department provides LIPA with administrative, budgetary, and procurement and material control support for the planning and execution of SNPS decommissioning.

FA is organized into the following divisions:

° Accounting

° Procurement/Contract Administration

° Project Controls

° Materials Management

Accounting Division

The Accounting Division (AD) is responsible for tracking the historical cost of maintenance and decommissioning.

The AD will receive, track, and facilitate payment of vendor invoices.

It will maintain and reconcile thorough, timely, and accurate accounting records as required by the Site Cooperation and Reimbursement Agreement and Management Services Agreement dated January 24, 1990. AD will also interface with NYPA, LIPA and LILCO accounting personnel and coordinate implementation of accounting for costs attributable to Shoreham in addition to support of various audits of Shoreham records.

The AD is responsible for management of the accounting-related modules contained in the Power Authority Reporting and Information System (PARIS). This includes reconcilation of data in Shoreham PARIS and the interfaces between Shoreham PARIS and other systems.

Procurement/Contract Administration

The Procurement/Contract Administration Division (PD) is responsible for procurement of the materials, equipment, and services required to maintain and decommission Shoreham. It is also responsible for administering contracts governing such procurements.

For purposes of this description, Procurement refers to all of the actions which are required to obtain goods (equipment and materials) and services (professional, consulting, technical, subscriptions, etc.) from outside sources. Contract Administration refers to the actions that are required to obtain labor-burdened contracts from outside sources, the management of the commercial elements of performance within each contract, and the closing of each contract with reconciliation of contract values among the User Group, Accounting, Project Controls, and the Contractor.

The PD will:

- ensure that schedule needs and commitments are reflected in each contract/purchase order.
- ensure that work scope is clearly incorporated in each contract/purchase order.
- document contractor performance.
- minimize claims and risk of loss.
- ensure contract records are preserved.
- negotiate scope changes to achieve the most favorable commercial conditions.
- implement a back-charge program for recovering monies in the event of vendor non-performance.

The PD will receive and process approved purchase requests. As required, it will solicit and conduct the evaluation process of bids for purchases, and award purchase orders and contracts to the preferable bidder. It will also maintain the files of record for each purchase order and contract issued, from purchase request through the life of the P.O. and Contract.

The PD will comply with all established procurement regulations and LIPA's applicable Quality Assurance Program requirements in its purchasing activities. The PD will maintain relations with vendors consistent with corporate ethical standards.

Project Controls

The Project Controls Division (PC) is responsible for three functions:

- Planning and Scheduling
- Cost Estimating
- Budgeting and Cost Control

The Planning and Scheduling function will develop and maintain strategic schedules for maintaining and decommissioning Shoreham. Strategic schedules are milestone-level schedules required by internal managers to properly perform their functions. The PC will communicate schedules to other groups and solicit their input for updating these schedules.

The Cost Estimating function will develop and maintain the total cost estimate for maintaining and decommissioning Shoreham. It will also develop a system for maintaining, tracking and reporting the total cost estimate and other estimates required during maintenance and decommissioning. This group will also revise the estimate as required by the Management Services Agreement (MSA).

The Budgeting and Cost Control function will develop and maintain budgets as required by the MSA and by Shoreham managers to control the project. It will develop and maintain systems for tracking and reporting commitments and charges against these budgets, and for forecasting maintenance and decommissioning charges yet to be incurred.

Budgeting and Cost Control will assist in validating that services invoiced by vendors were performed and that costs incurred are representative of work performed. It will develop and analyze financial data and report these analyses to the appropriate managers. This group will also ensure that the appropriate accounting codes are assigned to purchase requests, commitments and charges.

Materials Management

The Materials Management Division (MD) is administratively responsible for receiving, storing, controlling, and issuing material and equipment to be used in maintenance and decommissioning. It will control the on-site warehouse and interact with LILCO to reserve and obtain existing inventory to use in maintenance and decommissioning.

The MD will communicate with the Operations & Maintenance and Decommissioning Departments to plan materials needs and to schedule materials purchases to coincide with scheduled work activities. It will also assist the Quality Assurance Department in ensuring safety-related materials meet or exceed LIPA's requirements and are properly controlled.

The MD will maintain appropriate records of and documentation for LIPA-purchased and/or controlled materials. It will also review the adequacy of and compliance with LILCO's shelf life and inventory preventive maintenance programs.

13.1.2.2 Plant Personnel Responsibility and Authority

The functions, responsibilities and authorities of key station personnel are delineated in the position descriptions contained in the LIPA Shorheam Nuclear Power Station Administrative Manual. The qualifications for the positions described therein meet the requirements of ANSI N18.1-1971, as appropriate for the permanently defueled status of SNPS, for comparable positions.

13.2 TRAINING PROGRAM

The description under this heading in the Shoreham USAR is superseded by the following.

13.2.1 Training To Support Maintenance In The Defueled Condition

In the defueled state, with the NRC operating license amended to remove operating authority, there is no requirement to maintain accredited training programs since the plant is no longer licensed to operate.

The LILCO Office of Training has non-nuclear training programs available to LIPA, developed via a "systematic approach to training" method, which can be requested by the Shoreham plant management for training of operators, technicians, and mechanics.

The Office of Training procedures outline the methods to be used to analyze training needs, and to establish or conduct required training. The Office of Training staff will be qualified in accordance with the LILCO "Training and Qualification Program".

Operators: Operators will be trained (or have been trained previously under LILCO ownership of the SNPS license) in the function and operation of those systems required to be operational during the defueled phase. The material used to conduct this training will be from the operator training program developed for nuclear operations.

This operator training program was originally developed by LILCO in order to license reactor operators and senior reactor operators in accordance with 10CFR55 for low power and then full power plant operation. Following issuance of the Shoreham Possession Only License (POL), however, LILCO received an exemption from 10CFR55 allowing the licensed operator re-qualification programs to be reduced commensurate with the more limited scope of activities authorized by the Shoreham POL. When the POL was transferred to LIPA, in turn, LIPA received permission to eliminate altogether the need for operators to be licensed under 10CFR55, as well as permission to adapt the reduced LILCO licensed operator requalification program for use as a LIPA certified operator requalification program. This approval also included permission for operators who 10CFR55 license qualifications under LILCO had not expired initially certified by LIPA for the remainder of the qualification term without a new examination.

Equipment Operator: Field operators will be trained (or have been trained) using portions of the Equipment Operator Training Program developed for nuclear operations. This training will include generic, non-nuclear theory, and the function and operation of those systems required to be operational during the defueled phase.

Control Technicians: Control technicians and computer technicians will be trained (or have been trained) in accordance with the Control Technician training program developed for power plant technicians.

Mechanics/Electricians: Mechanics/electricians attend formal training as part of LILCO's maintenance training programs. These programs qualify mechanics/electricians as apprentices with journeyman qualifications available in the area of welding, rigging, machinery, electrical, and general maintenance skills. The Shoreham maintenance force will be trained and qualified in accordance with existing LILCO maintenance training programs. This program is not available for contract maintenance work forces; contractors would provide qualified mechanics and electricians.

Rad Chem/Health Physics: The Radiochemistry and Health Physics technicians will be trained (or have been trained) using the training material developed for Health Physics and Rad Chem technicians for nuclear operation. However, the training will be limited to fundamentals and task specific training as required to support Rad Chem, Health Physics, and Radwaste operations during the defueled condition.

13.2.2 Training To Support Decommissioning Activities

The Training Program for decommissioning is described in Section 2.4 of the LIPA Shoreham Nuclear Power Station Decommissioning Plan.

13.3 EMERGENCY PLANNING

The description under this heading in the Shoreham USAR is superseded by the following.

The Emergency Plan for the Shoreham Nuclear Power Station is submitted as a corporate document titled "LIPA Defueled Emergency

Preparedness Plan," which was adapted from the prior LILCO Defueled Emergency Preparedness Plan. Changes from the LILCO document were limited to those necessary to reflect LIPA as the Shoreham licensee.

13.4 REVIEW AND AUDIT

The following information supersedes the information under this heading in the Shoreham USAR with respect to review and audit of activities conducted under the POL by LIPA.

A review and audit program, including in-plant and independent reviews, have been developed to: provide a system to ensure that plant design, operation, and decommissioning are consistent with company policies and rules, approved procedures, and license provisions; review important proposed plant decommissioning changes, tests, experiments, and procedures; assure that unusual events are promptly investigated and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer.

Review and audit during operating and decommissioning of the Shoreham Nuclear Power Station (SNPS) is an integral part of the Long Island Power Authority (LIPA) Quality Assurance Program. Provisions are established for a comprehensive system of planned and periodic audits to verify implementation of Quality Assurance Program requirements. These review and audit functions are fully described in Chapter 17, Quality Assurance, of this Defueled Safety Analysis Report (DSAR). In addition, LIPA utilizes a formal committee method for review and audit cognizance, functioning at two levels:

1. At the station operation level, the Site Review Committee (SRC)
2. At the corporate level, the Independent Review Panel (IRP), which is independent of direct responsibility for plant maintenance and decommissioning.

The review and audit program has been established to assure that the operation and decommissioning of the plant is in conformance with established procedures, license provisions, and quality assurance requirements and to review and approve changes to station systems/equipment and procedures as described in the DSAR or tests and experiments, which do not constitute an unreviewed nuclear safety question, as defined in 10 CFR, Part 50.59. All unreviewed safety questions and changes to the Technical Specifications are reviewed by the IRP as described below.

Reviews of nuclear safety related questions are made by the IRP as described below.

A continuing review is performed by the SRC to monitor plant maintenance, and plan future decommissioning activities, and to screen subjects that might be of interest to the IRP.

13.4.1 Site Review Committee

The SRC shall function to review plant operations and advise the Resident Manager on all matters related to nuclear safety, radiological and/or environmental protection, and decommissioning activities.

WRITTEN CHARTER

A written charter has been prepared covering such areas as group responsibility, subjects requiring review, reporting requirements, and organization.

The charter of the SRC reflects the consideration that committee review responsibilities extend to all station activities and proposed changes or modifications to station systems or equipment and are not limited to those designated safety related.

COMPOSITION

The SRC shall be composed of a chairman or alternate chairman and six or more members or alternate members of the plant staff as designated by the chairman.

ALTERNATES

All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in SRC activities at any one time.

MEETING FREQUENCY

The SRC shall meet at least once per calendar month and as convened by the SRC Chairman or his designated alternate.

QUORUM

The quorum of the SRC necessary for the performance of the SRC responsibility and authority provisions under the Defueled Technical Specifications shall consist of the Chairman or his designated alternate and four other members including alternates.

RESPONSIBILITIES

The SRC shall be responsible for:

a. Review of (1) all proposed procedures and programs required by Defueled Technical Specification 6.7 and changes thereto, and (2) any other proposed procedures or changes thereto as determined by the Resident Manager to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to the Possession Only License and Defueled Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Defueled Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the Chairman of the Independent Review Panel (IRP) and the Executive Vice President-Shoreham Project;

f. Review of all reportable events;

g. Review of decommissioning activities and facility operations to detect potential nuclear safety hazards;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Resident Manager, any member of the SRC, or the Chairman of the IRP;

i. Review of the Security Plan and implementing procedures;

j. Review of the Defueled Emergency Preparedness Plan and implementing procedures;

k. Review of the Fire Protection Plan and implementing procedures;

l. Review of the proposed changes to the Process Control Program (PCP);

m. Review of the proposed changes to the Offsite Dose Calculation Manual (ODCM);

n. Review of the proposed major changes to radioactive waste systems;

o. Review of Personnel Radiation Records annually to determine how exposures might be lowered consistent with ALARA principles. Document such considerations; and

p. Review of any accidental, unplanned, or uncontrolled onsite release of radioactive material, including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Executive Vice President-Shoreham Project and to the IRP.

q. Quality review of ALARA Review Committee (ARC) activities.

r. Review of proposed changes to the approved Decommissioning Plan.

The SRC shall:

a. Recommend in writing to the Resident Manager approval or disapproval of items considered under items a through e and l through n above prior to their implementation.

b. Render determinations in writing with regard to whether or not each item considered under items a through e and l through n above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours, to the Executive Vice President-Shoreham Project of disagreement between the SRC and the Resident Manager; however, the Resident Manager shall have responsibility for resolution of such disagreements pursuant to Shoreham Defueled Technical Specifications.

d. Function to advise the Resident Manager on all matters related to nuclear safety, radiological environmental operations, and decommissioning activities.

RECORDS

The SRC shall maintain written minutes of each SRC meeting that, at a minimum, document the results of all SRC activities performed under the "Responsibilities" Section of Defueled Technical Specification 6.5. Copies shall be provided to the Chairman of the IRP and the Executive Vice President-Shoreham Project.

13.4.2 Independent Review Panel (IRP)

FUNCTION

The IRP shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear engineering,
b. Chemistry and radiochemistry,
c. Radiological safety,
d. Mechanical and electrical engineering, and
e. Quality assurance practices.

The IRP shall report to and advise the Executive Vice President of Shoreham Project.

WRITTEN CHARTER

A written charter has been prepared covering such areas as group responsibility, subjects requiring review, reporting requirements, and organization.

The charter of the IRP reflects the consideration that IRP activities are not limited to items and functions that are designated as safety related. It is intended that IRP review and audit activities will also cover non-safety related structures, systems, components, and plant computer software to ensure that the safety significance given to them in the DSAR, the Technical Specifications, and the Emergency Operating Procedures will be maintained during the operation of Shoreham.

COMPOSITION

The IRP shall be composed of the permanent IRP Chairman and a minimum of four permanent IRP members. The chairman and all members of the IRP shall have qualifications that meet the requirements of Section 4.7 of ANSI/ANS 3.1-1978.

The membership shall include at least one individual from outside LIPA's or its contractors' organizations and at least one individual with substantial nuclear experience. The nuclear experience may be provided by the individual who is from outside LIPA's or its contractors' organizations.

MEETING FREQUENCY

The IRP shall meet at least once per six months.

QUORUM

The quorum of the IRP necessary for the performance of the IRP review functions of the Technical Specifications shall consist of the Chairman or his designated alternate and at least three but not less than one-half of the IRP members present including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

The IRP shall review:

a. The safety evaluations for (1) changes to equipment or systems and (2) tests or experiments completed under the provisions of 10CFR50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10CFR50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10CFR50.59;

d. Proposed changes to Technical Specifications of this Possession Only License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant deviations from normal and expected performance of station equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the SRC.

Audits of station activities shall be performed under the cognizance of the IRP. These audits and audit frequencies are as follows:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training and qualifications of the entire staff at least once per 12 mo hs;

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety, at least once per year;

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures, equipment and program implementation at least once per 24 months utilizing either a qualified offsite licensee fire protection engineer(s) or an outside independent fire protection consultant;

f. Any other area of station operation considered appropriate by the IRP, the President of Shoreham Project or the Executive Vice President of Shoreham Project;

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months; and

i. The Process Control Program and implementing procedures for solidification of radioactive wastes at least once per 24 months.

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

RECORDS

Records of IRP activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each IRP meeting shall be prepared, approved, and forwarded to the President of Shoreham Project and the Executive Vice President of Shoreham Project within 14 days following each meeting.

b. Reports of reviews encompassed by Technical Specification 6.5.2.7 shall be prepared, approved, and forwarded to the President of Shoreham Project and the Executive Vice President of Shoreham Project within 14 days following completion of the review.

c. Audit reports encompassed by Technical Specification 6.5.2.8 shall be forwarded to the President of Shoreham Project, Executive Vice President of Shoreham Project and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

13.5 STATION PROCEDURES

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged except for the following:

13.5.1 Administrative Control

1. Safety-related station procedures shall be processed through the Site Review Committee (SRC) and Nuclear Quality Assurance (NQA).
2. The Resident Manager shall approve Station Administrative Procedures, Security Plan Implementating Procedures, and Emergency Plan Implementing Procedures prior to implementation.
3. Other Station Operating Procedures shall be approved by the appropriate Division Manager or by the Operations and Maintenance Department Manager prior to implementation.
4. Revised Table 13.5.1-1 is attached.

13.5.1.1 Normal Operations

The NRB has been replaced by the IRP in accordance with 13.4.2 and a new Table 13.5.1-1 is supplied herein.

13.5.1.2 Routine Maintenance, Repairs, and Fuel Handling

LIPA QA personnel shall be responsible for the auditing of procurement documents to ensure that appropriate quality control requirements are fulfilled as defined in Section 17.2.

13.5.1.3 Modifications

The LIPA Site Review Committee is responsible for review of proposed modifications to safety related systems or components.

13.5.2 Procedures

Changes to subsections of USAR section 13.5.2 as a result of the permanently defueled plant configuration are identified below. Other information remains as described in the USAR.

13.5.2.1 Operating Procedures

1. The General Operating Procedures now only describe integrated station operation. Startup and Shutdown are no longer pertinent.

2. Operating Procedures are not necessarily performed by, or under the direction of, persons holding RO or SRO licenses.

13.5.2.1 Initial Test Procedures

This section is no longer pertinent.

13.5.2.2 Shoreham Nuclear Power Station Emergency Preparedness Plan

The "Radiological Emergency Preparedness Plan" has been replaced with a "Defueled Emergency Preparedness Plan" as indicated in section 13.3 of the DSAR.

13.5.2.3 Temporary Procedures

Temporary procedures for refueling are no longer required at Shoreham.

13.6 PLANT RECORDS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged except that the Manager, Operations and Maintenance Department or his designee shall be responsible for the compilation of operating records and event records as set forth in the Station Administration Procedures.

13.7 INDUSTRIAL SECURITY

The Security Plan, Training and Qualification Plan, and the Safeguards Contingency Plan for the Shoreham Nuclear Power Station have been submitted as separate documents. These documents are withheld from public disclosure pursuant to 10CFR2.79(d), "Rules of Practice." The Security Plan and the Safeguards Contingency Plan are also withheld from public disclosure pursuant to 10CFR73.21, "Requirements for the Protection of Safeguards Information."

TABLE 13.5.1-1

PROCEDURES PROVIDED FOR SHOREHAM NUCLEAR POWER STATION

A. Administrative Procedures shall be provided to cover the following types of administrative activites:

1. Authorities and Responsibilities for Safe Fuel Handling Operations
2. Equipment Control (e.g., locking and tagging)
3. Procedure Adherence and Temporary Change Method
4. Procedure Review and Approval
5. Schedule for Surveillance Tests
6. Shift and Relief Turnover - Recall of Personnel
7. Log Entries and Record Retention
8. Bypass of Safety Functions and Jumper Control
9. Operating Orders
10. Special Orders
11. Materials Control
12. Radiation Work Permits
13. Access Control to Controlled Area
14. Personnel Training and Qualification

B. Operating Procedures

1. General Operating Procedures have been provided to cover the following Integrated Plant Operating Activities:

   a. Surveillance.

2. System Operating Procedures shall describe Startup, Normal Operating, and Shutdown for the designated system. Abnormal Operation, where required, shall be contained in a section of the System Operating Procedure. Procedures are available for operating the systems listed in a through ac. below.

   a. 138kV and 69kV Power System
   b. Normal Station Service Transformer
   c. Reserve Station Service Transformer
   d. Well Water System
   e. 4,160 V System
   f. 480 V System
   g. Station Lighting Panels
   h. 120 V ac Instrument Bus
   i. 120 V ac Reactor Protection System Bus
   j. 120 V ac Uninterruptible Power Supply
   k. 125 V dc System

TABLE 13.5.1-1 (Cont'd)

B. Operating Procedures (Cont'd.)

l. Reactor Building Normal Ventilation System (RBNVS)
m. Service Water
n. Radwaste (Liquid)
o. Radwaste (Solid)
p. Communications System
q. Condensate Transfer
r. Deluge and Sprinkler System
s. Demineralized Water Transfer
t. Equipment and Floor Drains
u. Fire Protection System
v. HVAC - Control Room
w. HVAC - Turbine Building
x. HVAC - Radwaste Building
y. Makeup Water Treatment
z. Station Air System
aa. Smoke, Temperature, and Flame Detection System
ab. Turbine Building Closed Loop Cooling System
ac. CRAC Chilled Water

3. Emergency Procedures have been provided for combatting the following potential emergency conditions:

a. Acts of Nature
b. Abnormal Releases of Radioactivity
c. Fuel Handling Accident
d. Plant Fires
e. Loss of Electrical Power
f. Loss of Service Water

4. Abnormal Operation Procedures required to mitigate the consequences of the following abnormal conditions shall be contained in the appropriate System Operating Procedures(s):

a. None.

TABLE 13.5.1-1 (Cont'd)

Note: Procedures not designated as emergency procedures shall be incorporated in the Abnormal Performance section of the appropriate system or general operating procedures.

C. Alarm Response Procedures (ARP)

Alarm Response Procedures shall be provided as required for alarm windows in the main control room associated with the operation of safety related systems or equipment.

D. Fuel Handling Procedures shall be provided to cover the following fuel handling activities:

1. Special Nuclear Materials Control and Accountability Procedures
2. Spent Fuel Handling and Shipment
3. Handling and Storage of Sealed and Unsealed Sources

E. Health Physics Procedures shall be provided to cover the following radiation protection activities:

1. Dose Rate Radiation Surveys
2. Surface Radioactive Contamination Surveys
3. Personnel Contamination Survey
4. Personnel Decontamination
5. Areas and Equipment Decontamination
6. Monitoring for and Collecting and Recording of Occupational Radiation Exposure (ORE) data
7. Submission and Review of Suggestions by Plant Personnel for the Reduction of ORE
8. Use of Protective Clothing and Respiratory Equipment

F. Defueled Emergency Preparedness Implementing Procedures (DEPIPs) shall be provided to cover the following emergency plan activities:

1. Emergency Classification
2. Evacuation and Personnel Accountability
3. Operational Assessment and Damage Estimates
4. Support Systems and Activation
5. Surveys, Analyses, Sampling, Assessment, and Actions
6. Personnel and Equipment Decontamination
7. Notifications
8. Re-entry and Recovery
9. Emergency Organization, Drills, and Training

TABLE 13.5.1-3

FORMAT FOR STATION PROCEDURES

*SP Number ______________________

Revision ____ Eff. Date _________

	Signature	Date	TPC No.	Date Eff	Date Expr
Section Head	________________	___________	______	_______	________
Quality Control	______________	___________	______	_______	________
Div. Mgr.	________________	___________	______	_______	________
Resident Mgr.	________________ Signature or N/A	___________	______	_______	________

TITLE

1.0 PURPOSE

A brief description of the purpose for which the procedure is intended should be clearly stated. If the procedure is used to satisfy, in any part, a Technical Specification surveillance requirement, indicate the Technical Specification number here.

2.0 RESPONSIBILITY

Indicate the person directly responsible for ensuring the proper implementation of the procedure.

3.0 DISCUSSION

Provide a brief description of the applicable component, system, or task in sufficient detail for a knowledgeable individual to perform the required function without direct supervision. Include a list of topics or a table of contents generally describing the extent or scope of the procedure, with page location.

* For temporary procedures, SP Number assignment is TP YYXYYY.YY.

TABLE 13.5.1-3 (Cont'd)

EVENT ORIENTED EMERGENCY PROCEDURE FORMAT

Submitted: ______________________ SP Number ______________________
(Section Head)

Approved: ______________________ Revision ______________________
(Operations Manager)

Effective Date ________________

TITLE*

*Should be worded to indicate the purpose of the procedure.

1. SYMPTOMS: Symptoms should be included to aid in the identification of the emergency. This should include alarms, operating conditions, and probable magnitudes of parameter changes. If a condition is peculiar only to the emergency under consideration, it should be listed first.

2. AUTOMATIC ACTION: (Delete if not pertinent)

3. IMMEDIATE ACTION: These steps should specify immediate action for operation of controls or confirmation of automatic actions that are required to stop the degradation of conditions and to mitigate the consequences of degraded conditions.

4. SUBSEQUENT ACTION: Steps should be included to return the reactor to a normal shutdown period under abnormal or emergency conditions.

5. FINAL CONDITIONS: These steps should specify the documentation, authorizations, and plant conditions that must be completed prior to resumption of Normal Operation, defined in 22XYYY.YY.

6. DISCUSSION: A brief explanation of the procedure.

This section should contain background information, causes, effects, and other information that may assist in clarifying the procedure and analyzing symptoms.

Note: Attempt to get 1, 2 and 3 on cover page of procedure to allow rapid evaluation and action by the operator.

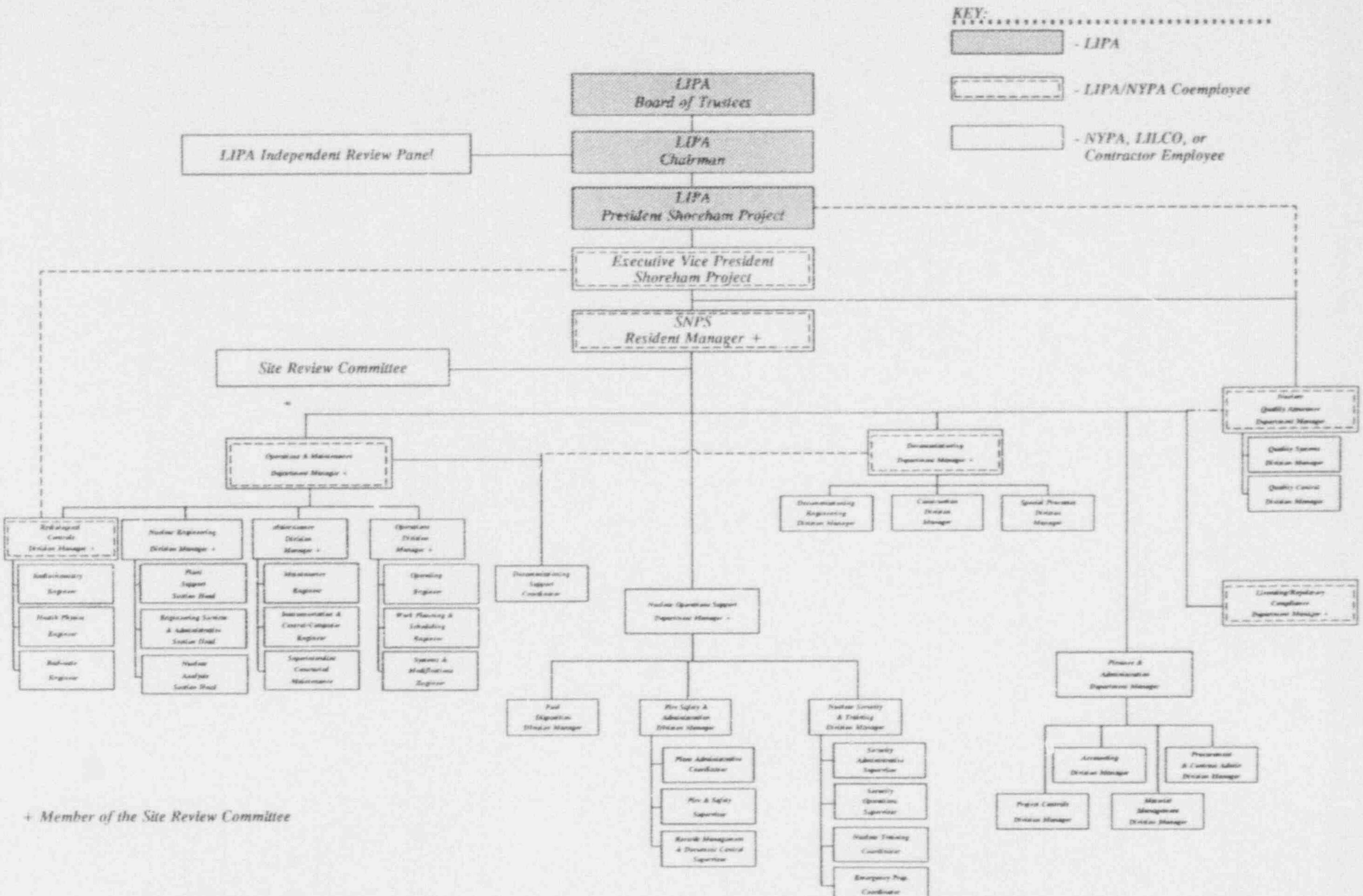

Figure 13.1-1
LIPA Corporate and Plant Organization
Defueled Safety Analysis Report
Revision 4 July 1992

CHAPTER 15

ACCIDENT ANALYSIS

15.1 GENERAL

Analytical Objective

Chapter 15 of SNPS USAR provides the results of analyses of the spectrum of transient and accident events which are postulated to occur with the plant operating initially at up to maximum power. The purpose of this analysis is to identify USAR transients and accidents that apply to the storage and handling of the low burnup fuel.

The analysis is based on the defueled condition of the plant, i.e., the fuel is removed from the core and is stored in the spent fuel pool. The total decay heat is approximately 550 watts, which is small enough that it could be removed by passive cooling and would not require the fuel pool cooling system. Normal and emergency makeups are discussed in Chapter 9.

As the reactor will not be operated and the fuel is not in the reactor, most of the USAR Chapter 15 events cannot occur.

Approach to Safety Analysis

The safety parameter evaluated for each transient of USAR Chapter 15 is the Minimum Critical Power Ratio (MCPR) which is a measure of fuel cladding integrity. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) is the safety parameter for the reactor LOCA-related accidents, and indicates whether the peak cladding temperature and the zirconium-water reaction is below the specified limits. As the decay power level is extremely low during spent fuel storage, and will not increase, MCPR and MAPLHGR limits cannot be exceeded and are not applicable.

Those transients and accidents of USAR Chapter 15 which pose the potential for a radiological release outside the primary containment are of primary concern.

Heat Generation Analysis

One result from the ORIGEN2 calculation is a graph of decay heat or thermal power (in watts), as a function of time. Results of this analysis are presented in Figure 15.1-1 The calculated decay heat load as of June 1989 is approximately 0.55 kw.

It must be recognized that there are some limitations in the ORIGEN2 model, and potential inaccuracies in the calculational processes of the code and its supporting data sets. For instance, ORIGEN2 is a "point reactor" model, and cannot deal conveniently with the spatial variations in fuel enrichment and burnup. In addition, there are uncertainties associated with averaging of nuclear cross-section data within the thermal, resonance, and fission neutron energy ranges. Nevertheless, it is not expected that large uncertainties should occur in heat load estimates. See the comparison of calculated to measures dose rates in DSAR Section 12.2. This gives evidence that the decay heat load calculations are reasonable, as the same analysis (ORIGEN2) was used to generate both sets of data.

Analytical Categories

Each USAR Chapter 15 event is assigned to one of six analytical categories. The analytical categories and the events in each analytical category are discussed below.

1. Decrease in Core Coolant Temperature

   This analytical category of USAR Chapter 15 events includes the following events:

   15.1.5 Pressure Regulator Failure - Open

   15.1.7 Feedwater Controller Failure - Maximum Demand

   15.1.8 Loss of Feedwater Heating

   15.1.9 Shutdown Cooling (RHR) Malfunction - Decreasing Temperature.

   In the spent fuel storage condition, the pressure regulator, feedwater controller, feedwater heating system and RHR system are not operating and all four transients are, therefore, not applicable.

2. Increase in Reactor Pressure

   Since the generator, turbine, main steam isolation valve, pressure regulator, feedwater system, condenser and RHR systems are not operating in support of nuclear fission, the following transients are not applicable:

   15.1.1 Generator Load Rejection

   15.1.2 Turbine Trip

   15.1.3 Turbine Trip with Failure of Generator Breakers to Open

   15.1.4 Main Steam Isolation Valve Closure

   15.1.6 Pressure Regulator Failure - Closed

   15.1.18 Loss of Feedwater Flow

15.1.21 Loss of Condenser Vacuum

15.1.26 Core Coolant Temperature Increase

The transient of this category applicable to spent fuel storage is the following:

15.1.19 Loss of AC Power

A loss of AC power condition can be postulated that will affect normal support systems. However, because of the very low heat generation rate (see Figure 15.1-1) and large thermal capacity of the pool active fuel pool cooling is not required. Loss of the spent fuel pool water makeup capability will result only in a very slow evaporative loss of the pool water. This evaporation rate is so slow that ample time exists to restore normal pool makeup sources so that pool level can be quickly restored. Thus, the passive protection provided by the spent fuel pool and low fuel decay heat eliminate the need for active makeup requirements. (The rate of evaporation is discussed in Chapter 9.)

The loss of AC power will not in itself result in any release of radioactivity, as fuel movement is disallowed by Tech Specs when AC power is lost (and is virtually impossible in any event), and the decay heat of the core is so low. Should the loss of AC power occur as part of any other event which causes damage to the fuel in the pool, while the release in this case would not be monitored, the offsite dose consequences would be insignificant. Doses and dose rates are bounded by the "puff release" results given in Sections 15.1.36 and 15.1.36A.

3. Decrease in Reactor Coolant Flow Rate

The recirculation pumps and recirculation flow controller are not operating in the defueled condition and therefore all the transients of this category are not applicable:

15.1.20 Recirculation Pump Trip

15.1.22 Recirculation Pump Seizure

15.1.23 Recirculation Flow Control Failure With Decreasing Flow

4. Reactivity and Power Distribution Anomalies

Events included in this category are those which cause rapid increase in power. Since the reactor is defueled, the following events are not applicable:

15.1.11 Continuous Control Rod Withdrawal During Power Range Operation

15.1.12 Continuous Control Rod Withdrawal During Reactor Startup

15.1.13 Control Rod Removal Error During Refueling

15.1.14 Fuel Assembly Insertion Error During Refueling

15.1.15 Off-Design Operational Transient Due to Inadvertent Loading of a Fuel Assembly into an Improper Location

15.1.16 Inadvertent Loading and Operation of a Fuel Assembly in Improper Location

15.1.24 Recirculation Flow Control Failure with Increasing Flow

15.1.25 Abnormal Startup of Idle Recirculation Pump

15.1.33 Control Rod Drop Accident

5. Increase in Reactor Coolant Inventory

Since the HPCI system is not required the following transient is not applicable:

15.1.10 Inadvertent HPCI Pump Start

6. Decrease in Reactor Coolant Inventory

6.A Events Not Applicable to Spent Fuel Storage

The safety relief valve and the feedwater system are not operating in the defueled condition; therefore the following events are not applicable:

15.1.17 Inadvertent Opening of a Safety Relief Valve

15.1.37 Feedwater System Piping Break

The following event is not a design basis event and is applicable only to power operation:

15.1.27 Anticipated Transient Without Scram (ATWS)

The single failure-proof polar crane design eliminates the following event:

15.1.28 Cask Drop Accident

Instrument line, coolant line and steam line breaks present no consequences due to their lack of interaction with the fuel and therefore the following events are not applicable:

15.1.30 Off-Design Operational Transient as a Consequence of Instrument Line Failure

15.1.34 Pipe Breaks Inside the Primary Containment (Loss-of-Coolant Accident)

15.1.35 Pipe Breaks Outside the Primary Containment (Steam Line Break Accident)

6.B Events Without Fuel Damage

15.1.29 Miscellaneous Small Releases Outside Primary Containment

Releases that could result from piping failures outside the primary containment include the pipe breaks in the fuel pool cleanup system. The resulting offsite dose will be negligible and are bounded by the Radwaste Tank Rupture accident.

15.1.29.1 Seismic Event

Because the spent fuel pool structure and fuel racks and handling equipment meet Seismic Category I requirements, a seismic event is not postulated to create a radiological release. However, certain predecommissioning and decommissioning activities may involve the temporary use of QA/Seismic Category II structures, systems and components which could fail during a seismic event, may damage fuel and may create a radiological release. No credible seismically induced accident will exceed the bounding radiological release postulated in Section 15.1.36A. Therefore, the radiological consequences of this very low probability event are bounded by those already analyzed and reported in Section 15.1.36A.

15.1.31 Main Condenser Gas Treatment System Failure

As the main condenser is not operating, there can be no offsite dose resulting from this event.

15.1.32 Liquid Radwaste Tank Rupture

Should accident occur radioactivity could be released to the environment but the effect would be negligible. The accident analysis described in DSAR Section 11.2.3.4.2 and 11.2.3.4.3 proves this.

15.1.36.6.1.3 Radiological Effects

Offsite

Radiological exposures have been evaluated for the meteorological conditions, parameters, and assumptions given in Table 15.1.36-1. The results are given in Table 15.1.36-2.

Control Room

Because the amount of radioactivity released is so small, the control room air intake monitors will not alarm and are not required. The control room HVAC system will continue to function in its normal operating mode. The resultant whole body and skin 30-day integrated doses are, at most, 9.59E-08 and 2.08E-04 mrem, respectively, well below the 10CFR50 GDC 19 limits.

Discussion

It is seen in Table 15.1.36-2 that the (0-2 hour) EAB and (0-30 day) LPZ integrated doses are many orders of magnitude below 10CFR100 guidelines. Results are graphically shown in Figure 15.1.36-1. Furthermore, th maximum (t=0) dose rates (whole body and skin) are very low and, with the exception of the RBNVS case, below Technical Specifications. This indicates that the HVAC system in use in the reactor building has no meaningful effect on radiological consequences to members of the public during a fuel handling accident with the present fuel source terms.

15.1.36A Worst Case Fuel Damage Event

Scenario

Several "worst case", extremely conservative scenarios were examined. Specifically, for the three reactor building HVAC cases analyzed in Section 15.1.36.5 (RBSVS operating, RBNVS operating, and puff release), instead of assuming the gap activity from 125 fuel rods is released (2.52 Ci Kr-85), it is assumed that all gaseous activity from the entire core in the spent fuel pool is released (1.56E+03 Ci Kr-85). This can only occur if all the fuel is postulated to be mechanically damaged and there is a complete release of gaseous isotopes. The assumption of a complete release of the gaseous inventory is also very conservative with respect to the Regulatory Guide 1.25 assumption of a 30% release fraction given the low burnup condition of Shoreham spent fuel. Doses and dose rates are thus a factor of 617 higher than for the corresponding Regulatory Guide 1.25 cases.

All other conditions and parameters indicated in Table 15.1.36-1 apply to these cases. Results are given in Table 15.1.36A-1.

Discussion

Even with the highly conservative release quantity postulated above, the calculated whole body and skin dose at the EAB and LPZ are very small fractions (less than 0.031%) of the 10CFR100 dose guidelines. Results are graphically shown in Figure 15.1.36A-1. Dose rates for the postulated worst case scenaric are above current ODCM limits, but the duration of the high dose rates in the RBNVS and puff release cases is quite short (two hours or less).

CHAPTER 17

QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged. Refer to USAR for information on this subject.

17.2 QUALITY ASSURANCE DURING THE DECOMMISSIONING PHASE

The description of the Quality Assurance Program during Shoreham Nuclear Power Station operational phase under this heading in the latest revision of the Shoreham USAR is revised. However, many of the structures, systems and components designated as Quality Assurance Category I (safety related) in USAR Table 3.2.1-1 have been redesignated as Quality Assurance Category II in this DSAR. The applicability of the USAR Section 17.2 Operational phase Quality Assurance Program as modified in this DSAR to the Quality Assurance (QA) Categories in DSAR Table 3.2-1 are as follows:

QA Category I	-	The USAR Section 17.2 Quality Assurance Program as modified by DSAR Section 17.2 applies to the safety related structures, systems and components which meets the intent of 10CFR50, Appendix B.
QA Category IIA (formerly safety related)	-	Deleted
QA Category II - these (non safety related)		Appropriate measures are applied to structures, systems, and components in accordance with QA corporate policy to assure that the safety significance given to them in the DSAR, Technical Specifications, and Emergency Operating Procedures are maintained.

The specific modifications of the USAR Section 17.2 applicable to the Shoreham decommissioning phase are as follows:

17.2.1 Organizations

The Long Island Power Authority (LIPA) is responsible for the establishment and execution of the Quality Assurance (QA) Program during the Shoreham decommissioning. LIPA has established the organization structure shown on Figure 13.1-1, LIPA Organization for Quality Assurance, to fulfill this responsibility. The organization depicted in Figure 13.1-1 is subject to the QA

Program requirements set forth in the LIPA Quality Assurance Manual. Nuclear Quality Assurance (NQA) Department personnel verify compliance by means of review, audit, surveillance, inspection, testing, or other appropriate methods.

The Executive Vice President-Shoreham Project reports directly to the LIPA President of Shoreham Project and is responsible for the overall direction, radiological and industrial safety, cost and schedule of the project. He is the corporate officer responsible for QA Program implementation and review, protection of occupational and public safety, and coordination with regulatory agencies.

He also has overall responsibility for the engineering, testing, licensing, modification, safety, reliability and maintenance, security and decommissioning of the Shoreham Nuclear Power Station and the implementation of the LIPA QA Program. He delegates the administration of these functions to the Resident Manager who has delegated to the Department Managers (Operations and Maintenance; Decommissioning; Finance and Administration; Nuclear Operations Support and Licensing/Regulatory Compliance), the responsibility to assure compliance with the QA Program requirements in their organizations.

The Shoreham Plant Resident Manager reports directly to the Executive Vice President - Shoreham Project, and has overall responsibility for day-to-day management of all station activities. Through his subordinates, he directs the technical, administrative and regulatory functions to accomplish all of the tasks and activities comprising the LIPA project.

He also has the overall responsibility for the implementation of the LIPA QA Program and maintenance of safety-related structures, systems and components as defined in the DSAR Section 17.2.

The Nuclear Quality Assurance (NQA) Department Manager reports directly to the Executive Vice President and has direct access to the LIPA President of Shoreham Project as he deems necessary. The NQA Department Manager is responsible to the Shoreham Plant Resident Manager for administration of the QA program. This organizational arrangement provides the necessary independence between personnel performing activities subject to the controls of the QA program and those responsible for performing the checks, audits, and inspections. The NQA Department Manager is responsible for directing the activities of the Quality Control (QC) and Quality Systems (QS) Division Managers. His principal objective is to ensure that the Shoreham plant and all support organizations establish and conform to adequate standards and procedures in accordance with the LIPA QA Manual. He has the authority to stop work when circumstances so warrant.

The Manager, NQA Department, is responsible for the development and implementation of the overall QA Program during the decommissioning of the Shoreham Nuclear Power Plant.

The QC Manager and QS Manager report to the Manager, NQA Department. This organizational and functional relationship assures that the LIPA QS personnel who audit or otherwise verify quality related activities are free from undue cost and scheduling influences and are independent of personnel who perform or are responsible for the activities.

The Manager, NQA Department, is responsible for maintaining a working interface and communication with other organizations, regulatory agencies, consultants, contractors, inspection firms, and others as required to effectively execute the policies stipulated in the QA Program. He is responsible for assuring the establishment and continuous implementation of QA indoctrination and training programs for LIPA QA and other involved personnel. The indoctrination and training will cover the quality related policies, procedures, and requirements applicable to the personnel involved. He is responsible for review and approval of applicable documents to assure the inclusion of appropriate quality requirements as indicated in Section 17.2.6. He is responsible for the performance of audits as described in Section 17.2.18. In determining the applicability of the QA Program, the Manager, NQA Department shall consider the safety significance accorded to nonsafety related structures, systems, components, and plant computer software given to them in the Defueled Safety Analysis Report (DSAR), Technical Specifications, and Emergency Operating Procedures.

The Manager, NQA Department, is responsible for defining the content and changes to the LIPA QA Manual subject to review and approval as indicated in Section 17.2.6 and Appendix D of the LIPA QA Manual.

The Manager, NQA Department is authorized to evaluate the manner in which all activities, both at the station and offsite, are conducted with respect to quality by means of reviews, audits, surveillance, and/or inspections. He shall perform this evaluation on a planned and periodic basis to verify that the QA Program is being effectively implemented. He is responsible for periodically evaluating and reporting on the status and adequacy of the QA Program to the appropriate LIPA management. He has the authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions through designated channels, and to verify implementation of solutions. He has the authority to initiate stop work action or to control further processing, delivery, or installation of nonconforming material through appropriate channels as described in the applicable QA procedure.

The Manager, NQA Department is assisted in carrying out his responsibilities by the QC and QS Managers.

NQA is composed of engin rs and technical and nontechnical personnel as needed. Additionally, the NQA staff shall be supplemented when necessary by consultants, contractors, or other organizations within LIPA. Line responsibility, coordination, and communication in such cases shall be through the QC and QS Managers.

The QC and QS Managers are jointly responsible for assuring full implementation of the LIPA QA Program, inc uding additions and changes thereto. Each is responsible within his delegated scope of duties to establish and implement appropriate QA procedures and instructions; review applicable documents as indicated in Section 17.2.6; and perform audits, surveillances, and/or inspections as indicated in Sections 17.2.10 and 17.2.18. Each has, within his scope of responsibilities, the authority and organizational freedom to identify and report quality problems; to initiate, recommend, or provide solutions through designated channels; and to verify implementation of solutions. Each has the authority to initiate stop work action through appropriate channels or to control further processing, delivery, or installation of nonconforming material as described in the applicable QA procedures.

17.2.2 Shoreham Quality Assurance Program

Responsibility for assuring that the Shoreham station will be decommissioned safely rests with LIPA. The LIPA Corporate Statement of Quality Assurance Policy imposes a QA Program designed to meet the requirements of Title 10 of the Code of Federal Regulations, Part 50, Appendix B, and identifies the QA Manual as the document that establishes the requirements for quality affecting activities during the decommissioning phase. The QA Manual, which is distributed on a controlled basis to responsible managers and key supervisory and QA personnel, contains this corporate policy statement.

The QA Program is designed to assure that activities such as design, procurement, fabrication, shop inspection and testing, shipping, storage, construction, erection, cleaning, installation, fuel handling activities, equipment and system operation, maintenance, repair and modification of materials, structures, systems, components, services are accomplished in accordance with the criteria of 10 CFR 50, Appendix B. The QA Program is applied to the safety related structures, systems, and components as appropriate. Nonsafety related structures, systems, components, and services shall be accorded, as a minimum, the safety significance given to them in the DSAR, the Defueled Technical Specifications, and Emergency Operating Procedures. This practice will assure that the safety significance accorded to nonsafety related structures, systems, and components is maintained during the decommissioning of Shoreham. Also, the Shoreham preventive and corrective maintenance program, the design change control program, procedures for procurement of equipment, and procedures for modification and removal of equipment from service shall ensure that LIPA continues to accord to nonsafety related structures, systems, and components the safety significance given to them in the DSAR, Technical Specifications, and Emergency Operating Procedures.

Thus, the responsible personnel implementing these programs and procedures shall, in exercising their judgment on the appropriate measures to be applied to nonsafety related structures, systems, and components, do so in accordance with the corporate QA policy.

The QA Program, described in the LIPA QA Manual, is supplemented by QA Procedures and Instructions, which provide the detailed instructions and checklists necessary to implement or verify implementation of QA Program requirements. These procedures are delineated in Section 17.2.5. QA procedures are issued, reviewed and approved as shown in Table 17.2.6-1. The QA Manual, Procedures, and Instructions shall be controlled in accordance with the requirements of Section 17.2.6.

The QA Program requires that activities affecting quality shall be accomplished in accordance with documented policies, procedures, and instructions throughout the decommissioning of Shoreham. These activities shall be accomplished under suitably controlled conditions. Controlled conditions include, as applicable, appropriate equipment, suitable environmental conditions, and assurance that required prerequisites have been satisfied. Also considered shall be the need for special controls, processes, and requirements for verification of quality by inspections, examinations, or tests.

The QA Procedures for decommissioning are derived from the program requirements established in the QA Manual. Organizations described in Section 17.2.1, performing activities that affect quality, shall prepare their procedures incorporating requirements of the QA Manual and referenced codes, standards, and guides. These procedures shall also receive a QA review to assure that all program requirements have been addressed.

The Corporate Statement of QA Policy, contained in the LIPA QA Manual, imposes the mandatory QA Program requirements on all personnel and organizations performing activities affecting the quality of safety related structures, systems, and components during the decommissioning of Shoreham. The Manager, NQA Department is responsible for periodically engaging an organization independent of the organization being reviewed to assess LIPA quality related activities and to evaluate the scope, implementation, and effectiveness of the QA Program. This periodic review assures that the program is adequate and complies with corporate QA policies, goals, objectives, and 10 CFR 50, Appendix B criteria. The requirement for independent QA Program evaluation is further imposed, as appropriate, on other organizations participating in the LIPA QA Program. The LIPA QA auditing program is described in Section 17.2.18.

The Manager, NQA Department is responsible for establishing and implementing the QA Program. Provisions have been established for the referral of quality related problems to the highest level of management necessary for resolution. The Manager, NQA Department, is responsible for regularly assessing the status and adequacy of the QA Program. He shall, through written reports and or periodic meetings, inform the Executive Vice President - Shoreham Project and Shoreham Plant Resident Manager, of the effectiveness of the QA Program and of significant quality trends.

These regular assessments shall be conducted in accordance with the requirements outlined in Section 17.2.18 and as detailed in Section 18 of the QA Manual. The requirement for regular QA Program evaluation shall be extended to other participating organizations for the portions of the program they are executing.

The QA Program requires that procedures be established for the indoctrination, training and, if appropriate, certification of personnel performing or verifying safety related activities. These procedures shall document the scope, objective, and method of implementing the indoctrination and training program and contain provisions for documenting training sessions including content, date, attendance and results.

LIPA and/or supplier organizations shall provide for the initial qualification and refresher training of personnel to assure that they achieve and maintain proficiency to satisfactorily perform their safety-related functions. Training and qualification records shall be maintained.

Programs shall be established for indoctrination, training and, if appropriate, certification of personnel performing or verifying safety-related activities. The NQA Department Manager shall provide for QA indoctrination and specific training of NQA personnel, and for assurance of the satisfactory QA indoctrination of other personnel engaged in safety-related activities.

Formal QA training shall be accomplished in accordance with written procedures. These procedures shall describe the scope, objective and method of implementing the indoctrination and training program and contain provisions for documenting training sessions. These training documents are to include content, date, attendance and results. Personnel proficiency shall be maintained as necessary by means of refresher courses, reexamination and/or recertification.

This QA Program is designed to comply with the requirements of 10 CFR 50 Appendix B, 10 CFR 50.55a, other applicable Federal regulations, applicable NRC Regulatory Guides, and ANSI and ANS Standards as committed to in the Shoreham Defueled Safety Analysis Report (DSAR). The requirements stated in the QA Manual are applicable to materials, structures, systems, components and services whose satisfactory performance is required for safe storage and handling of nuclear fuel.

Outside contractors that perform safety related functions shall be required to comply with those portions of 10 CFR 50, Appendix B, and the LIPA Program that are applicable to the services provided. LIPA QA Procedures shall require that a review and evaluation report of a supplier's QA Program be available and accepted by LIPA NQA Department prior to the issuance of purchase orders for safety items, or services to assure that the program meets the applicable elements of 10 CFR 50, Appendix B.

Compliance with QA Program requirements by both internal and external organizations shall be assured by a comprehensive system of audits and reviews performed by NQA under the direction of the Manager, NQA Department. Significant changes to the DSAR that may occur between general review cycles shall be transmitted to organizations as defined in the applicable administrative procedures.

17.2.3 <u>Design Control</u>

The LIPA QA Program establishes measures to control design activities that affect the quality of safety related structures, systems, and components during the decommissioning phase. These measures are applicable to all organizations performing design, design review, or design audit activities including changes or modifications thereto. Section 3 of the LIPA QA Manual describes the QA Program requirements established to provide this control.

The program requires that design and modification activities be accomplished in a planned, controlled, orderly manner in accordance with established procedures. Design control measures shall assure the translation of applicable design bases, regulatory requirements, codes, and standards (which includes the selection of suitable materials, parts, equipment, and processes) into specifications, drawings, and documented procedures and instructions. The program requires that the quality requirements be included in the design documents.

Deviations from or changes to specified quality requirements in design documents shall be controlled. Suitable design control measures are required for design analysis such as physics, stress, thermal, seismic, hydraulic, radiation, and accident analysis; compatibility of materials; accessibility for maintenance, repair and rework and acceptance criteria for inspections and tests. Design control procedures shall identify and control design interfaces both internal and external to LIPA.

Design verification, such as design reviews, alternative calculations, or qualification testing, shall be properly selected and accomplished. Responsibility for such verification is described later in this section. Where qualification testing of a prototype is used to verify adequacy of design, testing shall be performed under the most adverse design conditions. The program requires that design verification be performed by individuals or groups other than the original designer and the designer's immediate supervisor, but verification may be performed by individuals from the same organization.

Design changes shall be subject to design control measures commensurate with those applied to the original design. Design control measures shall provide for the suitable review and selection of standard "off the shelf" commercial or previously approved material, parts, equipment, and processes that are essential to safety related structures, systems, and components.

Design documents and revisions thereto shall be distributed to the responsible individuals in a timely and controlled manner to prevent inadvertent use of superseded documents. Control of design documents is further described in Section 17.2.6. Design documents and reviews, records, and changes thereto are collected, stored, and maintained in accordance with Section 17.2.17. Errors or deficiencies that may arise during the design process shall be addressed in accordance with Sections 17.2.15 and 17.2.16.

Organizations supplying equipment and/or services are responsible for imposing the applicable requirements of this section on their internal operations and on those vendors and contractors performing work within the scope of their activity as required by the procurement documents. These organizations are responsible for assuring by means of audit or surveillance that design control as defined in their respective programs is being effectively implemented. LIPA is responsible for assuring program adequacy and implementation by external suppliers through planned and periodic audits.

The design change control programs also include provisions to ensure that nonsafety related structures, systems, components, and plant computer software shall continue to be accorded the safety significance given to them in the DSAR, Defueled Technical Specifications, and Emergency Operating Procedures.

The OMD and DED Managers are responsible for determining, initially, whether proposed modifications or repairs involve unreviewed safety questions or changes in technical specifications as described in 10 CFR 50.59. This determination shall be reviewed by the Site Review Committee (SRC) and forwarded to the Resident Manager for approval. Procedures shall provide documentation and control of such determinations.

Technical evaluation, including design verification, shall be the responsibility of the appropriate organization. The LIPA NQA Department is responsible for verifying overall program establishment and implementation through planned and periodic audits.

17.2.4 <u>Procurement Document Control</u>

The LIPA QA Program provides for the control of procurement documents for safety related material, equipment, and services whether purchased by LIPA or suppliers, during decommissioning. Section 4 of the LIPA QA Manual describes the QA Program requirements established to assure procurement document control.

The program requires that procedures establish measures to assure control of the preparation, review, approval, and concurrence of procurement documents. Document control procedures as described in Section 6 and as delineated in Table 17.2.6-1 shall be applied to procurement documents including changes and revisions. The procurement documents shall be reviewed by qualified personnel, as defined within this section, assuring the adequacy of the quality requirements. The review shall be utilized to assure that the quality requirements, including preparation, review, and approval, have been properly defined, that the procured items are inspectable and controllable, and that the acceptance criteria are adequately specified.

The program requires that procurement documents such as purchase specifications contain or reference the design bases technical requirements, which include codes, industry standards, and regulatory requirements; material and component identification requirements; drawings and/or specifications; test and inspection requirements; and special process instructions. In addition, procurement documents are required to identify the following:

1. Requirements of 10 CFR 50, Appendix B, with which the supplier QA Program must comply.

2. The document requirements for drawings; specifications; procedures; personnel and procedure qualifications; material, chemical, and physical test results; and inspection and test records that must be prepared, maintained, submitted, or made available for review and/or approval.

3. The requirements for the retention, control maintenance, and/or delivery of records.

4. LIPA's right of access to supplier's facilities and records for source inspection and audits.

Procurement documents for spare or replacement parts shall be subject to program requirements equivalent to those used for the original equipment or to those specified by a properly reviewed and approved revision.

The LIPA Finance and Administration Department is responsible for the commercial aspects associated with procuring items or services, which includes the processing of purchase orders. LIPA's organizations are responsible for assuring that the procurement documents contain technical and quality requirements as indicated above. Authorized release, assuring acceptability of both technical and quality content, is required prior to releasing a purchase order.

LIPA's Shoreham organizations shall prepare those procurement documents pertaining to their scopes of responsibilities and shall present those documents to the Finance and Administration Department for processing. The NQA Department is responsible for reviewing the procurement documents for quality requirements, and for the review of and concurrence with selected suppliers' QA Programs.

Consultants, architect-engineers, testing companies, etc. (collectively "contractors"), assigned responsibility by LIPA for procurement activities associated with safety related material, equipment, or services shall impose the control requirements indicated above. LIPA's contractors shall establish the requirements in procedures, instructions, drawings, etc. These requirements shall be imposed on the contractors' internal operations and on any vendors or contractors performing work within the scope of their activities as required by the procurement documents. The contractors shall assure the adequacy of program implementation through audit or surveillance. LIPA shall verify program adequacy and implementation by suppliers through planned and periodic audits consistent with the complexity, importance, and quality of items or services.

Personnel exercising their judgment with regard to procurement of nonsafety related structures, systems, components, and plant computer software shall assure that the safety significance accorded to them in the DSAR, Technical Specifications, and the Emergency Operating Procedures is maintained throughout the decommissioning of Shoreham.

17.2.5 Instructions, Procedures, and Drawings

The LIPA QA Program establishes provisions for activities affecting the quality of safety related structures, systems, and components during decommissioning to be accomplished and controlled in accordance with instructions, procedures, and drawings. Section 5 of the LIPA QA Manual describes the QA Program requirements for the control of instructions, procedures, and drawings. Organizational procedures delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of instructions, procedures, and drawings.

Suppliers, vendors, and contractors have the responsibility for establishing instructions, procedures, drawings, and other documents to control the quality related activities of their own operations and those of their subsuppliers, as required by the procurement documents. A description of the associated procurement document control requirements is in Section 17.2.4.

LIPA organizations are responsible for establishing instructions, procedures, and drawings or for utilizing established procedures, instructions, and other documents to control the quality related activities they perform. The required station procedures are described in Section 13.5 of the DSAR. All responsible organizations establish provisions such that the development and implementation of instructions, procedures, and drawings, including changes thereto, are clearly identified and controlled.

The LIPA NQA Department is responsible for performing review, surveillance, and audit functions to verify that the instructions, procedures, drawings, and other documents used for safety related structures, systems, and components are controlled to meet the requirements of 10 CFR 50, Appendix B.

Activities affecting the quality of safety related structures, systems, and components are defined in specifications, instructions, procedures, drawings, and other documents. These documents include qualitative and quantitative acceptance criteria for the activity being conducted. These criteria are used to control quality affecting activities; and to define special process controls, codes, standards, and regulatory requirements.

The LIPA NQA Department reviews all safety related test, calibration, special process, maintenance, modification and repair procedures, drawings and specifications, and changes thereto, with respect to quality requirements as indicated in Section 6 and delineated in Table 17.2.6-1.

17.2.6 Document Control

The LIPA QA Program provides for the control of documents, including changes thereto, which affect the quality of safety related structures, systems, and components during decommissioning. The applicable documents include, but are not limited to, the QA Manual; QA Procedures and Instructions; the Defueled Safety Analysis Report; design drawings; component specifications; procurement documents; supplier technical manuals, procedures and instructions. Section 6 of the LIPA QA Manual describes the QA Program requirements established to assure document control.

The program requires that a document control system be established in accordance with approved procedures and instructions for review, approval, and issuance of the documents, including changes thereto, to assure that they are adequate and incorporate the quality requirements prior to release. LIPA organizations that issue, review, and approve documents shall establish provisions for the identification of individuals or groups responsible for performing review, approval, issuance, or revision activities.

The program requires that changes to documents be reviewed and approved by the organization responsible for conducting the original review and approval or, as deemed necessary by LIPA, such changes will be reviewed and approved by another qualified and responsible organization. In the event that another qualified organization is charged with the responsibility for revision, that organization shall have access to pertinent background information for adequate understanding of the requirements and intent of the original document. Procedures and instructions provide measures to assure the prompt distribution of approved changes and revisions, including control of obsolete or superseded documents to prevent their inadvertent use. The program requires that the documents be available at the location where the activity will be performed prior to the start of work. Change or revision identification will be established and verified through the utilization of document distribution lists. Updating and distribution to personnel of such lists will be consistent with the nature of the document.

Suppliers of safety related items and services are responsible for imposing the above document control requirements on their internal operations and on those vendors and contractors performing work within the scope of their activities as required by the procurement documents. Suppliers shall assure program adequacy and implementation through planned and periodic audits. LIPA is responsible for assuring program adequacy and implementation by external suppliers through planned and periodic audits.

LIPA organizations that issue, review, and approve documents, including changes thereto, are responsible for establishing and implementing a document control system in accordance with the requirements indicated above. The LIPA NQA Department is responsible for assuring overall program adequacy and implementation through planned and periodic audits.

17.2.7 Control of Purchased Material, Equipment, and Services

The LIPA QA Program establishes measures to assure that safety related material, equipment, and services procured during decommissioning either directly or through contractors, conform to the procurement document requirements. Section 7 of the LIPA QA Manual describes the QA Program requirements established to provide this control.

The program establishes provisions for source evaluation and selection. Source evaluation and selection may be based upon historical quality performance data, source surveys or audits, or source qualification programs. This evaluation and selection process will determine the supplier's capability to supply the item or service in compliance with the design, manufacturing, and quality requirements as stipulated in the procurement documents. Measures are established to provide for both a technical and quality evaluation of those suppliers providing safety related components or services. LIPA's Shoreham organizations shall perform the technical evaluation, and the NQA Department shall perform the quality evaluation. These functions may also be accomplished through the utilization of qualified independent organizations. Personnel performing the evaluations, such as auditors, shall be qualified. Source evaluation and selection information shall be documented and filed.

The program provides for source inspection, surveillance, and audit of suppliers to assure conformance to procurement document requirements. The inspections, surveillance, and audits shall be conducted in accordance with documented procedures. Source inspection procedures provide for instructions to be established for specifying the characteristics to be witnessed, inspected or verified, and accepted; for indicating responsibility; and for determining documentation requirements.

Source audits or surveillance shall be conducted, as necessary, to assure compliance with quality requirements. Source inspection or audit may not be necessary when the quality of the item can be verified by review of test reports, inspection upon receipt, or other means.

The program requires that receiving inspection be accomplished in accordance with documented procedures and instructions. The receiving inspection procedures and instructions establish measures to assure that the item is properly identified and corresponds to the receiving documentation, that the item and acceptance records are determined to be acceptable in accordance with the inspection instructions prior to use, that the receiving documentation is available at the plant prior to use, and that the inspection status is identified as indicated in Section 17.2.14. The QA Program specifies that procurement documents require suppliers to furnish documentation identifying any procurement requirements that have not been met together with a description of these nonconformances marked "accept as is" or "repair". Responsible NQA and technical personnel shall perform a review and approval of the supplier's recommended disposition. Nonconforming items shall be identified and controlled as indicated in Section 17.2.15. Inspections shall be conducted based upon the nature of the item being procured.

When required by code, regulation, or contract requirements, documentary evidence that items conform to procurement requirements shall be available and readily retrievable at the plant. This documentary evidence shall specifically identify the item and codes and/or specifications met by the item. When not precluded by other requirements, such documentation may take the form of written certification of conformance identifying the requirements met by the items. LIPA QA Procedures require that suppliers' certificates of conformance be periodically evaluated by audits or tests to assure that they are valid.

Suppliers of safety related material, equipment, and services are responsible for imposing the control requirements indicated above on their internal operations and on any vendors or contractors performing work within the scope of their activities as required by the procurement documents. Suppliers shall assure through audit or surveillance the adequacy of program implementation. The LIPA Finance and Administration Department is responsible for commercial aspects associated with procuring items or services. The LIPA organizations that requisition items and/or services and the NQA Department are responsible for assuring that the procurement documents contain the information as required above. Procedures have been established to control spare and replacement part procurement documents, through technical and NQA review, to ensure that the controls for safety related items are equal to or better than the original equipment. The QA Program requires that a technical evaluation and NQA review be performed to determine which requirements are to be applied to the procurement of spare and replacement parts when the original equipment requirements are not known.

Procurement document control is described in Section 17.2.4. LIPA shall assure program adequacy and implementation of suppliers through planned and periodic audits consistent with the complexity, importance, and quality of the item or service. The LIPA NQA Department is responsible for evaluating suppliers. This evaluation shall include utilization of qualified independent organization surveys. Source inspection, as necessary, shall be conducted by LIPA or qualified independent organization. The LILCO Project Organization is responsible for receipt of items at the station.

The NQA Department, which is responsible for conducting receiving inspections of items with respect to quality requirements, assures overall program establishment and implementation through planned and periodic audits and surveillances.

17.2.8 Identification and Control of Materials, Parts, and Components

The LIPA QA Program requires the establishment of an identification and control system to prevent the use of defective, unapproved, or incorrect safety related material, parts, and components. Section 8 of the LIPA QA Manual describes the QA Program requirements established for this purpose.

The program requires that the identification system, including unique part or mark numbers developed during the design and construction phases, be maintained and expanded as necessary during decommissioning. A system for identification and control of materials, parts, and components (including partially fabricated subassemblies) shall be based on documented procedures and/or instructions. Identification is referenced in specifications, drawings, purchase orders, or other appropriate documents providing traceability to associated documentation such as manufacturing and inspection documents, deviation reports, heat numbers, and mill test reports. The identification may be placed either on the item or on records directly and readily traceable to the item. Physical identification shall be used to the maximum extent possible and shall be applied in such a manner as not to affect the function of the item. Verification of identification shall be accomplished at appropriate stages throughout fabrication, assembly, shipping and receiving, and prior to installation.

During decommissioning, suppliers of safety related material, parts, and components are responsible for establishing a system of identification and control that addresses the requirements outlined above. Suppliers are responsible for imposing the requirements on their internal operations and on those vendors and contractors performing work within the scope of their activities as stipulated in the procurement documents. Suppliers shall assure, through audit or surveillance, the adequacy of program implementation. LIPA shall assure program adequacy and implementation through planned and periodic audits of the suppliers.

The Material Management Division (MMD) is responsible for maintaining and expanding the identification and control system for safety related material, parts, and components. If a design change is necessary, the MMD is responsible for supplying identification requirements to the associated organizations and for assuring the continued implementation of the established identification and control system. The NQA Department is responsible for assuring overall program establishment and implementation through planned and periodic audits, surveillance, and inspections at the station.

17.2.9 Control of Special Processes

The LIPA QA Program imposes on organizations performing special processes the requirement to develop a system of special process controls. Special processes include, but are not limited to, special inspection or test processes, welding, heat treating, nondestructive examination (NDE), decontamination and radiological and chemical analyses. Section 9 of the LIPA QA Manual describes the QA Program requirements established for control of special processes.

The program requires that organizations performing special processes on safety related equipment at the nuclear power station or at an offsite facility do so using approved procedures, instructions, or the equivalent, and that equipment and personnel be qualified in accordance with applicable codes, standards, specifications, or special requirements. Special process procedures, in addition to providing for the qualification of equipment and personnel, shall provide for the documentation of accomplished activities. Where special processes are not covered by existing codes or standards, or where certain item quality requirements exceed the requirements of established codes or standards, the necessary qualification of personnel, equipment, or procedures shall be required. Special process procedures and qualification records shall be filed, maintained, and available for verification.

Suppliers of equipment and services whose scope of activity includes utilization and control of special processes are responsible for imposing these requirements on their internal operations and on those suppliers, vendors, or contractors performing work within the scope of their activity as required by the procurement documents. Special process controls shall be submitted to the suppliers for approval as specified in the procurement documents. Suppliers shall verify through audit or surveillance the adequacy of program implementation.

LIPA shall verify overall program adequacy and implementation by internal organizations and suppliers through planned and periodic audits.

17.2.10 <u>Inspection</u>

The LIPA QA Program provides for inspection of activities that affect the quality of safety related structures, systems, and components during decommissioning. Section 10 of the LIPA QA Manual describes the QA Program requirements established for inspection.

It provides for an inspection program to be implemented in accordance with applicable procedures, instructions, and checklists. Inspections shall be performed by individuals other than those who performed or directly supervised the activity being inspected. Inspection procedures, instructions, or checklists contain identification of responsibility for performance of the inspection, method of inspection, characteristics to be inspected, acceptance/rejection criteria, verification, evaluation, and documentation of the results of the inspection. The program requires that inspection procedures or instructions be made available for use, with supporting documents such as drawings and specifications, prior to the performance of inspection operations. Information concerning inspections shall be obtained from design specifications, drawings, and/or other controlled documents including codes, standards, and regulatory requirements. The inspections are conducted by inspectors who have been qualified and certified in accordance with codes, standards, and/or LIPA training programs. The inspection program requires that inspector qualifications be kept current. The respective managers shall be responsible for certifying their inspection personnel.

When notification or hold points are established in procurement or other documents, the inspection program requires that:

1. Work does not progress beyond the hold point until released by the designated authority.

2. The notification and acknowledgement has been satisfied prior to continuation of work.

Inspection of rework, repair, replacement, or modification activities shall be conducted in accordance with inspection requirements or by means of an approved alternative. Such alternatives shall be evaluated on both a technical and quality basis. When direct inspection is not possible, provisions are established for indirect control by monitoring of processing methods, equipment, and personnel.

Suppliers of safety related material are responsible for imposing the above requirements on their internal operations and on those vendors or contractors performing work within the scope of these activities as required by the procurement documents. Suppliers shall assure, through audit or surveillance the adequacy of program implementation.

LIPA, through planned and periodic audits, surveillance, and participation in selected inspections shall verify conformance of inspection programs delegated to external organizations. When inspections or other safety related activities are conducted by LIPA or an outside contractor at the station, the NQA Department is responsible for verifying that the inspection program complies with the requirements outlined above. The LIPA NQA Department is responsible for reviewing maintenance and modification procedures to assure that requirements such as the need for inspection, identification of personnel, and documentation of results have been addressed.

17.2.11 Test Control

The LIPA QA Program establishes provisions to assure that testing required to demonstrate satisfactory inservice performance of safety related structures, systems, and components is conducted in accordance with an approved, documented test program. Section 11 of the LIPA QA Manual describes the QA Program requirements established for test control during the decommissioning phase.

It is required that the test program be identified, documented, and accomplished in accordance with procedures that are written, approved, and controlled. The basis for determining when proof, preoperational, and operational tests are required to demonstrate satisfactory inservice performance are addressed in Section 17.2.14 and in the LIPA QA Manual. The QA Program has established that modifications, repairs, and replacements shall be tested in accordance with the original design and testing requirements or acceptable alternatives. Technical and NQA reviews provide assurance that the testing does accomplish this end. The test procedures contain or reference the requirements and acceptance limits from the applicable design or procurement documents. The procedures establish provisions to assure that prerequisites for a given test have been met. Prerequisites include: Test equipment is adequate and in satisfactory operating condition; test instrumentation has been properly calibrated; personnel are trained, qualified, and certified if necessary for the various test functions; preparation, condition, and completeness of the item to be used have been satisfactorily accomplished; suitable environmental conditions are available; provisions for data acquisition have been established; if necessary, mandatory inspection hold points for witness by the designated authority are included; appropriate acceptance and/or rejection criteria are established; and methods for documenting data and results are established. The program requires that test results be documented in sufficient detail to prevent misinterpretation, that they be evaluated to the established criteria, and that the acceptance status be identified by a qualified, responsible individual or group. Test records shall be appropriately filed upon completion of the test and evaluation.

Suppliers of safety related material and services are responsible for imposing the above requirements on their internal operations and on those vendors and contractors performing work within the scope of their activities as stipulated in the procurement documents. Suppliers shall assure, through audit or surveillance, the adequacy of program implementation. LIPA shall verify program adequacy and implementation by external suppliers through planned and periodic audits.

Responsibility for the station testing programs has been assigned to the Operations and Maintenance Department during Shoreham's decommissioning. The LIPA NQA Department is responsible for verifying overall program establishment and implementation through planned and periodic audits and surveillances.

17.2.12 Control of Measuring and Test Equipment

The LIPA QA Program imposes requirements for control of measuring and test equipment on organizations whose activities affect the quality of safety related structures, systems, and components. The program requires calibration control for the measuring and test instruments, tools, gauges, fixtures, reference and transfer standards, and nondestructive test equipment. Section 12 of the LIPA QA Manual defines the QA Program requirements established for control of measuring and test equipment.

The program requires that calibration procedures describe the technique, frequency, and maintenance for measuring and test equipment. The QA Program requires procedures to establish methods for identification of measuring and test equipment and associated calibration data including provisions to assure that documented control system to indicate the date of the next calibration. The frequency of calibration is established for measuring and test equipment on an individual basis or generic grouping thereof. It is based upon the type of equipment, required accuracy, stability characteristics, purpose, degree of usage, experience, manufacturers' recommendations, and recognized industry standards. The reference and transfer standards are traceable to nationally recognized standards and, for any exceptions, provisions are established to document the basis for calibration. The calibration program requires that, in the event an instrument is found to be out of calibration, an investigation shall be conducted and documented to determine the validity of previous measurements. It is required that calibration records be established and maintained to provide objective evidence that measuring and test equipment is being controlled, calibrated, and maintained in accordance with approved procedures.

Provisions assure that calibrating standards have an accuracy, range, and stability adequate to verify that the equipment being calibrated is within specified tolerance and can meet all other specified requirements.

The reference standard used as the working (shop) standard shall have a tolerance not greater than one-fourth the specified tolerance of the measuring and test equipment being calibrated, except when equipment acceptable for nuclear power plants applications is not commercially available. In those cases, instruments of equal or greater accuracy shall be used. The reference standards used to calibrate the working (shop) standards shall have an accuracy greater than that of the working (shop) standard. When reference standards used to calibrate the working (shop) standard have an accuracy equal to that of the working (shop) standard, the basis for the use of standards having the same accuracy shall be documented by responsible management.

Procedures shall be written to control and monitor the use of measuring and test equipment and reference standards to assure that the above requirements are maintained within the limitations noted. These procedures also assure that permanently installed operating instrumentation is calibrated against measuring and test equipment having a tolerance not greater than the specified tolerance of the installed instrumentation.

During decommissioning, suppliers of equipment and services whose scope of activity includes the utilization of measuring and test equipment on safety related structures, systems, and components are responsible for imposing the above control requirements on their internal operations and on those vendors and contractors performing work within the scope of their activities as required by the procurement documents. Suppliers shall assure, through audit and surveillance, the adequacy of program implementation. LIPA shall verify program adequacy and implementation through planned and periodic audits of suppliers.

LIPA station organizations such as Radiochemistry, Health Physics and Maintenance are responsible for maintaining control over the M&TE they utilize and for complying with the applicable requirements of this section.

The LIPA NQA Department is responsible for verifying program establishment and implementation through planned and periodic audits and surveillance.

17.2.13 Handling, Storage, and Shipping

The LIPA QA Program imposes control requirements on organizations whose scope of activity includes the handling, storage, and shipment of safety related structures, systems, and components during Shoreham's decommissioning. Section 13 of the LIPA QA Manual describes the QA Program requirements established for handling, storage, and shipment. Certain requirements are applied as necessary to non safety-related materials, equipment and services.

The program requires that organizations performing handling, storage, and shipping activities (including cleaning, packaging, and preservation) do so using written procedures or instructions. These procedures shall be developed in accordance with applicable design and specification requirements and shall provide for control of the aforementioned activities to preclude damage, loss, or deterioration of safety related materials, components, and equipment. Special environmental conditions (such as special coverings, inert gas atmosphere, allowable moisture content, and temperature level) shall be detailed, and their existence shall be verified and documented. Provisions for necessary cleaning operations, as required by the nature of the material or equipment, shall be included and their verification documented. Special handling requirements shall be provided and controlled to ensure safe and adequate handling, including associated verification and documentation. The procedures or instructions provide for inspection operations to verify conformance to established criteria, use of qualified personnel, and associated documentation. In addition, the procedures and instructions shall provide for the controlled release of safety related material, components, or equipment from storage for shipment or installation and for the verification and documentation thereof.

The program requirements are applicable to the stages of fabrication, manufacturing, and installation associated with decommissioning. Suppliers are responsible for imposing the requirements, as specified in the procurement documents, on their internal operations and on those vendors and contractors performing work within the scope of their activities. Suppliers also assure the adequacy of program implementation.

The LIPA NQA Department shall verify overall program adequacy and implementation by internal organizations and by suppliers through planned and periodic audits.

17.2.14 Inspection, Test, and Operating Status

The LIPA QA Program provides measures for indicating the inspection, test, and operating status of safety related structures, systems, and components. Section 14 of the LIPA QA

Manual describes the QA Program requirements r identification and control of inspection, test, and operating status.

The QA Program requires that organizations responsible for fabrication, storage, installation, testing, and operation of safety related components and systems identify and control the inspection, test, and operating status of these items. The status is identified and controlled through the utilization of status indicators (such as tags, markings, logs, shop travelers, stamps, inspection, or test records).

In addition, the QA Program requires the establishment of measures to control the use of the status indicators, including responsibility and authority for their application and removal and the unique identification of the individual involved. Associated procedures establish provisions to assure the performance of required tests and inspections including requirements that the identification of the status be known at any given time. The bypassing of required inspections, tests, and other critical operations is controlled through station administrative procedures. These administrative procedures shall be reviewed by the NQA Department. Procedures establish measures to indicate the operating status to prevent inadvertent operation of safety related systems, equipment, and components. They establish provisions so that the identification of operating status is known at any given time.

The programs assure that functions performed out of sequence are adequately documented and do not compromise system integrity. Procedures provide for the positive identification and control of nonconforming items in accordance with Section 17.2.15, to prevent their inadvertent use.

The program requirements are applicable to stages of fabrication, installation, testing, and operation associated with the decommissioning. Suppliers are responsible for imposing the requirements, as specified in the procurement documents, on their internal operations and on those vendors and contractors performing work within the scope of their activities. Suppliers also assure through audit or surveillance the adequacy of program implementation.

The LIPA NQA Department shall verify overall program adequacy and implementation by internal organizations as well as by suppliers through planned and periodic audits.

17.2.15 Nonconforming Materials, Parts, or Components

The LIPA QA Program imposes requirements for control of nonconforming safety related material, parts, and components. These requirements are applicable to organizations whose activities affect the quality of such safety related items during the decommissioning phase. Section 15 of the LIPA QA Manual describes the QA Program requirements established to assure control on nonconforming items to prevent their inadvertent use or installation.

The QA Program requires that a control system be established to address nonconformances in accordance with documented, approved procedures. The procedures establish measures to assure that nonconforming items and services are properly identified, documented, reviewed, segregated if practical, dispositioned, and reported to affected organizations.

In addition, the procedures establish provisions for designation of responsibility and authority for approval of the dispositioning of nonconforming items. The program requires that nonconforming items be documented and that such documentation include a clear identification of the nonconformance, a description of the nonconformance, the appropriate disposition including the approval signature, and the applicable inspection and test requirements. Nonconforming items shall be clearly identified as such and placed in a controlled segregated area, when practical, until proper disposition has been effected.

Nonconforming items may be dispositioned by accepting "as is," scrapping, repairing, or reworking. The acceptability of repaired or reworked nonconforming items is verified by reinspection. The reinspection of the item shall be in accordance with the original inspection requirements or by acceptable alternatives. The program requires that the appropriate repair, rework, and inspection procedures be documented. Nonconformance reports verifying the "accept as is" or "repair" disposition shall be made part of the required inspection records.

Suppliers of safety related materials, parts, and components are responsible for imposing the above requirements on their internal operations and on those vendors and contractors performing work within the scope of their activities as required by the procurement documents. They also assure, through audit or surveillance, program adequacy and implementation. LIPA is responsible for conducting audits to verify program adequacy and implementation by suppliers. The LIPA NQA Department is responsible for assessing the adequacy and implementation of suppliers' nonconformance control systems. This assessment is in addition to technical reviews of applicable nonconformance reports by other LIPA organizations. Safety related nonconformance reports shall be analyzed periodically to determine the existence of quality trends. Trends, if any, shall be reported to the appropriated LIPA management.

When a LIPA organization discovers a nonconformance related to a LIPA activity, it is that organization's responsibility to generate and control a nonconformance report in accordance with the requirements stated herein. In general, the organization responsible for the nonconforming condition is responsible to provide an acceptable disposition. The reporting organization and the NQA Department are required to review and accept the disposition before it may be implemented.

17.2.16 Corrective Action

The LIPA QA Program provides measures to assure that conditions adverse to quality are promptly identified, reported, and corrected. Section 16 of the LIPA QA Manual describes the QA Program requirements for corrective action and control thereof.

The program provides for a corrective action system implemented through the use of approved written procedures. The procedures provide for identification and documentation of deficiencies, including nonconformance reports, and determination of the need for corrective action. The procedures provide for reporting significant conditions adverse to quality, assessment of their probable root causes, and that the preventive and corrective actions taken be documented and reported to appropriate levels of management for review and assessment. Follow-up action shall be taken to assure proper implementation and timely closeout of corrective action.

Suppliers are responsible for establishing and implementing a corrective action program commensurate with the function they perform. The supplier systems provide measures that comply with the requirements outlined above and are imposed on internal operations as well as on vendors and contractors performing work within the scope of their activities as required by the procurement documents. Suppliers also assure, through audit or surveillance, the adequacy of implementation. LIPA shall verify overall program adequacy and implementation through planned and periodic audits.

The LIPA NQA Department shall be informed of corrective action determinations associated with safety related structures, systems and components. In addition, the NQA Department is responsible for verifying proper implementation of internal corrective action associated with safety related structures, systems, and components.

17.2.17 Quality Assurance Records

The LIPA QA Program imposes requirements on organizations performing safety related functions for QA records, which furnish documentary evidence of the quality of items and of activities affecting quality during the decommissioning phase. Section 17 of the LIPA QA Manual describes the QA Program requirements established for QA records.

The program requires that records documenting evidence of the quality of items and activities include results of reviews, inspections, tests, audits, and material analyses; monitoring of work performance; qualification of personnel, training procedures, and equipment; maintenance and modification activities; abnormal occurrences; and other documentation such as drawings, specifications, procurement documents, calibration procedures and reports, nonconformance reports, and corrective action reports. Requirements for identification, transmittal, retention, and maintenance of quality related records subsequent to completion of work or prior to release of material or equipment for installation are to be indicated in procurement documents, specifications, procedures, or instructions and are to be consistent with applicable codes and standards. The program requires that inspection and test records specify a description of the type of observation, identification of the inspector or data recorder, evidence of completion or verification of manufacturing, inspection or test operation, the data and results of the inspection or test, information related to nonconformances, and acceptability of the item inspected or tested.

The permanent plant filing system, developed during the design and construction phases and maintained during the operational phase, is known as the Shoreham Records Management System and is under the direction of the Nuclear Operations Support Department. This system assures that QA records are readily identifiable and retrievable. The QA Program requires that the record storage facilities be constructed or located, and secured to prevent damage or loss of records due to fire, flooding, or environmental conditions such as temperature or humidity or, alternatively, to maintain duplicate records stored in a separate remote location.

Suppliers performing safety related activities are responsible for imposing requirements for the generation, collection, storage, and maintenance of QA records on their internal operations and on those vendors and contractors performing work within the scope of their activities as specified in the procurement documents. Suppliers also assure, through audit or surveillance, the adequacy of program implementation.

The NQA Department shall verify overall program adequacy and implementation by LIPA internal organizations and suppliers through planned and periodic audits.

17.2.18 Audits

The LIPA QA Program establishes provisions for a comprehensive system of planned and periodic audits to verify implementation of program requirements. Section 18 of the QA Manual describes the QA Program requirements for audits.

The program requires that a comprehensive system of audits be established for both internal and external functions that affect safety related structures, systems, and components to verify compliance with QA Program requirements as well as with approved QA procedures, the Shoreham Defueled Technical Specifications, administrative controls, and regulatory requirements. Audits shall include evaluations of quality related practices, effectiveness of implementation, conformance to policy, work areas, activities and processes, and reviews of documents and records.

Audits shall be conducted to predetermined schedules. These schedules shall be reviewed, published annually, and updated as required. Audit frequency shall be based on the status, safety and importance of the audited activity and results of prior audits. Audits shall be scheduled to ensure that implementation of QA Program requirements and related supporting procedures receive a comprehensive audit at least every two (2) years. Those applicable elements of the QA Program in which quality related activities are more intensive and impacting upon daily operation shall be audited at least annually. Audits of nonroutine operations such as major modifications shall be scheduled as necessary.

Audits shall be conducted in accordance with written, approved procedures, plans, and checklists by qualified personnel not directly responsible for the area being audited. Audits shall provide for objective evaluation of the status and adequacy of the area audited.