

ATTACHMENT 3 TO SNRC-1664

The Shoreham Nuclear Power Station Defueled Safety Analysis Report

SHOREHAM DSAR
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General Load Reduction

Turbine Trip

Turbine Trip and Failure of Generator

Breakers to Open

Main Steam Isolation Valve Closure

Pressure Regulatory Failure - Open

Pressure Regulatory Failure - Closed

Feedwater Controller Failure-Maximum Demand

Loss of Feedwater HEating

Shutdown Cooling (RHR)

Malfunction-Decreasing Temperature

Inadvertent HPCI Pump Start

Continuous Control Rod Withdrawal During

Power Range Operation

Continuous Control Rod Withdrawal During

Reactor Startup

Control Rod Removal Error During Refueling

Fuel Assembly Insertion Error During

Refueling

Off-Design Oper Transient Due to Inadvertent

Loading of a Fuel Assembly

Inadvertent Loading and Operation of Fuel

Assembly in Improper Location

Inadvertent Opening of a Safety Relief Valve

Loss of Feedwater Flow

Loss of AC Power

Recirculation Pump Trip

Loss of Condenser Vacuum

Recirculation Pump Seizure

Recirculation Flow Control Failure -

Decreasing Flow

Recirculation Flow Control Failure With

Increasing Flow

Abnormal Startup of Idle Recirculation Pump

Core Coolant Temperature Increase

Anticipated Transient Without Scram (ATWS)

Cask Drop Accident

Miscellaneous Small Release Outside Primary

Containment

Off-Design Operational Transient as a

Consequence of Instrument Line Failure

Main Condenser Gas Treatment System Failure

Liquid Radwaste Tank Rupture

Control Rod Drop Accident

Pipe Breaks Inside the Primary Containment

(Loss-of-Coolant Accident)

Pipe Breaks Outside the Primary Containment

(Steam Line Break Accident)

Fuel Handling Accident

Worst Case Fuel Damage Event

Feedwater System Piping Break

Failure of Air Ejector Lines

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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 Lighting Company, hereafter known as LILCO, in support of its application to amend Facility Operating License NPF-82 as described in SNRC-1664.

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.

This report is intended to provide sufficient information to enable the NRC Staff to issue the license amendment as requested in SNRC-1650.

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, 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 1989. The estimated fuel heat load will reduce to approximately 250 watts by June 1991. Figure I-1 (taken from DSAR Section 15.1) 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.

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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 1.1-2. 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

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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 and are in a protected state. 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.
6. Secondary containment integrity will be maintained utilizing the Normal Ventilation System to provide a controlled and monitored release capability as discussed in Chapter 15 Safety Analysis.

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7. The steam and power conversion systems are not required to be operable or functional and are protected from significant degradation as described in DSAR Chapter 10.
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 LILCO 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. A new Quality Assurance Category IIA is defined in DSAR Chapter 3 for systems, structures, and components that no longer fulfill a safety function in support of a defueled reactor.
4. The Fire Protection Program is maintained as described in DSAR Section 9.5.1 and the FHAR.
5. An offsite Radiological Environmental Monitoring Program (REMP) is maintained as described in DSAR Section 11.6.
6. Changes to the LILCO Security Plan are being provided separately from the DSAR.
7. A Defueled Emergency Plan is being submitted separately for NRC review and approval via SNRC-1651.

1.2 GENERAL PLANT DESCRIPTION

The descriptions and design criteria contained under this heading in the latest revision of the Shoreham USAR remain unchanged.

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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 protected from degradation.

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 preservation concerns.
3. Protected - Those systems not to be operated in the defueled mode. These systems will be left in a deenergized safe state and layed-up in accordance with System Lay-up Implementation Package (SLIPs), which specify maintenance and custodial services necessary to protect them pending disposition of LILCO's operating license.

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.

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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.

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TABLE 1.2-1

STATUS OF PLANT SYSTEMS IN THE DEFUELED CONFIGURATION FOR AN
EXTENDED PERIOD OF TIME

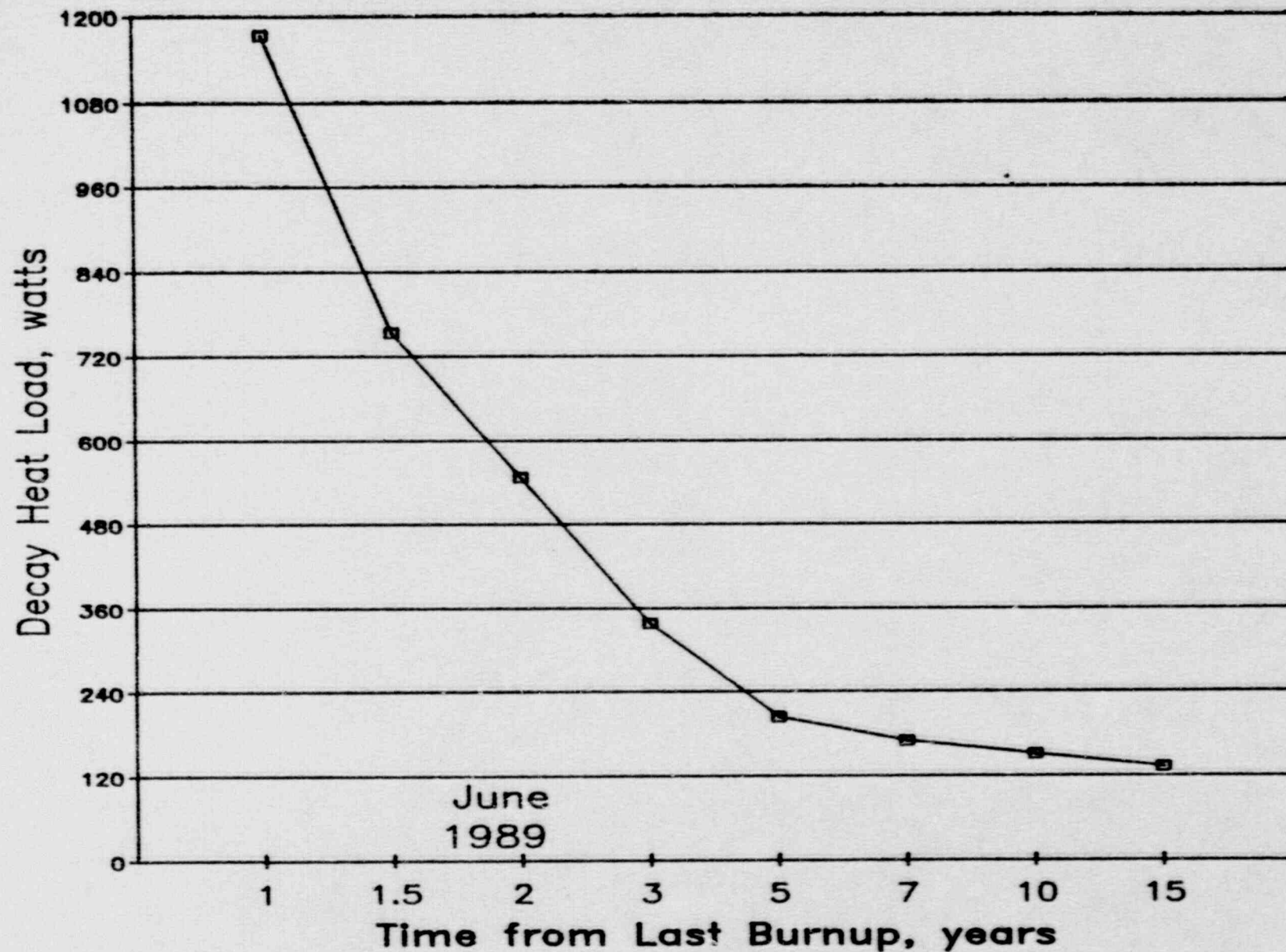
OPERABLE

(RB) Cranes, Hoists and Elevators
Reactor Building Superstructure
Process Radiation Monitoring
Area Radiation Monitoring
Servicing Aids (Fuel)
Refueling
Radwaste
Fire Protection (Mechanical)
Meteorological Monitoring
Station Transformer (NSS)
Non-Segregated Buses
Metal Clad Switchgear
Load Centers and Unit Substations
Fire Detect & Station Security
(Electrical/I&C)
138/69kv Switchyard Pot. Transf.
138kv Switchyard Relay Panels
Reactor Building
Reactor Building Ventilation
Reactor Building Standby Ventilation (shared
portion only)
Seismic Monitoring

OPERABLE: System(s) maintained to meet Technical Specifications.

DSAR FIGURE 1.1-1

SNPS Spent Fuel Decay Heat Load



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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

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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 model the dispersion of plant gaseous effluents and are used as input to required periodic reports.

To ensure the operability of the system, quarterly 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 quarterly basis with identical equipment which have been calibrated in the laboratory of a qualified vendor. Vendor personnel perform the actual sensor substitutions under the direction of LILCO technicians. LILCO technicians perform the normal monthly maintenance procedures on instrumentation at the base of 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 rapid replacement of any malfunctioning components of the system. In the event that a meteorological tower is damaged, with one or more monitoring instrumentation channels 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. Refer to USAR for information on this subject.

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.

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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.

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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.

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CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, 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 not to be applicable to a defueled reactor:

I Overall Requirements

- GDC5 Sharing of structures, systems, and components
- Shoreham is a single unit, thus the above criterion does not apply.

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II Protection By Multiple Fission Product Barriers

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 Protection And Reactivity Control Systems

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

IV Fluid Systems

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

V Reactor Containment

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

Applicable Criterion Conformance

Quality Standards and Records (Criterion 1)

Criterion

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 LILCO QA program described in DSAR Chapter 17 assures

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that quality practices and documentation are maintained commensurate with the classification that is identified in this Defueled Safety Analysis Report (DSAR). A new Q.A. Category IIA is provided for USAR safety-related structures, systems, and components that no longer fulfill a safety function for a defueled reactor.

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.

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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.

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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 systems 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

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fuel drop accident. Two 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. Emergency Diesel Generators are not required. A further discussion of electric power requirements can be found in Chapter 15.

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.

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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

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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.

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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 secondary containment negative pressure. 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 the Reactor Building Standby Ventilation System is not 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

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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

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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

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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 of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

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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, LILCO Quality Assurance Category, with the notation I.

A key of definitions is provided at the end of Table 3.2-1. A new designation, Q.A. Category IIA, is utilized for systems, structures and, components originally Q.A. Category I that are no longer required to meet 10CFR50 Appendix B in the defueled condition. Chapter 17 discusses the graded level of Q.A. requirements for this equipment.

A Q.A. Category IIA designation thus indicates that the item was re-classified with respect to USAR Table 3.2.1-1.

A IIA and II classification provided together for a given system on Table 3.2-1 implies that the portion of the system that was originally Q.A. Category I in the USAR is now Q.A. Category IIA and the Q.A. Category II portion remains unchanged.

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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. That protection is adequately provided by the reactor building wall and roof structures and also by the spent fuel pool structure.

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. That protection is adequately provided by the reactor building wall and roof structures and also by the spent fuel pool structure.

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.

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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. This equipment is given a Q.A. Category IIA designation (See Chapter 17 and Table 3.2-1) so that deviations from original seismic requirement can be tracked.

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 IIA so that deviations from the EQ program can be tracked.

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 IIA whereby deviations can be tracked in accordance with the LILCO Q.A. Program.

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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.

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TABLE 3.2-1EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

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

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TABLE 3.2-1
(Continued)EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

<u>SYSTEM/ COMPONENT</u>	<u>LILCO QUALITY ASSURANCE CATEGORY</u>	<u>SEISMIC CATEGORY</u>	<u>COMMENTS</u>
XV <u>In-vessel Service Equipment</u>			
1. Control rod grapple	I	I	
XVI <u>Refueling Equipment</u>			
1. Refueling platform	I	I	
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 <u>Storage Equipment</u>			
1. New Fuel Storage Racks	IIA	N/A	NR
2. Defective fuel storage container	I	I	
3. Spent fuel pool, dryer/sep. pool, reactor cavity	I	I	
4. Spent fuel storage racks	I	I	
XVIII <u>Radwaste System</u>	IIA & II	N/A	
XIX <u>Reactor Water Cleanup System</u>	IIA & II	N/A	
XX <u>Fuel Pool Cleanup Subsystem</u>			
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	

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TABLE 3.2-1
(Continued)EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

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

* Certain components such as fans and valves will remain functional to support RBNVS operations.

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TABLE 3.2-1
(Continued)EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

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

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TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

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

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TABLE 3.2-1
(Continued)

EQUIPMENT CLASSIFICATION
SPENT FUEL STORAGE

<u>SYSTEM/ COMPONENT</u>	<u>LILCO QUALITY ASSURANCE CATEGORY</u>	<u>SEISMIC CATEGORY</u>	<u>COMMENTS</u>
XLIII <u>Primary Containment Structure</u>	IIA	N/A	NR
XLIV <u>Safety Parameter Display System</u>	IIA & II	N/A	NR
XLV <u>Post Accident Sample System</u>	II	N/A	NR
XLVI <u>Containment Isola- tion Valve Position Indicator</u>	IIA & II	N/A	NR
XCVII <u>Accident Monitoring Instrumentation</u> (NUREG 0578)	IIA	N/A	NR

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TABLE 3.2-1
(Continued)

KEY

LILCO Quality Assurance Category:

- I - Meets 10CFR50 Appendix B requirements (same as USAR).
- IIA - Systems which were safety related are now non-safety related. Deviations from safety related requirements will be documented (see Section 17.2.2).
- II - Meet requirements of purchase specification (same as USAR).

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

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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 Section 5.4.

4.1.2 Reactor Internal Components

Fuel Rod

A fuel rod consists of uranium dioxide (UO_2) pellets and a zircaloy-2 cladding tube. A fuel rod is made by stacking pellets into a zircaloy-2 cladding tube that has been evacuated and backfilled with helium. The tube is sealed by welding zircaloy end plugs in each end of the tube.

The BWR fuel rod is designed as a pressure vessel. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in the mechanical design and stress analysis of the fuel rod.

The rod is designed to withstand the applied loads, both external and internal. The fuel pellet is sized to provide sufficient volume within the fuel tube to accommodate differential expansion between the fuel and cladding. Overall fuel rod design is conservative in its accommodation of the mechanisms affecting fuel in a BWR environment.

Fuel Bundle

Each fuel bundle contains 62 fuel rods and two water rods, which are spaced and supported in a square (8 x 8) array by a lower and upper tie plate. The fuel bundle has two important design features:

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1. The bundle design places minimum external forces on a fuel rod; each fuel rod is free to expand in the axial direction.
2. The unique structural design permits the removal and replacement, if required, of individual fuel rods.

The fuel assemblies that make up the core are designed to meet all criteria for core performance and to provide ease of handling. Selected fuel rods in each assembly differ from the others in uranium enrichment. This arrangement produces more uniform power production across the fuel assembly, thereby significantly reducing the amount of heat transfer surface required to satisfy the design thermal limitations.

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 remains unchanged. Refer to the USAR for information on this subject.

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).⁽¹⁾

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.

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CHAPTER 5

REACTOR COOLANT SYSTEM

The reactor coolant system includes those systems and components that contain or transport fluids coming from or going to the reactor core. In the plant's defueled condition, the fuel is not in the core and the reactor is depressurized. Therefore, the reactor coolant system is not required and all sections, including Appendices, of USAR Chapter 5 are not applicable to the DSAR. The possible exception is that if the reactor is layed up wet, the RWCU System will be utilized.

5.5.7 Residual Heat Removal System

The Residual Heat Removal (RHR) System is described in the USAR. In the defueled status of the Shoreham Nuclear Power Station the RHR System serves no function. This system is protected.

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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 transient may result in loss of subatmospheric conditions within the secondary containment. However, as explained in Chapter 15 the time period before any

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significant loss of fuel pool water level is long, and appropriate corrective action is postulated to be taken before fuel damage occurs. There are no radiological consequences associated with this event.

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 Core Spray System

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. The CS System is secured in a protected state.

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6.4 HABITABILITY SYSTEMS

The systems, aside from the control room air conditioning portion, are secured in a protected state 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).

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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).

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CHAPTER 7

INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This chapter presents the details of the control and instrumentation systems in the plant except radiation monitoring systems, and electrical power systems which are described in Chapters 8, 11, and 12.

7.1.1 Identification and Classification

All of the instrumentation and control systems listed below which were classified in the USAR as Q.A. Category I have been reclassified as Q.A. Category IIA. USAR Figure 7.1.1-1 is no longer applicable due to the defueled status of the plant.

7.1.1.1 Identification of Individual Systems

This section identifies the individual systems which are retained and presents a general description of their instrumentation and control functions.

7.1.1.1.1 Reactor Protection System

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

7.1.1.1.2 Nuclear Steam Supply Shutoff 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.

7.1.1.1.3 Emergency Core Cooling System

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

7.1.1.1.4 Neutron Monitoring System

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

7.1.1.1.5 Refueling Interlocks

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

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7.1.1.1.6 Reactor Manual Control System

Reactor vessel instrumentation monitors and transmits information concerning key reactor vessel operating variables. Portions of this system will only be used if a wet layup of the reactor vessel is utilized.

7.1.1.1.7 Reactor Vessel Instrumentation

This system will be used only if a wet layup of the reactor vessel is utilized.

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.

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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

The reactor water cleanup (RWCU) system instrumentation and controls provide manual initiation of system equipment to maintain high water purity in the reactor water. This system will be used only if the reactor system is placed in a wet layup condition.

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 not needed to support the storage of the fuel in the fuel pool.

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 pressurized air compressors.

7.1.1.1.22 Service Water System

Under normal conditions the service water system provides cooling for the plant components.

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.

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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.

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.

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 has been secured 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 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 the air supply fans.

7.1.1.1.29 Chiller Equipment Room Ventilation System

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

7.1.1.1.30 Diesel Generator Room Emergency Ventilation Systems

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

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 are maintained functional to automatically control the ventilation system to maintain these rooms at their design temperature and system performance.

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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.

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.

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.

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.

7.1.1.2 Classification

Section 3.2 provides a reclassification of systems based on their importance to safety. Q.A. Category IIA applies to those systems that no longer fulfill a safety function for a defueled reactor.

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7.1.2 Identification of Safety Design Bases and Nonsafety Design Bases Criteria

The following sections remain as stated in the USAR with the exception that Q.A. Category I systems or portions of systems have been reclassified as Q.A. Category IIA:

- 7.1.2.1.7 Reactor Vessel Instrumentation
(If the reactor is layed up wet)
- 7.1.2.1.12 Screenwell Pumphouse Ventilation System
- 7.1.2.1.16 Reactor Water Cleanup System
(If the reactor is layed up wet)
- 7.1.2.1.19 Radwaste System
- 7.1.2.1.21 TBCLCW System
- 7.1.2.1.22 Service Water System
- 7.1.2.1.27 Fuel Pool Cooling and Cleanup System

The cooling portion of this system is not required for the defueled plant.
- 7.1.2.1.28 Control Room Air Conditioning System
- 7.1.2.1.29 Chiller Equipment Room Ventilation System
- 7.1.2.1.31 Relay Room, Emergency Switchgear Room, and Computer Room Air Conditioning System
- 7.1.2.1.32 Battery Room Ventilation System
- 7.1.2.1.35 Motor Control Center Room Ventilation System
- 7.1.2.1.36 Motor Generator Room Ventilation System

All other systems listed in subsections of the USAR under 7.1.2 are not needed.

7.2 REACTOR PROTECTION SYSTEM

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

7.3 ENGINEERED SAFETY FEATURE SYSTEM

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

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7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

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

7.5 SAFETY RELATED DISPLAY INSTRUMENTATION

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

7.6 ALL OTHER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

The following sections remain as stated in the USAR with the exception that safety related systems have been reclassified to nonsafety related systems:

7.6.1 Description

Only the cleanup portion of the fuel pool cooling and cleanup system is required.

7.6.1.2 Fuel Pool Cooling and Cleanup System Instrumentation and Controls

The cooling portion of this system is no longer required because the transferred fuel in the pool has minimal decay heat and therefore active cooling and circulation of water in the spent fuel pool is not required.

Fuel Pool Level Instrumentation:

The spent fuel pool level is indicated and alarmed on high and low levels in the main control room. The purpose of this instrumentation is to ensure that the water level in the spent fuel pool is maintained at sufficient height to provide shielding for normal building occupancy. If the low level alarm annunciates, the control room operator will notify the fuel handling personnel to evacuate. To ensure that the refueling floor personnel know what the radiation levels are on the refueling floor, three area radiation monitors are provided and are set to alarm at 5 mr/hr.

Fuel Pool Temperature Instrumentation

The spent fuel pool temperature is indicated and alarmed on high temperature in the main control room. The purpose of this instrument is to ensure that the maximum bulk pool temperature does not exceed 125°F design temperature. Based on the low fuel heat load it is not expected that the pool could reach this temperature.

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7.6.2.2.1 General Functional Requirements

Conformance of Fuel Pool Cooling and Cleanup System Instrumentation and Controls

All other USAR Part 7.6 sections not listed above are not needed in the defueled condition.

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

The following sections remain as stated in the USAR:

7.7.1.6 Liquid Radwaste Control System Instrumentation and Controls

7.7.1.7 Turbine Building Closed Loop Cooling Water System Instrumentation and Controls

7.7.1.8 Reactor Water Cleanup System Instrumentation and Controls

Required functional to support wet layup of reactor only.

7.7.1.9 Reactor Vessel Instrumentation As It Pertains To Water Level Instrumentation Only

Required functional for wet layup of reactor systems.

7.7.1.11 Refueling Interlocks Instrumentation and Controls

7.7.2.6.1 General Functional Requirements Conformance for Liquid Radwaste System Instrumentation and Controls

7.7.2.7.1 General Functional Requirements Conformance for Turbine Building Closed Loop Cooling Water System Instrumentation and Controls

7.7.2.8.1 General Functional Requirements Conformance for Reactor Water Cleanup System Instrumentation and Controls

Required functional to support wet layup of reactor only.

7.7.2.9.1 General Functional Requirements Conformance for Reactor Vessel Instrumentation as it pertains to water level instrumentation only

Required functional to support wet layup of reactor only.

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7.7.2.11.1 General Functional Requirements Conformance for
Refueling Interlocks Instrumentation and Controls

All other USAR Part 7.7 sections not listed above are not needed.

7A Plant Nuclear Safety Operational Analysis

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

7B Analog Transmitter/Trip Unit System for Engineered Safeguard
Sensor Trip Units

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

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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. Therefore, as stated in Section 8.3.1 item 2, diesel generator and safety related equipment will not be required while the plant is in the defueled mode.

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 from the 138KV switchyard. These units are connected into one of the 138KV

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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 OnSite 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. However, the Emergency Diesel Generators are not required to operate to maintain safe conditions.

8.1.5 On Site DC Power System

The DC system consists of two independent systems containing total of 6 separate and independent 125V DC battery sources. These are designed to provide a minimum of two hours of DC emergency power. However, since the defueled status of the system has reduced the emergency power load on these batteries, this will extend the duration of power supply from each source.

The onsite DC power system as described in the USAR remains unchanged in the defueled condition except for the following:

- a) 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.
- b) The safety related battery source and equipment is not required with the plant in the defueled state.

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.

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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 described in DSAR Section 8.2.1.2 are not required to be maintained as safety related while in the defueled condition. They are reclassified as Category IIA.

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 except as follows:

1 - Figure 8.2.1-1 Main one line diagram

The diesel generators shown are not operational in the defueled condition.

2 - Figure 8.2.1-2 One line diagram of 138kV and 69kV systems in Shoreham Area

Added to the system are the New Brookhaven Combustion Turbines.

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 IIA 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:

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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.

8.3.1 AC Power Systems

The Plant electrical power (AC) layout configuration as designed/described in the USAR is not changed in the defueled condition, except as follows:

- 1- Equipment, switchgear, or buses built and designed to safety standards are not maintained as safety related or service inspected in the defueled condition.
- 2- Diesel generator sets with safety related equipments are no longer needed for the plant in the defueled condition. (see USAR 8.3.1.1.5 onsite standby power supply).
- 3- Adequate equipment protection and emergency measures are available for the required plant electrical systems in the defueled condition.

Appropriate layup measures have been taken to protect the longevity of the various electrical equipments. The equipment, switchgear, and bus have been reclassified to Q.A. Category IIA.

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 IE/safety related functions of the DC system no longer exist.

Appropriate layup measures have been taken to protect the longevity of the various DC electrical equipment. Q.A. Category I equipment is now Q.A. Category IIA.

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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.

8.3.1 AC Power Systems

The Plant electrical power (AC) layout configuration as designed/described in the USAR is not changed in the defueled condition, except as follows:

- 1- Equipment, switchgear, or buses built and designed to safety standards are not maintained as safety related or service inspected in the defueled condition.
- 2- Diesel generator sets with safety related equipments are no longer needed for the plant in the defueled condition. (see USAR 8.3.1.1.5 onsite standby power supply).
- 3- Adequate equipment protection and emergency measures are available for the required plant electrical systems in the defueled condition.

Appropriate layup measures have been taken to protect the longevity of the various electrical equipments. The equipment, switchgear, and bus have been reclassified to Q.A. Category IIA.

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. X
- 2- All class IE/safety related functions of the DC system no longer exist. X

Appropriate layup measures have been taken to protect the longevity of the various DC electrical equipment. Q.A. Category I equipment is now Q.A. Category IIA.

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For further information on this subject, refer to the USAR.

Tables - 8.3.1.1, thru table 8.3.1.7A - related Emergency Diesel Generator System loads, demands, sequencing and margin test results are no longer applicable in the defueled condition.

Tables - 8.3.2.1 and 8.3.2.2 - related to plant design basis loads are no longer applicable in the defueled condition.

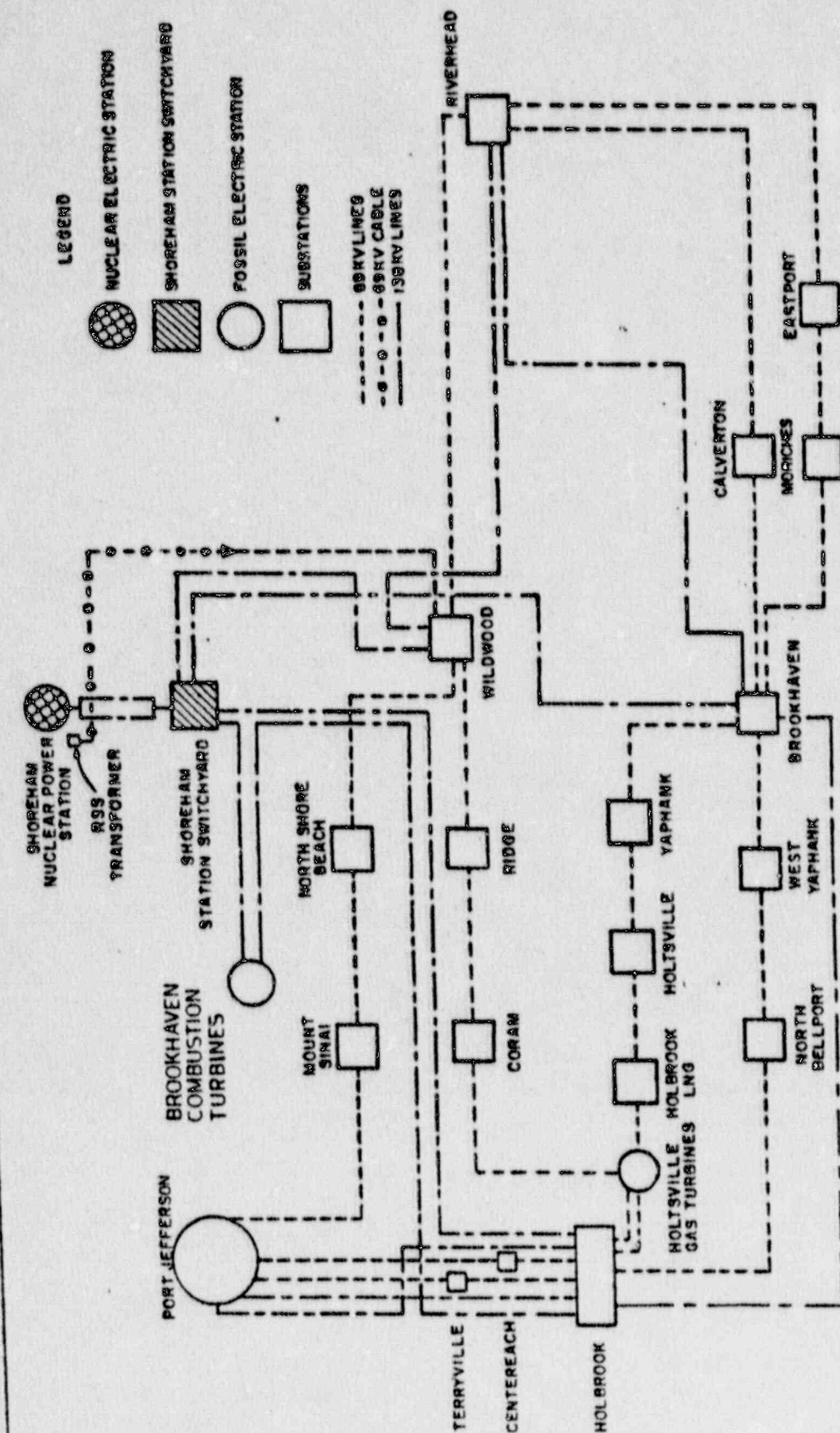


FIGURE 6.2.1-2
ONE LINE DIAGRAM OF 138 KV
AND 69 KV SYSTEMS IN THE
SHOREHAM AREA
SHOREHAM NUCLEAR POWER STATION
DEFUELED SAFETY ANALYSIS REPORT

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CHAPTER 9

AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 New Fuel Storage

Since no new fuel will be received, this section of the USAR is not required in the defueled condition.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

1. Spent fuel storage space is designed to accommodate 2,176 fuel assemblies (390 percent of the full core fuel load); however, currently only 1,420 storage cavities have been installed and are available to receive spent fuel assemblies (See Figure 9.1.2-1 revised for DSAR).

The remainder of this Section, paragraphs 9.1.2.1.2 through 9.1.2.1.10, is identical to the USAR.

9.1.2.2 Facilities Description

Spent fuel storage racks provide a place in the spent fuel pool for storing spent fuel received from the reactor vessel. The location of the spent fuel pool within the reactor building is shown on Figure 3.8.1-4. The general arrangement of the storage space, illustrated on DSAR Figure 9.1.2-1, will permit the storage of 2,176 fuel assemblies (the current installed capacity in the spent fuel pool is for 1,420 fuel assemblies) plus 144 control rods.

The remainder of this Section is identical to that in the USAR, up to the first paragraph of page 9.1-8. From this point to the end of page 10, the text is deleted.

9.1.2.3 Safety Evaluation

This section remains identical to that in the USAR except that in the DSAR Appendix 9A provides the criticality analysis.

9.1.2.4 Tests and Inspection

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

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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, Fire Protection Water System, and the Service 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.

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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 applicable.

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 9.2.1.4 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. No service water is required for RHR, diesel engine cooling, RBCLCW, drywell cooling, and makeup water to the reactor vessel ultimate cooling connection (UCC). The testable check valve in the UCC will not require

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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) All pumps and motor operated valves will actuate upon manual initiation signals only. Automatic start/initiation due to accident signals will be defeated.
- d) The double isolation valves which split the RBSV from the TBSW subsystems may be opened to intertie the subsystems as required.
- e) Normal operation will now consist of only one TBSW 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, the circulating water pump bearing and the fish retention pool. Cooling water for the vacuum priming pump seal cooler is not required. The second TBSW pump would remain in automatic standby while the third pump would be off and rotate in the operational mode with the other two pumps.
- f) The only tests and inspections to be performed on the TBSW system in the defueled condition are those that are deemed to be required for proper operation and maintenance.
- g) Table 9.2.1-1 has been revised.

Section 9.2.1.5 remains unchanged.

9.2.2 Reactor Building Closed Loop Cooling Water (RBCLCW) System

None of the equipment that uses RBCLCW (refer to USAR Section 9.2.2.2) is required to operate with the fuel in the spent fuel pool. The one possible exception is the RWCU pump which may be needed if the reactor is layed up wet. Therefore, this system is not required for operation in the defueled condition.

9.2.3 Makeup Water Demineralizer System

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

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1. SBLC, RBCLCW, seal water injection, and vacuum priming are no longer considered principal 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.

For further information relative to this system refer to the USAR.

9.2.4 Potable and Sanitary Water Systems

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.

9.2.5 Ultimate Heat Sink

With the fuel in the Spent Fuel Pool, the ultimate heat sink (Long Island Sound) no longer has any safety significance, since the decay heat of the fuel is insignificant. However, the ultimate heat sink will continue to be used as a source of cooling water for normal plant needs (refer to DSAR Section 9.2.1).

9.2.6 Condensate Storage Facilities

While in the defueled condition the condensate storage facilities provide makeup water for the fuel storage pool. The description of this system in the USAR remains unchanged except as follows:

1. Condensate, feedwater, reactor systems, HPCI and RCIC will no longer be primary users.
2. HPCI test discharge and CRD pump return lines to the CST are not required to be active.
3. The first three paragraphs of USAR 9.2.6.3 are no longer applicable.
4. The last paragraph of USAR 9.2.6.4 and 9.2.6.5 is no longer applicable.

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9.2.7 Turbine Building Closed Loop Cooling Water System

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged in the defueled condition. The only exception is that many of the coolers listed in DSAR Table 9.2.7-1 will normally be valved out of service while the plant remains in the defueled state.

For further information on this subject refer to the USAR.

9.2.8 Main Chilled Water System

This system will not be maintained as an operable system since it is not needed with the plant in the defueled condition.

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

Redundancy in this system is not needed since the RBSVS system is not required to operate 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. 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 IIA.
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.

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9.3.2 Process Sampling System

The Process Sampling System provides monitoring of certain process operations while fuel is in the spent fuel pool for either short or long term storage. The process monitoring is accomplished as necessary by means of measuring, analyzing and/or recording for conductivity, pH, and silica concentration, as shown on DSAR Table 9.2.2-1.

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.

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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 IIA and the filter portion of the system 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. All automatic initiation systems and interlocks for the habitability portion of the system will be secured 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 in 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. Support monitor for radioactive release through the exhaust air system.
4. Induces negative pressure in reactor building for secondary containment integrity.

For further discussion on this system refer to the USAR.

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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. 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 accelerate the exhaust air from the radwaste building and the reactor building. However, only one fan will be needed for this purpose, allowing one fan to be secured and still maintain a 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 IIA.

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 IIA.

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.

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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

This system is not needed while the fuel is stored in the spent fuel pool. This system is reclassified to Q.A. Category IIA.

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. Refer to USAR for information on this subject. This system is reclassified to QA Category IIA.

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 program 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.

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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 9.5.1.

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 APCS 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 IIA. 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 IIA. 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.

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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 IIA. Therefore, this proposed modification will not be implemented. The CO₂ systems for those rooms are in electric lockout. When a fire is detected, the CO₂ 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 IIA, 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 IIA, 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.

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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 IIA. 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 IIA. 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:

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Areas protected by CO₂ 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.

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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

Since emergency power is no longer required with the fuel in the Spent Fuel Pool, the Emergency Diesel Generators are not required, and sections 9.5.4 - 9.5.7 of the USAR no longer apply.

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.

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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.
8. Air for standby diesel generators.

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.

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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 maximum loading enrichment of 2.33 wt. % is found to be about -6.0% in k ($k - 0.060$). This brings the basic cell k under nominal storage conditions for the current fuel down to 0.85, which is well below the regulatory acceptance criterion of $k = 0.95$. 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) 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 ($k = 0.85$), no additional credit will be claimed here.

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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.

TABLE 9.2.1-1SERVICE WATER SYSTEM COMPONENT DESIGN DATA

<u>Component</u>	<u>Quantity</u>	<u>Nominal Capacity Each (gpm)</u>	<u>Number of Components Utilized in Defueled Condition</u>
Reactor Building Service Water Pumps	4	8,600	1
Turbine Building Service Water Pumps	3	8,000	1
Reactor Building Subsystem Components:			
Reactor Building Service Water Strainers	4	250	1
Diesel Generator Jacket Coolers	3	700	---
Residual Heat Removal Heat Exchangers	2	8,000	---
Reactor Building Closed Loop Cooling Water Heat Exchangers	2	6,370	---
Reactor Building Standby Ventilation System Chiller Condensers	4	525	1
Main Chilled Water System Chiller Condensers	3 1	1,500 400	---
Drywell Cooling Booster Heat Exchangers	2	1,460	---

TABLE 9.2.1-1SERVICE WATER SYSTEM COMPONENT DESIGN DATA (Cont'd.)

<u>Component</u>	<u>Quantity</u>	<u>Nominal Capacity Each (gpm)</u>	<u>Number of Components Utilized in Defueled Condition</u>
Turbine Building Subsystem Components:			
Turbine Building Service Water Strainers	2	420	1
Circ Water Pump Bearing Cooling	4	6	---
Fish Retention Pool	1	185	1
Turbine Building Closed Loop Cooling Water Heat Exchangers	2	14,200	1
Vacuum Priming Pumps Seal Water Coolers	3	100	---

TABLE 9.2.7-1

LIST OF COOLERS CAPABLE OF BEING SERVICED BY TURBINE BUILDING
CLOSED LOOP COOLING WATER SYSTEM

<u>Component Coolers</u>	<u>Quantity</u>
Hydrogen coolers	4
Excitor alternator cooler	1
Generator leads cooler	1
Generator stator winding coolers	2
EHC coolers	2
Air compressors	3
Condensate pump motor thrust bearing coolers	2
Condensate booster pump lube oil coolers	4
Sample temperature bath coolers	2
Condensate air removal pump oil coolers	2
Condensate Air removal pump sealing water heat exchangers	2
Reactor feed pump turbine oil coolers	4
Main turbine lube oil coolers	2
Offgas vent coolers	2
Waste evaporator overhead condenser	1
Regenerant evaporator overhead condenser	1
Waste evaporator distillate cooler	1
Regenerant evaporator distillate cooler	1
Waste evaporator bottoms cooler	1
Regenerant evaporator bottoms cooler	1
Mechanical seal cooler heat exchangers on various radwaste process pumps	> 10

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TABLE 9.3.2-1

PROCESS SAMPLING SYSTEM

<u>Description of Sample</u>	<u>Type of Sample</u>	<u>Purpose</u>
<u>RADWASTE SYSTEM</u>		
Regenerant Liquid and Evaporator Feed Tanks Recirculation Line	CC, Cph, Grab	Sample regenerant liquid evaporator feed tanks for process data
Discharge Waste Sample Tanks Recirculation Line	CC, Grab	Sample discharge waste sample tank for process data
Waste Collector Tanks Recirculation Line	CC, Grab	Sample waste collector tank for process data
Floor Drain Collector Tanks Recirculation Line	CC, Grab	Sample floor drain collector tank for process data
Recovery Sample Tanks Recirculation Line	CC, Grab	Sample floor drain collector tank for process data
Radwaste Demineralizer Outlet	Grab	Demineralizer efficiency
Radwaste Filter Effluent	Grab	Filter efficiency
Final Discharge Sampling Point	Grab	Process data, prior to discharge
Floor Drain Filter Effluent	Grab	Process data
Laundry Drain Tanks Discharge	Grab	Process data
<u>MAKEUP DEMINERALIZER SYSTEM</u>		
Individual Demineralizer Effluents	CC, Grab, IX	Demineralizer efficiency and makeup water quality
Dilute Caustic for Regeneration	CC	Caustic concentration

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TABLE 9.3.2-1 (Cont'd)

PROCESS SAMPLING SYSTEM

<u>Description of Sample</u>	<u>Type of Sample</u>	<u>Purpose</u>
Dilute Acid for Regeneration	CC	Acid concentration
Waste Regeneration Neutralizing Tank and Acid and Caustic Waste Sump Return	CpH	pH of nonradioactive regeneration wastes, prior to discharge
<u>FUEL POOL CLEANUP SYSTEM</u>		
Demineralizer Inlet	CC, Grab	Indication of fuel pool water quality
Demineralizer Outlet	CC, Grab	Demineralizer efficiency
<u>AUXILIARY BOILER SYSTEM</u>		
Auxiliary Boiler Blowdown	Grab	Water quality, boiler performance
Auxiliary Boiler Feed Pumps Discharge	Grab	Water quality data
Auxiliary Boiler Condensate	Grab	Condensate quality
Auxiliary Boiler Steam	Grab	Boiler performance data
Hot Water for Heating	Grab	Water quality data
<u>COOLING WATER SYSTEMS</u>		
Turbine Building Closed Loop Cooling Water Heat Exchanger Discharge	CC	Cooling water quality

Note: CC - Continuous Conductivity Monitoring
 Grab - Grab Sample
 IX - Intermittent Silica Monitoring
 CpH - Continuous pH Monitoring

TABLE 9.5.9-1

STORAGE OF GAS UNDER PRESSURE (6)

<u>Gas</u>	<u>Container Design (PSIG)</u>	<u>Oper. Press. (PSIG)</u>	<u>Max Press. (1) (PSIG)</u>	<u>No. Contain.</u>	<u>Est. Tank Volume (ft3) each</u>	<u>Max Energy Release(2) if Ruptured (ft lbx10⁶)</u>		<u>Location</u>
						<u>One Tank</u>	<u>All Tanks</u>	
<u>Carbon Dioxide(3)</u>								
Fire Protection	363 ⁽⁴⁾	300	341	1	320	230.8	230.8	Yard
<u>Halon 1301</u>								
Fire Protection Reactor Shutdown	2650	600@70°F	1250	2	0.74	.002	.004	Reactor Bldg.
Fire Protection TSC Computers (3)								
Above Floor	600	360	600	2	4	.25(5)	.50(5)	TSC Bldg.
Below Floor	1000	600	1000	2	1.8	.14(5)	.28(5)	
<u>Air</u>								
Instrument & Service Receivers	150	125	145	3	415	11.8	35.4	Turbine Bldg.
Control Room Breathable								
Air System A	10000	2400	4000	10	1.73	1.17	11.67	Turbine Bldg.
B	5000	2216	3693	5	0.30	0.16	0.80	
SRV Accumulator	145	95	115	11	0.205	0.005	0.051	Primary Cont.

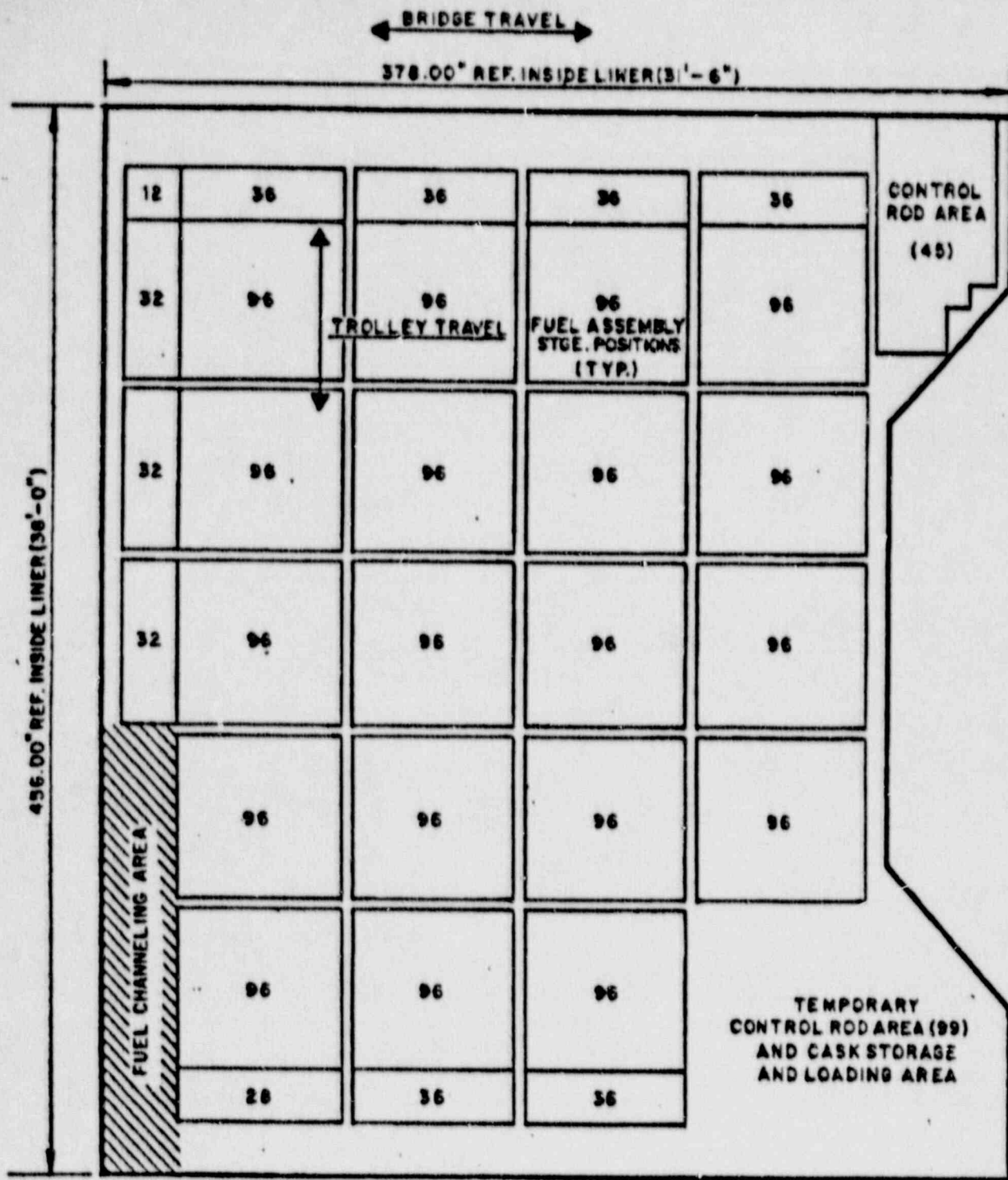
TABLE 9.5.9-1 (Cont'd)

STORAGE OF GAS UNDER PRESSURE

<u>Gas</u>	<u>Container Design (PSIG)</u>	<u>Oper. Press. (PSIG)</u>	<u>Max Press. (1) (PSIG)</u>	<u>No. Contain.</u>	<u>Est. Tank Volume (ft³) each</u>	<u>Max Energy Release (2) if Ruptured (ft lbx10⁶)</u>		<u>Location</u>
						<u>One Tank</u>	<u>All Tanks</u>	
<u>Nitrogen</u>								
Hot Water Heating	10000	2600	4000	2	1.73	1.04	2.08	Turbine Bldg.
Glycol Heating	10000	2600	4000	2	1.73	1.04	2.08	Turbine Bldg.
<u>Propane</u>								
Aux. Blr Ignition	480	204	220	2	85	1.25	2.51	Yard

Notes

- (1) Safety valve set point.
- (2) Reversible adiabatic expansion from maximum container pressure.
- (3) Stored as liquified gas.
- (4) Maximum working pressure.
- (5) Max energy release calculated on basis of 5/3 operating press being equal to max press. This does not significantly alter the max. energy release number.
- (6) Variable quantity and type of welding gas tanks not listed. Type and quantity is variable based on maintenance requirements.



NOTES: 2176 FUEL STORAGE POSITIONS
144 CONTROL ROD STORAGE POSITIONS

FIGURE 9.1.2-1
HIGH DENSITY RACK PLACEMENT
IN SPENT FUEL POOL
SHOREHAM NUCLEAR POWER STATION
DEFUELED SAFETY ANALYSIS REPORT

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CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

10.1 STEAM AND POWER CONVERSION SYSTEM

The purpose for which the steam and power conversion system was built no longer exists. The components of this system as described in the USAR will not be required in the defueled condition.

10.2 TURBINE GENERATOR

The purpose for which the turbine-generator system was built no longer exists. The components of this system as described in the USAR will not be required in the defueled condition.

There is no longer a concern for turbine generated missiles.

10.3 MAIN STEAM SUPPLY SYSTEM

The purpose for which the main steam supply system was built no longer exists. The components of this system as described in the USAR will not be required in the defueled condition.

In the defueled condition the main steam system will not serve any safety-related function and therefore will be reclassified as Q.A. Category IIA.

10.4 OTHER FEATURES OF STEAM & POWER CONVERSION SYSTEM

10.4.1 Condenser

The purpose for which the condenser was built no longer exists. The components of this system as described in the USAR will not be used in the defueled condition.

10.4.2 Main Condenser Air Removal System

The purpose for which the main condenser air removal system was built no longer exists. The components of this system as described in the USAR are not required in the defueled condition.

10.4.3 Steam Seal System

The purpose for which the steam seal system was built no longer exists. The components of this system as described in the USAR are not required in the defueled condition.

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10.4.4 Turbine Bypass System

The purpose for which the turbine bypass system was built no longer exists. The components of this system as described in the USAR are not required in the defueled condition.

The portion of the bypass system upstream of the bypass valves was built to ASME III cc2 criteria. As the function of the bypass system no longer exists in the defueled condition, the bypass system is reclassified Q.A. Category IIA.

10.4.5 Circulating Water System

The purpose for which the circulating water system was built no longer exists. The components of this system as described in the USAR will not be required in the defueled condition. The only exception is that the circulating water discharge system will be used to provide dilution capacity for elimination of liquid radwaste and SPDES limits on chlorine and suspended solids to the Long Island Sound.

10.4.6 Condensate Demineralizer System

Since there is no fuel in the Reactor and no Reactor steam produced, there is no need for the Condensate Demineralizer System. This system will be protected. However, the Acid and Caustic Storage Tanks (1N52-TK-035 and -TK-036) will remain operable to provide regeneration chemicals for the continued operation of the Demineralizer and Makeup Water System (P31). The Chemical Waste Sump (1N52-TK-113) will remain operable as a pathway for further treatment of non-radioactive regenerant waste from the Demineralizer and Makeup Water System.

10.4.7 Condensate and Feedwater System

The purpose for which the condensate and feedwater system was built no longer exists. The components of this system as described in the USAR will not be required in the defueled condition.

Piping built and designed to ASME III ccl is considered Q.A. Category IIA while in the defueled condition. Inservice inspection according to ASME XI need not be performed while in the defueled condition.

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CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

11.1 RADIATION SOURCE TERMS

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged as it is used to develop the basic design criteria of the plant. However, the actual source terms in the plant's present defueled condition are as follows:

a. Liquid Radioactivity Sources

As of August 1989, since all SNPS' fuel had been placed in the spent fuel pool, there were no liquid sources with nuclide concentrations greater than the Lower Limit of Detection (LLD), outside of radwaste streams. It must be recognized that in the future some concentrations greater than LLD will be seen (e.g., as sludge at the bottom of sumps and the suppression pool are processed to Radwaste). However, these should be minor and temporary occurrences. Sources related to the decontamination and decommissioning should also be minor, as the degree of overall plant contamination is low. These liquid sources would be dealt with in accordance with the Liquid Radwaste, ALARA, and Health Physics programs as discussed in DSAR Sections 11.2, 12.1, and 12.5, respectively.

Isotopic concentrations above the LLD levels in the Radwaste System as of 6/30/89 are indicated in Table 11.1-1, from References 2, 3 and 4.

b. Gaseous Sources

There are no detectable gaseous sources at SNPS, either present or anticipated. This statement is supported by the fact that the Semi-Annual Radiological Effluent Release Report for the first and second quarter 1989 (Reference 1) indicates there were no detectable releases during the six-month period, either from the offgas system or the various building exhaust systems.

c. Activated Materials Sources

It is expected that materials which were located in the reactor vessel during low power testing (eg, control rods, TIPS, IRMs, and the like) have been activated to some extent. With the exception of some portions of the liquid radwaste system (10 mrem/hr maximum), dose rates outside of plant systems are very low, less than 0.5 mrem/hr. These low dose rates are indicative of a low deposition of sources within plant systems.

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There may be a minor amount of activation source material deposited within plant systems. However, the level of this activity, and indeed of the activation products within the reactor vessel itself, are not considered significant compared to the spent fuel sources described in Section 12.2.

REFERENCES

General

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

1. "Semiannual Radioactive Effluent Release Report - First and Second Quarter 1989", transmitted by letter SNRC-1619, 8/29/89.
2. "SNPS HIC Package Data for November/December 1988", 6/5/89, Memorandum L. Hall to T. Gillett.
3. "Transmittal of Data for Dose Projection", 5/16/89. Memorandum, P. Lynch to M. Beer.
4. Gamma Spectrometer Scan of Floor Drain Collector Tanks, Waste Collector Tanks, and Recovery Sample Tanks, 6/15/89, Memorandum M. Ma to T. Gillett.

11.2 RADIOACTIVE LIQUID WASTE SYSTEM

11.2.1 Design Objectives

The Radioactive Liquid Waste System is described in the USAR. With the Reactor defueled and the fuel assemblies stored in the Fuel Pool, the sources, quantity and activity of the radioactive waste are greatly diminished. Certain portions of the Radioactive Liquid Waste System are not required.

11.2.2 System Descriptions

The estimated influent to the radwaste system is reduced from 25,000 gpd to approximately 2,000 gpd.

The regenerant chemical subsystem is no longer required, except for the Chemical Waste Sump, the Regenerant Liquid and Evaporator Feed Tanks and Pumps.

The Waste Evaporator portion of the Floor Drain Subsystem is not required.

The Phase Separator System serving the RWCU System is not required unless RWCU is required if reactor is layed up wet.

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11.2.2.1 Summary

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 (unless the reactor is layed up wet).

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.

11.2.3.4 Liquid Radwaste Equipment Quality Group Classification

This Section remains as presented in the USAR.

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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.

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).

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11.2.3.4.4 Results and Consequences

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

<u>Source</u>	<u>Dose, millirem</u>		
	<u>Whole body Gamma*</u>	<u>Beta Skin</u>	<u>Maximum Organ**</u>
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	<u>1.8E-05</u>	<u>2.8E-06</u>	<u>1.3E-03</u>

* 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.

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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 ($7.83\text{E-}08$ uCi/cc of Co-60, from DSAR Table 11.1-1), the estimated release is:

$1.5\text{E-}03$ 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

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dose is conservatively obtained by assuming each batch liquid release contains the maximum batch activity concentration of $7.83\text{E-}08$ 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:

1. meet either 10CFR20 or 10CFR50 Appendix I limits or
2. ensure plant operability or availability.

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.

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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 secondary containment.

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.

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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.

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

Tables 11.5.1-1B 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

For the defueled condition, this receives backwashed resin and filter media from the Radwaste Demineralizer and the Fuel Pool Cleanup Demineralizer and Filters (and Phase Separators if the RWCU System remains operable). (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 HIC's are then dewatered by portable air-operated diaphragm pumps which draw suction from specially designed piping internals in the HIC's. When convenient, HIC's may be dewatered while in the fill cubicles.

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A 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.

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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

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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 were: 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 sampling locations and/or range of analyses are discontinued. The current Post-Defuel REMP program is outlined in Tables 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 1988), SNPS has not generated radioisotopes since the last 5.0% power test, completed on June 6, 1987. Comparisons between the above two phases of

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REMP were documented in each Semiannual Radiological Effluent Release 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, virtually 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 normal 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 the reduced surveillance program 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:

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

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 (4.0 mCi and 1560.0 Ci, respectively), 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) for which REMP is applicable are:

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

SNPS' Post-Defuel REMP Surveillance Program Outline

- | | | |
|----|---|--|
| 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, Noble
Gas and Gross Beta | - Retain, Kr-85, particu-
lates and solubles still
exist. |

Quarterly Surveillance Frequency

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- | | |
|----------------------------------|---|
| 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, 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 to the requirements as delineated in DSAR Chapter 16, developed after 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.

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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

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results 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 detectable 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 radioactive 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 preoperational and operational REMP data from 1983 through 1988.

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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

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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.

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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 samples (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 time the sample is taken.

Air samples are collected quarterly at St. Joseph's Villa and analyzed for noble gases (Krypton-85). The samples are collected using a modified low pressure air compressor. An interim holding tank is evacuated to 20 in. Hg. Outside air is drawn into the interim holder and then transferred to a sample tank for transport to the laboratory for analysis.

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 (CaSO₄:Dy) TLDs, each containing four separate readout areas. The TLDs are annealed by LILCO 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.

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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 Technical Specifications
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"

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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):

<u>Isotope</u>	<u>Activity Concentration, uCi/cc</u>	<u>% of Activity</u>
*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):

<u>Isotope</u>	<u>Activity Concentration, uCi/cc</u>	<u>% of Activity</u>
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

		(----- Operational REMP -----)				(--- Preoperational REMP ---)	
<u>SAMPLE TYPE</u>	<u>Unit/Isotope</u>	<u>1988</u>	<u>1987</u>	<u>1986</u>	<u>1985</u>	<u>1984</u>	<u>1983</u>
Potable Water	pCi/l (H-3)	240 - 410	140 - 450	130 - 420	150 - 290	120 - 540	70 - 220
Game	pCi/Kg (Cs-137)	76.7 - 9270	35.1 - 6490	54 - 3230	992 - 4330	641 - 5340	34.0 - 6310
Direct	mrem/ Mnth	2.3 - 5.2	2.8 - 6.9	1.9 - 5.7	3.0 - 6.2	2.7 - 6.9	2.3 - 5.7
Radiation	Std Mnth Qtr	2.7 - 4.8	2.9 - 5.0	2.9 - 4.9	2.8 - 5.5	3.1 - 6.2	2.8 - 5.4
Air:Gross Beta	[x1.0E-3]	5.0 - 44.0	4.0 - 32.0	5.0 - 360	6 - 47	4.2 - 61.	5 - 54
Particulate Sr-90	pCi/m ³ 3 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/m ³ 3 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-137)	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	0.98 - 13.0	0.86 - 4.60	0.69 - 5.3	0.9 - 7.7
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	NA
Food	pCi/Kg (I-131)	LT 4.0	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/m ³ (Kr-85)	28 - 44	24 - 45	21 - 48	24 - 40	30	18 - 49
	pCi/m ³ (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.

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SHOREHAM DSAR TABLE 11.6.3-1

Post-Operational Radiological Environmental Monitoring Program (REMP)

<u>Media</u>	<u>Sampling Locations</u>	<u>Sampling Frequency</u>	<u>Analysis</u>
Direct (1) Radiation	1S1, 2A2, 3S1, 4S1, 5S2, 6S2, 7A2, 8A3, 9S1, 10A1, 11A1, 12A1, 13S3, 14S2, 15S1, 16S2, *5E2, *6E1	Quarterly	Gamma Exposure
Fish and (2) Invertebrates	3C1, 14C1, *13G2	Semi-annually or when in season	Gamma-isotopic Sr-89/90
Fruits, (3) and Vegetables	8B1, 6B21, *12H1	At time of Annual Harvest	Gamma-isotopic
Airborne (4) Particulates	6S2, 2A2, 3S1, 7B1, *11G1	Quarterly	Gross-Beta Gamma-isotopic Sr-89/90
Milk (5)	13B1, *10F1, or *8G2	Quarterly. During Pasture Season, Monthly	Gamma-isotopic Sr-89/90
Surface Water	3C1 or 14C1, and *13G2	Semiannual Grab Sample	Gamma-isotopic H-3, Sr-89/90
Noble Gases	14S2	Monthly	Krypton-85

(*) Designates Control Locations

- (1) Eighteen monitoring stations (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 in the least prevalent wind direction.
- (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.

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SHOREHAM DSAR TABLE 11.6.3-2
REMP SAMPLING LOCATIONS

<u>DESIGNATION</u>	<u>LOCATION</u>
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

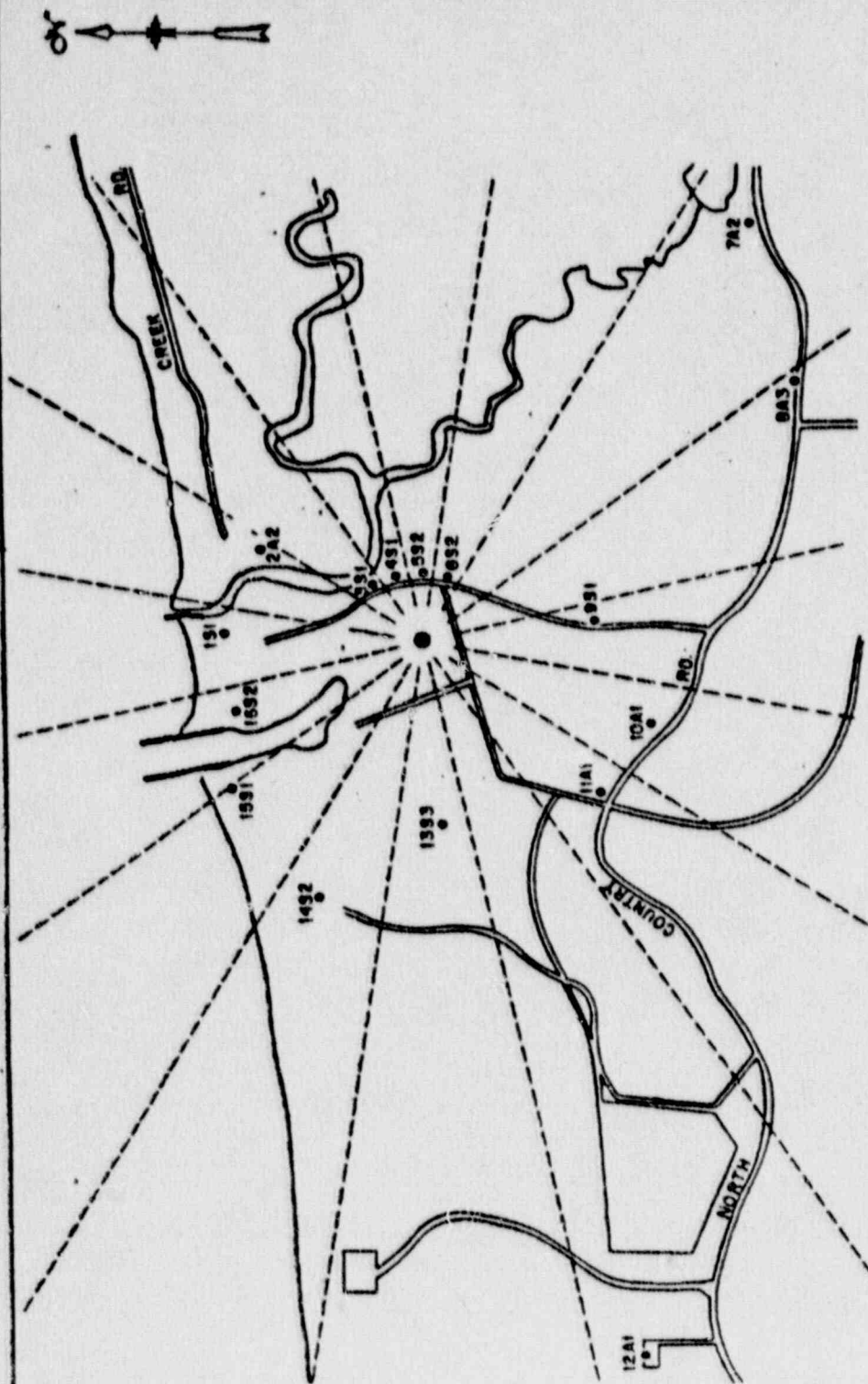


FIGURE 11.8.3-1
ON SITE SAMPLING LOCATIONS
RADIOLOGICAL ENVIRONMENTAL
MONITORING PROGRAM
SHOREHAM NUCLEAR POWER STATION
DEFUELED SAFETY ANALYSIS REPORT

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CHAPTER 12

RADIATION PROTECTION

12.1 ASSURING THAT OPERATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

The Shoreham ALARA Program, the intent of which is to maintain operational radiation exposures (OREs) to levels as low as is reasonably achievable (ALARA), is described in the USAR, Section 12.1. The program is applicable in its entirety to Shoreham in a defueled condition, with the following exceptions:

- A) Within the Nuclear Engineering Support Organization, the radiation protection function is found within the Nuclear Analysis function, as described in Chapter 13 of the DSAR.

(Reference USAR Section 12.1.1.3.2, Review of Modification of Operations. The basis of this change is a reorganization of the Nuclear Engineering Support Organization.)

- B) Several of the items listed under Section 12.1.2, Design Considerations, are no longer applicable or have been revised. Specifically,

charcoal has been removed from the Radwaste vent filters (considerations 10 & 12),

the Radiation Monitoring System is now discussed in DSAR Section 12.3 (consideration 11),

demin water hose stations are located on the radwaste building floor (consideration 7),

shielded radwaste shipping casks and remote handling techniques are not generally used (considerations 15 and 16),

and the discussion of gaseous radwaste sources in consideration 15 is no longer relevant, as the offgas system is shut down.

(Reference USAR Section 12.1.2, Design Considerations. The justification for lack of charcoal is the fact that as per DSAR Sections 11.1 and 12.2, Shoreham does not possess a meaningful quantity of radioiodines. The low levels of radioactivity in the solid radwaste do not justify using shielded casks and remote handling.)

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- C) All visitors within the Protected Area are escorted by qualified personnel. Those visitors requiring access to the radiologically controlled area (RCA) are given an appropriately abbreviated indoctrination in protection against radiation, prior to their entry into the RCA.

(Reference USAR Section 12.1.3.1.1, Use of Individual Personnel Monitoring Devices. The basis of this change is a more stringent application of security requirements for visitors to Shoreham).

- D) With the generally very low dose rates associated with the plant's defueled condition, there is no longer a requirement to have all personnel (permanent and temporary) equipped with approved dosimetry devices upon their entry to the radiologically controlled area (see Section 12.5.2.1, Access Control. Rather, only individuals working on a Radiation Work Permit (RWP) may have approved dosimetry issued. That the requirements of 10CFR20.202 are met by this approach will be assured by the ongoing station radiation surveillance program (as described in USAR Section 12.5.3.1), as well as the posting of thermoluminescent dosimeters (TLDs) in general access areas of the RCA.

(Reference USAR Section 12.1.3.1.1, Use of Individual Personnel Monitoring Devices. This change is justified by the low dose rates seen presently at Shoreham, and by the very low historical man-rem data in Section 12.5 of the DSAR.)

It should be noted that the Shoreham station's original physical design for radiation protection (e.g., shield walls, penetrations, sample stations, etc.) remains generally unchanged from that described in the USAR, Sections 12.1 and 12.3. This is despite the fact that the actual source strengths and unshielded dose rates do not necessitate the degree of protection afforded. The physical design is based upon the presumption of plant operations, with the associated source terms and unshielded dose rates as described in the USAR. Although they will not generally be needed, operational considerations described in the USAR (e.g, the precautions for high dose rate jobs -- in excess of 100 mrem/hr) will be maintained.

12.2 RADIATION SOURCES

12.2.1 Contained Sources

Fuel Sources

The Shoreham reactor core has undergone three periods of low power (0-5%) testing over the past four years. The low power tests are summarized below:

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Test Period	Duration	Specific Burnup	Power
		MWD/MT	Range %
7/77-10/77/85	93 days	27.8	0.0 - 3.3
8/5-8/30/86	26 days	13.8	0.0 - 4.0
5/26-6/6/87	12 days	6.7	0.0 - 3.5
	Total	48.3	

The detailed profiles of the above three low power test periods have been input to the ORIGEN2 (Reference 1) burnup code, along with the physical characteristics of the reactor fuel and bundle structural elements. Results of this analysis (Reference 2) are given in Table 12.2-1. The activities correspond to two years decay after the last burnup period, and reflect total core inventories for those isotopes with greater than 10 curies.

That the source strengths given in Table 12.2.-1 are reasonable is evidenced by measurements taken during the defueling of the reactor. Dose rate measurements were taken at one foot from a number of spent fuel bundles, and the maximum values from each bundle were tabulated. Dose rates at one foot (as a function of bundle burnup) were calculated from the source terms presented in Table 12.2.-1, using the point kernel code QADMOD (Reference 5), and the results compared to the measured dose rates. Results are given in Table 12.2.-2. That the calculated and measured bundle maximum dose rates agree within about 10% on average gives assurance that the calculated source terms in Table 12.2-1 are reasonably accurate.

As can be seen from the Table 12.2-1 only long-lived isotopes remain from the original actinides and fission/ activation products created, along with their equilibrium daughters. By far the most radiologically significant, from a gamma dose rate standpoint, are the Cs-137/Ba-137m pair; about 80% of the whole body dose rate from a spent fuel bundle is due to the Ba-137m photon (Reference 3). For dose assessment of accidental gaseous releases (e.g., a postulated fuel handling accident), only Kr-85 is meaningful (Reference 4).

12.2.2 Airborne Radioactive Material Sources

The statements below apply when systems are closed up. When potentially contaminated systems are opened, the RWP controls, as stipulated in DSAR Section 12.5, will minimize airborne sources.

Reactor Building

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

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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.

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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

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normally have access and where there is a possibility 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 exist. 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. 1297-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. 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 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.

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

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<u>Time Period</u>	<u>Dose, man-rem</u>
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

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 in the defueled condition. Doses will of course 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 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.

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(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) The monitoring station where personnel exiting from the controlled area frisk themselves will no longer be in general use, unless justified by the overall workload in the controlled area. Similarly, the frisking station between the Turbine Building and the Control Room will be removed. In general, frisking will only be done at the exit of the specific jobsite, as controlled by the RWP.

(Reference USAR Section 12.5.2.1, Location of Equipment, Instrumentation and Facilities, Section 12.5.3.3.1, Access Control, and Figure 12.5.3-1. The basis of this change is the fact that there is no significant contamination within the controlled area, except perhaps at specific jobsites (controlled by the RWP). This fact is confirmed by the ongoing surveillance program.)

- 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 the reduced surveillance requirements and number of plant personnel.)

- 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:

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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 Corporate ALARA Review Committee (CARC) now administratively reports to the Assistant Vice President, Nuclear Operations.

(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. For justification, see DSAR Section 12.1.3.1.1.)

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, the procedures and commitments remain in place in the extremely unlikely event that they should be required.

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TABLE 12.2-1

Fuel Source Terms

<u>ISOTOPE</u>	<u>CURIES</u>	<u>HALF-LIFE</u>
H-3	1.77E+02	1.23E+01 years
Mn-54	3.36E+01	3.13E+02 days
Fe-55	8.06E+02	2.70E+00 years
Co-60	5.64E+02	5.27E+00 years
Ni-63	4.28E+01	1.00E+02 years
Kr-85	1.56E+03	1.07E+01 years
Sr-89	1.54E+01	5.05E+01 days
Sr-90	1.37E+04	2.86E+01 years
Y-90	1.37E+04	6.41E+01 hours
Y-91	6.81E+01	5.85E+01 days
Zr-95	1.48E+02	6.40E+01 days
Nb-95	3.49E+02	3.51E+01 days
Ru-106	5.98E+03	3.68E+02 days
Rh-106	5.98E+03	2.99E+01 seconds
Sn-119m	3.30E+02	2.93E+02 days
Sb-125	1.45E+03	2.77E+00 years
Te-125m	3.53E+02	5.80E+01 days
Te-127	1.49E+01	9.35E+00 hours
Te-127m	1.52E+01	1.09E+02 days
Cs-134	1.33E+02	2.06E+00 years
Cs-137	1.48E+04	3.02E+01 years
Ba-137m	1.40E+04	2.55E+00 minutes
Ce-144	3.55E+04	2.84E+02 days
Pr-144	3.55E+04	1.73E+01 minutes
Pr-144m	4.26E+02	7.20E+00 minutes
Pm-147	2.95E+04	2.62E+00 years
Sm-151	3.60E+02	9.00E+01 years
Eu-154	1.18E+01	8.80E+00 years
Eu-155	4.47E+01	4.96E+00 years
U-234	1.02E+02	2.45E+05 years
Th-234	3.38E+01	2.41E+01 days
Pa-234m	3.38E+01	1.17E+00 minutes
U-238	3.38E+01	4.47E+09 years
Pu-239	2.77E+02	2.41E+04 years
Pu-241	5.58E+01	1.44E+01 years

Total 1.76E+05

Note: Only isotopes with activity greater than 10 curies are listed.

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TABLE 12.2-2

Comparison of Measured (maximum) vs.
Calculated Spent Fuel Bundle Dose Rates*

Date	Cell Number	Burnup gwd/st**	Dose Rates, rem/hr		Measured/ Calculated
			Measured	Calculated	
7/14/89	07-46	0.0045	0.4	0.4	1.00
7/19	01-32	0.0047	0.7	0.4	1.75
7/21	45-34	0.0385	5.1	3.5	1.46
7/24	25-48	0.0421	5.5	3.9	1.41
7/26	45-10	0.0126	1.6	1.2	1.33
7/26	47-14	0.0082	0.9	0.8	1.13
7/26	15-44	0.0398	3.2	3.7	0.86
7/28	43-30	0.0547	6.0	6.0	1.00
7/29	21-44	0.0516	5.3	5.6	0.95
7/31	09-34	0.0514	3.8	5.6	0.68
8/01	39-16	0.0607	6.0	6.6	0.91
8/01	39-32	0.0624	6.0	6.8	0.88
average					1.11

* in water at 12 inches from fuel bundle

** gwd/st = gigawatt-days/short ton

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TABLE 12.3.4A

PROCESS AND EFFLUENT MONITORS

PM 13	Liquid RW Discharge
PM 21	Low Range Station Vent Monitor
PM 29 Gas	Reactor Building Vent
PM 30 Part	Reactor Building Vent
PM 41 Part	Station Vent
PM 42 Gas	Station Vent
PM 55 Gas	Radwaste Vent
PM 56 Part	Radwaste Vent
PM 79	Saltwater Drain Tank

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TABLE 12.3.4B

DATA FOR EFFLUENT AND PROCESS RADIATION MONITORS

<u>Monitor</u> ⁽¹⁾	<u>Monitoring Function</u>	<u>Location</u>	<u>Type of Monitor</u>	<u>Range (2)</u> <u>Sensitivity</u>	<u>Action</u> <u>Taken on Alarm</u>
<u>Normal Operation</u>					
Station Ventilation Exhaust	Monitor final release for all low level gaseous effluents to the main plant vent airstream	Downstream of the last point of activity	Offline gas Particulate	10^{-6} uCi/cc gases (Kr-85)	Investigate and correct cause of high activity release rate exceeding instantaneous technical specification dose rate limit.
Liquid Radwaste Effluent	Monitor activity release rate during planned liquid waste discharge periods	Effluent pipe prior to discharge into the circulating water system	Offline Liquid	10^{-6} uCi/cc (Cs-137)	Automatic closure of liquid waste discharge valve exceeding 10CFR20 limits to unrestricted areas.
Reactor Building Salt Water Drain Tank	Monitor activity release of service water drained during maintenance	Downstream of the collection tank	Offline Liquid	10^{-6} uCi/cc (Cs-137)	Investigate and correct cause of high activity release rate exceeding 10CFR20 limits to unrestricted areas
Reactor Building Ventilation	Monitor all low level gaseous effluents in the reactor bldg ventilation duct	Effluent duct prior to discharge into the station vent	Offline gas Particulate	10^{-6} uCi/cc gases (Kr-85)	Investigate and correct cause of high activity release rate
Radwaste Building Ventilation	Monitor all low level gaseous effluents in the radwaste bldg. ventilation duct.	Effluent duct prior to discharge into the station vent	Offline gas Particulate	10^{-6} uCi/cc gases (Kr-85)	Investigate and correct cause of high activity release rate.

TABLE 12.3.4B

DATA FOR EFFLUENT AND PROCESS RADIATION MONITORS

<u>Monitor</u> ⁽¹⁾	<u>Monitoring Function</u>	<u>Location</u>	<u>Type of Monitor</u>	<u>Range (2)</u> <u>Sensitivity</u>	<u>Action</u> <u>Taken on Alarm</u>
<u>Post-Accident</u>					
Station Ventilation Exhaust (Low Range)	Monitor final release for all gaseous effluents to the main plant vent air- stream during an accident	Downstream of the last point of activity	Offline gas Particulate	10^{-6} uCi/cc N/A gases (Kr-85)	

(1) One unit unless specified otherwise.

(2) Dynamic range is a minimum of 4 decades above sensitivity.

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TABLE 12.3.4C

AREA MONITORSArea Monitor

		Alert/High Setpoints (mrem/hr)	Range (mrem/hr)	<u>Location</u>
1D21-RE-001	Floor Drain Sump Tank	5/100	0.1-1000	Reactor Bldg. el 8-0
-010	Fuel Pool Cleanup Pumps	5/100	0.1-1000	Reactor Bldg. el 112-9
-012	Fuel Pool Equipment Area	5/100	0.1-1000	Reactor Bldg. el 150-9
-013	Contaminated Equip. Storage	5/100	0.1-1000	Reactor Bldg. el 150-9
-014	Fuel Storage Pool	5/10	0.1-1000	Reactor Bldg. el 175-9
-015	Reactor Head Insulation Storage	5/100	0.1-1000	Reactor Bldg. el 175-9
-022	Chemistry Laboratory Mezzanine	1/5	0.01-100	Heater Bay el 31-0
-024	Radwaste Bldg. Decontamination Area	5/100	0.1-1000	Radwaste Bldg. el 15-6
-026	Storage Vaults for Containers	10/100	1.0-10 ⁴	Radwaste Bldg. el 15-6
-027	Sample Room	20/100	0.1-1000	Radwaste Bldg. el 37-6
-028	Floor Drain Filter Area	5/100	0.1-1000	Radwaste Bldg. el 37-7
-029	Radwaste Filter and Demineralizer Area	5/100	0.1-1000	Radwaste Bldg. el 37-6
-032	Radwaste Bldg. Demineralizer Area	20/100	0.1-10 ⁴	Radwaste Bldg. el 15-6
-033	Radwaste Bldg. Hoist Area	5/100	0.1-10 ⁴	Radwaste Bldg. el 50-6
-035	Equipment Drain Tank Area	5/100	0.1-10 ⁵	Reactor Bldg. el 8-0
-036	TIP Drive Room	5/100	1.0-10 ⁴	Reactor Bldg. el 78-7
-037	TIP Drive Area	5/100	0.1-10 ⁴	Reactor Bldg. el 78-7
-038	New Fuel Storage Area	2/10	0.1-10 ⁴	Reactor Bldg. el 175-9
-042	Radwaste Bldg. Pump Area	5/100	0.1-10 ⁴	Radwaste Bldg. el 15-6

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Table 12.4-1RADIATION ZONES
(Original plant design basis)

<u>Zone Designation</u>	<u>Zone Description</u>	<u>Maximum Allowable Dose Rate (mrem/hr)</u>	<u>Estimated Occupancy Time (hr/wk)</u>
I	Unrestricted Area - Continuous Access	less than 0.2	Unlimited
II	Unrestricted Area - Periodic Access	less than 2	50
III	Restricted Area - Controlled Frequent Access	less than 5	20
IV	Radiation Area - Controlled Infrequent Access	less than 20	5
V	Radiation Area - Controlled Infrequent Access	less than 100	1
VI	High Radiation Area - Not Normally Accessible	greater than 100	-

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CHAPTER 13

CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

The description contained under this heading in the latest revision of the Shoreham USAR changed to be as described below.

13.1.1 Corporate Organization

- A) The Office of Nuclear organization is shown on DSAR Figure 13.1.1-2. The Assistant Vice President, Nuclear Operations has assumed all the responsibilities of the Vice President, Nuclear Operations. The position of Vice-president, Nuclear Operations has been eliminated.
- B) The Manager, Nuclear Quality Assurance Department (NQAD), reports directly to the Assistant Vice President, Nuclear Operations but has maintained direct access to the President of the Company as he deems necessary.
- C) The Nuclear Engineering Department has been reorganized to reflect a reduced level of activity in the defueled condition. Now entitled the Engineering and Technical Support Organization, it reports to the Manager, Engineering & Administrative Support. It no longer includes an Engineering Assurance function.
- D) The Safety Engineering and Reliability organization within the NQAD is eliminated. This includes the ISEG and Reliability Sections.
- E) The Director, Office of Training and the Manager, Nuclear Emergency Preparedness Division report to the Vice President, Corporate Services.
- F) DSAR Figure 13.1.1-1 shows revised direction of executive responsibility.

13.1.2 Engineering And Administrative Support Organization

The Engineering and Administrative Support Organization consists of the following organizational units:

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Licensing, Nuclear Contracts & Material Controls, Engineering & Technical Support and Nuclear Financial Services. These units have as many staff specialists as required to support Shoreham.

Engineering & Administrative Support personnel provide expertise for supplementary support functions such as licensing and regulatory activities, including assessing evolving regulations, managing all nuclear litigation and evaluating regulatory documents for impact on plant design. Other responsibilities cover cost control, estimating, budget and cost administration for the Office of Nuclear, nuclear records management and administration of site clerical administrative personnel. The units are also responsible for nuclear contract development and administration, administration of site warehouses, spare parts and inventory control.

The Engineering & Technical Support Organization consists of the following units:

Plant Support, Engineering Administration & Services, Modification Support and Nuclear Analysis. Responsibilities include, (1) Systems, Mechanical & I&C Engineering, (2) Procurement Support, (3) General administrative support for procedures, training and document control, (4) Nuclear Analysis Support for the following technical functions: Radiological Engineering & Health Physics, Radiological Monitoring Program (REMP), and Engineering and Nuclear Fuels Engineering, (5) Modification Engineering Unit which requests and coordinates implementation of station modifications, coordinates post-modification retesting, and return to service.

The Engineering & Technical Support Organization also coordinates work performed by off-site support Engineering which includes Corporate Engineering and outside contractors. Outside contractors include the original plant architect - engineer and the NSSS Vendor.

13.1.3 Operating Organization

The Shoreham Nuclear Power Station Organization, as shown in Figure 13.1.1-2, consists of 4 Divisions:

Operations, Maintenance, Radiological Controls & Security

13.1.3.1 Operations

Operations is responsible for complying with the rules and regulations of the governing regulatory agencies and the monitoring of the station performance. It is composed of Operations, Operations Staff, System Engineering, Compliance, and Work Planning and Scheduling.

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Operation activities of this unit primarily consist of the routine operation of the station systems and equipment.

The Systems Engineering Unit is part of Operations and its responsibilities include rapidly providing specialized, operationally oriented technical expertise in station systems and equipment. Systems Engineering also performs support functions for other operating organizations as appropriate.

The Operations Staff Unit provides station administrative support and assurance that the station is in compliance with the requirements of the Operating License.

The Operational Compliance Unit implements the station surveillance programs and reviews surveillance activities to ensure compliance with the station's Technical Specifications.

The Work Planning and Scheduling Unit performs planning and scheduling associated with plant activities.

13.1.3.2 Maintenance

Maintenance is responsible for maintaining the Station's mechanical, electrical, instrumentation, and computer systems. It is composed of the Instrument and Controls Maintenance, and Fire Protection and Safety.

The Instrument and Control Unit is responsible for the calibration, maintenance, and testing of instruments and control systems in the nuclear power station.

The Maintenance Unit has a staff experienced in mechanical and electrical maintenance of large steam-electric generating stations. Additionally, it can be supplemented with additional competent maintenance personnel from other LILCO power stations or organizations, or outside contractors, as may be required.

Fire Protection and Safety is responsible for implementing the Plant Fire Protection Program and for coordinating the activities of the Fire Brigade and the Site Safety Committee. The Supervisor holds the position as Fire Protection Program Manager responsible for maintaining compliance with applicable Federal, State and local government regulations regarding station fire protection and personnel safety.

13.1.3.3 Radiological Controls

Radiological Controls is responsible for the protection of the public, station personnel, and the environment from the effects of exposure to radiation. It maintains the radiation doses of station personnel and the public as low as reasonably achievable

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(ALARA) and assures proper handling, processing, and disposal of radioactive materials. Radiological Controls consists of Health Physics and Radiochemistry.

The Health Physics Unit responsibilities include the preparation of Radiation Work Permits, performance of radiological surveillances, maintenance of personnel exposure records, calibration and maintenance of fixed and portable radiation detection instrumentation, and proper disposal of radioactive material, and proper disposal of radioactive material.

The Radiochemistry Unit is responsible for activities such as detection and control of environmental releases, assessment of radiation doses to the public, and station chemical and radiochemical activities.

13.1.3.4 Security

The responsibilities of Security are described in the Security Plan.

13.1.4 Qualification Requirements for Station Personnel

This section is revised from that in the USAR.

All responsible station personnel, both supervisory and non-supervisory meet the requirements of ANSI 18.1-1971.

13.2 TRAINING PROGRAM

13.2.1 Program Description

The purpose of the accreditation program is to assist INPO member utilities in maintaining training programs that produce well-qualified, competent personnel to operate the nation's nuclear power plants (INPO 88-001).

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 Office of Training has non-nuclear training programs available, 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 "Training and Qualification Program".

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Operators: Operators will be trained 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 licensed operator training program developed for nuclear operations.

Equipment Operator: Field operators will be 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 in accordance with the Control Technician training program developed for power plant technicians.

Mechanics/Electricians: LILCO 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 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.

13.3 EMERGENCY PLANNING

The emergency plan for the Shoreham Nuclear Power Station is submitted as a separate document entitled, "Defueled Emergency Preparedness Plan".

13.4 REVIEW AND AUDIT

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged except as described below.

13.4.1 Review and Audit - Construction

No Change.

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13.4.2 Review and Audit - Test and Operation

- A) In 13.4.2.1, change the ROC membership, alternates and quorum requirements as follows:

Membership; A chairman or alternate chairman and four members or alternate members of the Plant Staff as designated by the Chairman.

Alternate; Only one alternate shall participate as a voting member in ROC activities at any one time.

Quorum; The Chairman or his designated alternate and two other members including alternates.

- B) USAR paragraph 13.4.2.2, Nuclear Review Board (NRB), is revised as follows:

The function of the NRB is to provide the management of Long Island Lighting Company, through the Assistant Vice President, Nuclear Operations, a mechanism for independently ascertaining that activities related to the nuclear station are performed safely and efficiently in accordance with company policies and regulatory requirements.

The NRB is established and functional; its initial membership comprised LILCO and consultant personnel.

Collectively, the membership has been selected to have the experience and capability to function effectively in the areas of responsibility as designated in license documents. The objectives are to ensure that a representative decision is reached on each issue and that Assistant Vice President, Nuclear Operations is appropriately advised. The NRB membership is selected so that a majority of members are not directly responsible for plant activities. All members, whether LILCO employees or consultants, are afforded equal voting status along with a defined route to advise the Assistant Vice President, Nuclear Operations, of the assessment of dissenting voters.

1. Written Charter

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

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The charter of the NRB reflects the consideration that NRB activities are not limited to items and functions that are designated as safety related. It is intended that NRE review and audit activities will also cover nonsafety 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.

2. Membership

The NRB will consist of the NRB Chairman and at least four permanent members. As a group, they will collectively have the competence required to review problems in the following areas: nuclear engineering, chemistry and radiochemistry, radiological safety, mechanical and electrical engineering, and QA practices.

The Chairman will be appointed by the Assistant Vice President, Nuclear Operations. The Chairman of the NRB is responsible for appointing individuals to NRB membership. Membership appointments are to be such that the collective membership includes the experience and capability noted in the foregoing subsection. Membership appointments are subject to concurrence by the Assistant Vice President, Nuclear Operations.

In the event a regular member is not able to participate in NRB activities, designated alternates are authorized to act in the place of the regular member. Any nominated alternates shall be appointed in writing by the Chairman of the NRB to serve on a temporary basis.

The NRB may obtain recommendations from scientific or technical personnel employed by LILCO or other consultant organizations whenever the NRB Chairman considers it necessary to obtain further scientific or technical assistance in carrying out its responsibility. Such individuals shall function as staff to the NRB, performing tasks and submitting reports as assigned by the action of the NRB.

Minimum qualifications of NRB members are as follows:

- a. The Chairman will be a college graduate or equivalent and will have at least 10 years of experience in the power generation field.
- b. Other members of the NRB and their designated alternates will be graduate engineers or equivalent and will have at least 3 years experience in the

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appropriately related scientific, technical, engineering, or power generation field. Members, or their designated alternates, may possess competence in more than one specialty area.

- c. If sufficient competence in the specialty areas as described in this subsection is not available within LILCO, the review and audit functions will be performed or supplemented by outside consultants or organizations.

The minimum quorum of the NRB necessary for the performance of review and audit functions shall consist of the Chairman (or his designated alternate) and at least three members, including alternates. Less than a majority of the quorum shall have line responsibility for the operation of the Shoreham Nuclear Power Station. A quorum shall be considered filled if conference telephone communications are established with the requisite number of members or alternates at remote locations. No more than two alternates shall participate as voting members in NRB activities at any meeting.

3. Meeting Frequency

The NRB shall meet at least once per six months.

Any member may request a special NRB meeting to consider a matter believed to involve a safety or radiological environmental problem.

4. Records

- a. Minutes shall be recorded for all meetings of the NRB. The minutes shall identify all documentary material reviewed and the findings, recommendations, and actions taken by the NRB. Meetings shall be numbered in sequence, and minutes of meetings shall be distributed to the President; the Assistant Vice President, Nuclear Operations; and NRB members within 14 days following each meeting.
- b. Reports of audits submitted to or conducted under the cognizance of the NRB, including recommendations of the NRB, shall be made in writing to the Assistant Vice President Nuclear Operations; and to the management positions responsible for the areas audited within 30 days after completion of the audit.

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5. Review Responsibilities

The NRB shall review:

- a. The safety evaluations for (1) changes to equipment or systems and (2) tests or experiments completed under the provision of 10CFR Section 50.59, to verify that such actions did not constitute an unreviewed safety question.
- b. Proposed changes to procedures, equipment, or systems that involve an unreviewed safety question as defined in 10 CFR, Section 50.59.
- c. Proposed tests or experiments that involve an unreviewed safety question as defined in 10 CFR, Section 50.59.
- d. Proposed changes to the Shoreham Technical Specifications or the Shoreham Station Operating License.
- e. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, or 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.
- i. Reports and meeting minutes of the Shoreham ROC.

6. Audit Responsibilities

Audits of Shoreham Station activities required by Technical Specifications shall be performed under cognizance of the NRB. These audits shall encompass:

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

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- b. The performance, training, and qualifications of the entire station staff at least once per 12 months.
- c. The results of actions taken to correct deficiencies occurring in station equipment, structures, systems, or methods of operation that affect nuclear safety at least once per year.
- d. The performance of activities required by the QA Program to meet the criteria of 10CFR50, Appendix B, 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 independent fire protection consultant.
- f. Any other area of station operation considered appropriate by the NRB or the Assistant Vice President, Nuclear Operations.
- 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.
- 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.

7. Authority

The NRB is organizationally responsible to the Assistant Vice President, Nuclear Operations.

8. Procedures

Written administrative procedures for the operation of the NRB will be prepared and maintained.

Those items submitted to the NRB as described in Paragraphs 5(b) through 5(d) above, reviewed by, and accepted by the NRB will be resolved as follows:

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- a. If the NRB is of the opinion that a proposed change, test or experiment does not require approval by the NRC under the terms of the license provisions, it so reports in writing to the Plant Manager, together with a statement of the reasons for its decision. The Plant Manager may then proceed with the change, test, or experiment.
- b. If the NRB is of the opinion that approval of the NRC is required, the Shoreham Nuclear Power Station staff, assisted by other LILCO nuclear organizations or by consultants, shall prepare a request for such approval, including an appropriate safety analysis in support of the request in accordance with approved procedures.

If, in the course of any additional reviews of facility operations, the NRB determines that a variation from the Technical Specifications or an unreviewed safety question exists, the NRB shall immediately notify the Plant Manager, who shall take the necessary steps to ensure nuclear safety.

13.4.3 Shoreham Independent Safety Engineering Group

The Shoreham Independent Safety Engineering Group (ISEG) is eliminated for the DSAR Phase. The ISEG was required to be established by NUREG-0737, TMI Action Plan Requirements by each applicant for an operating license. The ISEG was an independent organization dedicated to improving plant safety through examinations, reviews and audits of plant operations, modification, maintenance and operating characteristics, and NRC and other industry sources of plant design and operating experience and information that may indicate areas for improving plant safety.

During the DSAR Phase, Shoreham will not be operated and the fuel will remain in the spent fuel pool until removed from the plant. In the Shoreham defueled configuration, only maintenance and minor modifications will be performed. The principal function of the ISEG, to improve plant safety during operations, is no longer applicable. The remaining activities are adequately covered under the LILCO Quality Assurance Program for Shoreham which will remain unchanged.

13.5 STATION PROCEDURES

13.5.1 Administrative Control

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

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1. Safety-related station procedures shall be processed through the Review of Operations Committee (ROC) and Nuclear Quality Assurance (NQA).
2. The Plant 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 Plant Manager prior to implementation.

See DSAR Figure 13.5.1-1. Refer to the latest revision of the USAR for other information on this subject.

13.5.1.1 Normal Operations

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged with the exception that the NRB has been revised in accordance with 13.4.2C and a new Table 13.5.1-1 is supplied herein.

13.5.1.2 Routine Maintenance, Repairs, and Fuel Handling

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

13.5.1.3 Modifications

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

13.5.2 Procedures

13.5.2.1 Operating Procedures

The description contained under this heading in the latest revision of the Shoreham USAR remains unchanged except that the General Operating Procedures now only describe integrated station operation. Startup and Shutdown are no longer pertinent.

Operating Procedures are not necessarily performed by, or under the direction of, persons holding RO or SRO licenses.

13.5.2.2 Alarm Response Procedures

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

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13.5.2.3 Initial Test Procedures

This section is no longer pertinent.

13.5.2.4 Maintenance Procedures

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

13.5.2.5 Instrument and Control Systems Procedures

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

13.5.2.6 Surveillance Procedures

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

13.5.2.7 Shoreham Nuclear Power Station Emergency Preparedness Plan

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

13.5.2.8 Health Physics Procedures

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

13.5.2.9 Chemistry Procedures

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

13.5.2.10 Reactor Engineering Procedures

Procedures that describe the methods of nuclear performance and evaluation are originated, reviewed and approved in accordance with revised DSAR Figure 13.5.1-1.

13.5.2.11 Plant Security Procedures

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

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13.5.2.12 Radioactive Waste Management Procedures

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

13.5.2.13 Temporary Procedures

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

13.5.2.14 Temporary Changes To Approved Station Procedures

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

13.6 PLANT RECORDS

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

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."

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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 activities:

1. Authorities and Responsibilities for Safe Operation and Shutdown
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 ad. 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
 - l. Fuel Pool Cooling
 - m. Reactor Building Normal Ventilation System (RBNVS)
 - n. Service Water
 - o. Radwaste (Liquid)

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TABLE 13.5.1-1 (Cont'd)

B. Operating Procedures (Cont'd.)

- p. Radwaste (Solid)
- q. Communications System
- r. Condensate Transfer
- s. Deluge and Sprinkler System
- t. Demineralized Water Transfer
- u. Equipment and Floor Drains
- v. Fire Protection System
- w. HVAC - Control Room
- x. HVAC - Turbine Building
- y. HVAC - Radwaste Building
- z. Makeup Water Treatment
- aa. Station Air System
- ab. Smoke, Temperature, and Flame Detection System
- ac. Turbine Building Closed Loop Cooling System
- ad. 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. Fire in Control Room
- d. Fuel Handling Accident
- e. Plant Fires
- f. Loss of Electrical Power
- g. Loss of Instrument Air
- h. Loss of Service Water
- i. Loss of Turbine Building Closed Loop Cooling Water
- j. Secondary Containment Control
- k. Radioactive Release Control

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.

Note: Procedures not designated as emergency procedures shall be incorporated in the Abnormal Performance section of the appropriate system or general operating procedures.

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TABLE 13.5.1-1 (Cont'd)

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. Maintenance Procedures

Maintenance Procedures shall be provided to cover the following maintenance activities.

1. Control of Welding Processes, Materials, and Welder Qualifications
2. Preventive and Corrective Maintenance of Safety Related Equipment

E. Instrument and Control Procedures shall be provided to cover the following instrumentation and control activities:

1. Measuring and Test Equipment
2. Protective Relaying
3. Instrument Records
4. Surveillance Testing
5. Preventive Maintenance of Process Instrumentation

F. 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

G. 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

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TABLE 13.5.1-1 (Cont'd)

H. 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

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TABLE 13.5.1-2RESPONSIBILITY FOR ORIGATION OF STATION PROCEDURES

<u>Procedure Type</u>	<u>Responsible for Origination</u>
Administrative	Appropriate Section Head/Unit Manager
Operating	Appropriate Section Head
Alarm Response	Appropriate Section Head
Maintenance	Appropriate Section Head
Instrument and Control	Appropriate Section Head
Health Physics	Appropriate Section Head
Radiochemistry	Appropriate Section Head
Reactor Engineering	Appropriate Section Head
Solid Radioactive Waste Handling and Shipping	Appropriate Section Head
Gaseous and Liquid Radioactive Waste Effluent Control	Appropriate Section Head
Fuel Handling	Appropriate Section Head
Surveillance	Appropriate Section Head
Security	Appropriate Section Head

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TABLE 13.5.1-3

FORMAT FOR STATION PROCEDURES

		*SP Number _____	
		Revision _____	Eff. Date _____
	<u>Signature</u>	<u>Date</u>	
Section Head	_____	_____	TPC NO <u>Date Eff</u> <u>Date Expr</u>
Quality Control	_____	_____	_____
Div. Mgr.	_____	_____	_____
Plant Mgr.	_____	_____	_____
	<u>Signature or N/A</u>		
 <u>TITLE</u>			

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 XX.XXX.XX.

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TABLE 13.5.1-3 (Cont'd)

4.0 PRECAUTIONS

General precautions should be listed in this section before the description of the actual procedure.

Precautions should be established, as applicable, to alert the individual performing the task to those situations in which measures should be taken early or when care should be exercised to protect equipment and/or personnel. Precautionary notes applicable to specific steps in the procedure should be included prior to that step in the main body of the procedure and should be clearly identified.

5.0 PREREQUISITES

It is necessary to identify those independent actions or procedures which shall be completed and plant conditions which shall exist prior to performing the procedure. Prerequisites applicable only to specific section of a procedure should be so identified.

6.0 LIMITATIONS AND ACTIONS

Limitations on the parameters being controlled and appropriate corrective measures to return the parameter to normal should be specified when applicable.

7.0 MATERIALS OR TEST EQUIPMENT

Special tools, instrumentation, measuring devices, materials, etc. required to accomplish the work should be identified in this section.

8.0 PROCEDURE

Step-by-step instructions in the degree of detail necessary for performing the required function or task should be provided. These shall be numbered sequentially.

Note 1: Operating Procedures (Table 13.5.1-1, Sections B.2 and B.4) shall, as appropriate, be divided into two categories:

Normal Performance shall include step-by-step instructions to complete the required operation. Subcategories may include startup, routine operation at power, rotation of equipment, and shutdown.

Abnormal Performance shall include instruction to recognize the existence of and to correct out-of-normal conditions that occur during the normal performance. Included may be a statement of the out-of-normal condition, including limits of parameters and/or alarm annunciator action.

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TABLE 13.5.1-3 (Cont'd)

Note 2: Maintenance and/or Calibration Procedures

If technical manual instructions are written in sufficient detail to permit a safe and logical accomplishment of the required task, applicable sections of the technical manual may be referenced.

Note 3: Surveillance Procedures

The step-by-step instructions, with appropriate signoff or checkoff provisions for each step, shall be provided to ensure the proper performance of the surveillance activity.

9.0 ACCEPTANCE CRITERIA

Specific acceptance criteria against which the test results shall be judged for approval/disapproval must be stated clearly. Acceptance criteria may contain qualitative data (i.e., a given event does or does not occur) and/or quantitative data (such as set points, calibration curves, tolerances, etc.) as appropriate for the type of device being tested.

10.0 FINAL CONDITIONS

Provide a listing of those tasks required to return the applicable component or system to operational status and to compile the proper documentation of the procedure. Where applicable, verification of completion will be provided by a signature.

11.0 REFERENCES

This section contains applicable references including appropriate sections of the USAR, Technical Specifications, QA Manual, flow diagrams or other drawings, manufacturer's equipment manuals, other station procedures, and system descriptions.

12.0 APPENDICES

Applicable appendices (in the form of checklists, data sheets, diagrams, etc.) should be included when necessary to support the proper implementation of the procedure.

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TABLE 13.5.1-3 (Cont'd)

EVENT ORIENTED

EMERGENCY PROCEDURE FORMAT

Submitted: _____
(Section Head)

SP Number _____

Approved: _____
(Operations Manager)

Revision _____

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 22.XXX.XX.
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.

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TABLE 13.5.1-3 (Cont'd)

SYMPTOM ORIENTED OPERATING EMERGENCY PROCEDURE FORMAT

**** TITLE**

Submitted: _____ 1. _____
(Section Head) _____

Approved: _____ 2. _____

3. _____
TPC No Effect Expir.
Date of Date
TPC of TPC

**** Should be worded to indicate the purpose of the procedure.**

1.0 PURPOSE

(A brief description of the purpose for which the procedure is intended should be clearly stated).

2.0 ENTRY CONDITIONS

(This section should specify the plant conditions and/or plant procedures which identify the need for performing this procedure).

3.0 OPERATOR ACTIONS

(These steps should specify operation of controls or confirmation of automatic actions that are required to fulfill the purpose of this procedure).

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TABLE 13.5.1-3 (Cont'd)

A ALARM RESPONSE PROCEDURE (ARP)
FORMAT

Submitted: _____ ARP _____
(Section Head) (Window Number)

Approved: _____
(Operations Mgr.) (Panel Number)

(Panel Sub-Section)

Effective Date: _____

Revision: _____

ALARM TITLE

Instr. No. _____ Set Point: Trip _____
Reset _____

POSSIBLE CAUSE

List those conditions which
might have initiated the alarm;
list the most probable cause first.

IMMEDIATE ACTION

List the immediate actions
in order for each possible
cause.

SUBSEQUENT ACTION

List the procedures by title and number that would give follow-up action.

Note 1: ALARM RESPONSE PROCEDURES - will include specific instructions to mitigate the consequences of the condition indicated by the alarmed annunciator. Alarm Response Procedures should be filed in numerical sequence in Appendix I to Volume II of the Station Operation Manual.

REFERENCES

The procedure with which the ARP is associated should be identified.

The reference drawing(s) that details the input and/or control signal to the annunciator and/or its initiating device(s) should be identified.

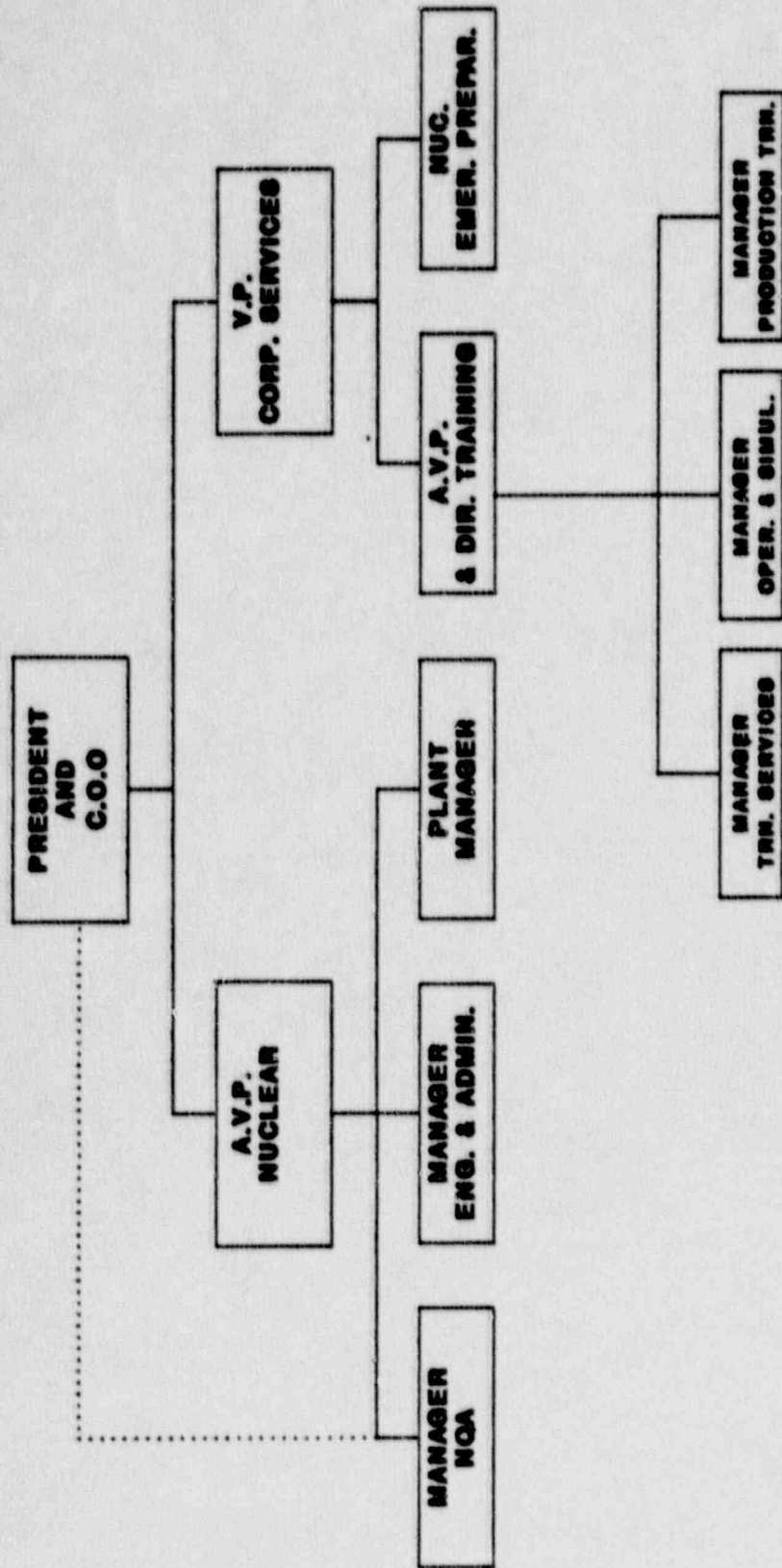


FIGURE 13.1.1-1
DIRECTION OF EXECUTIVE RESPONSIBILITY
SHOREHAM NUCLEAR POWER STATION
DEFUELED SAFETY ANALYSIS REPORT

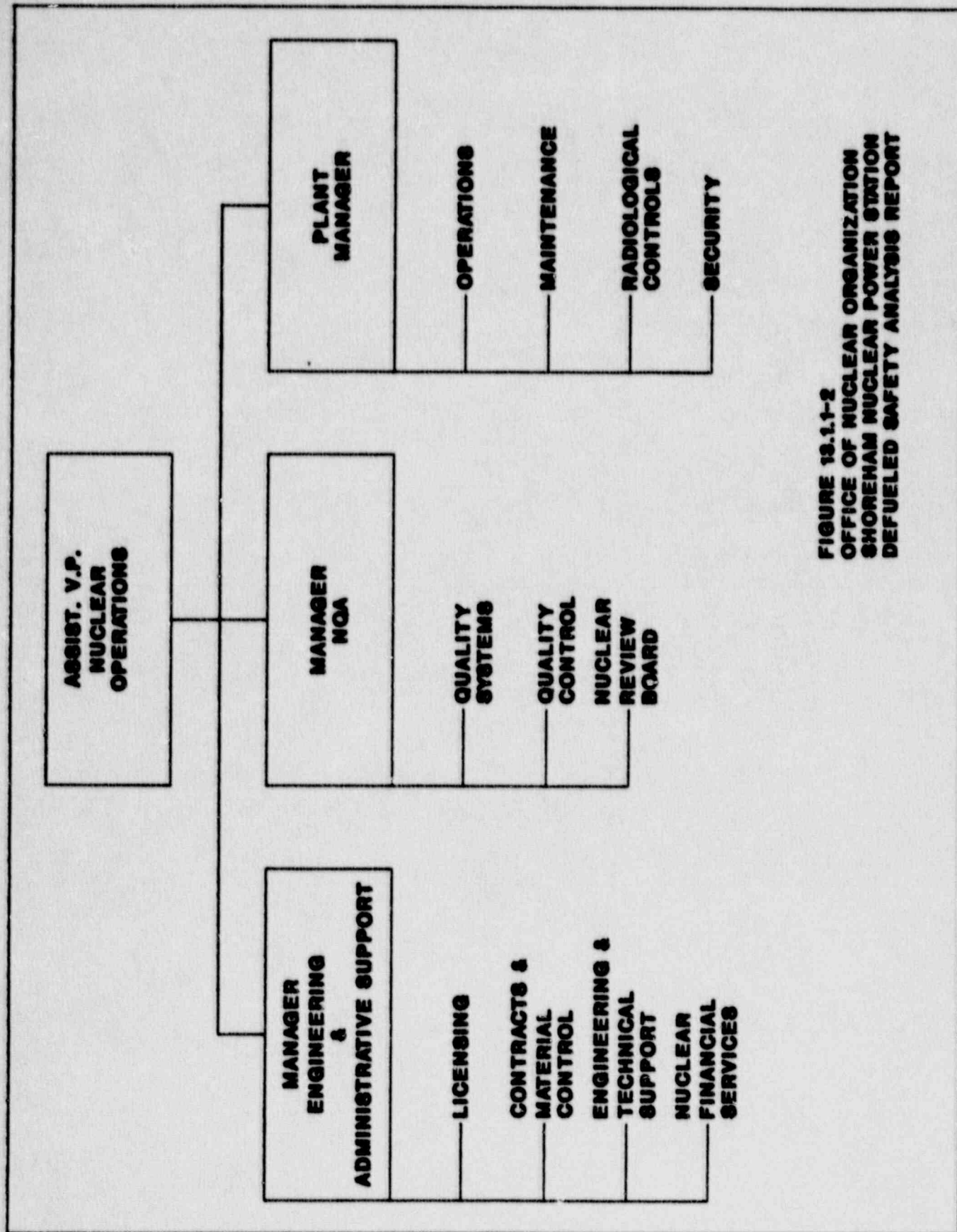


FIGURE 19.1.1-2
OFFICE OF NUCLEAR ORGANIZATION
SHOREHAM NUCLEAR POWER STATION
DEFUELED SAFETY ANALYSIS REPORT

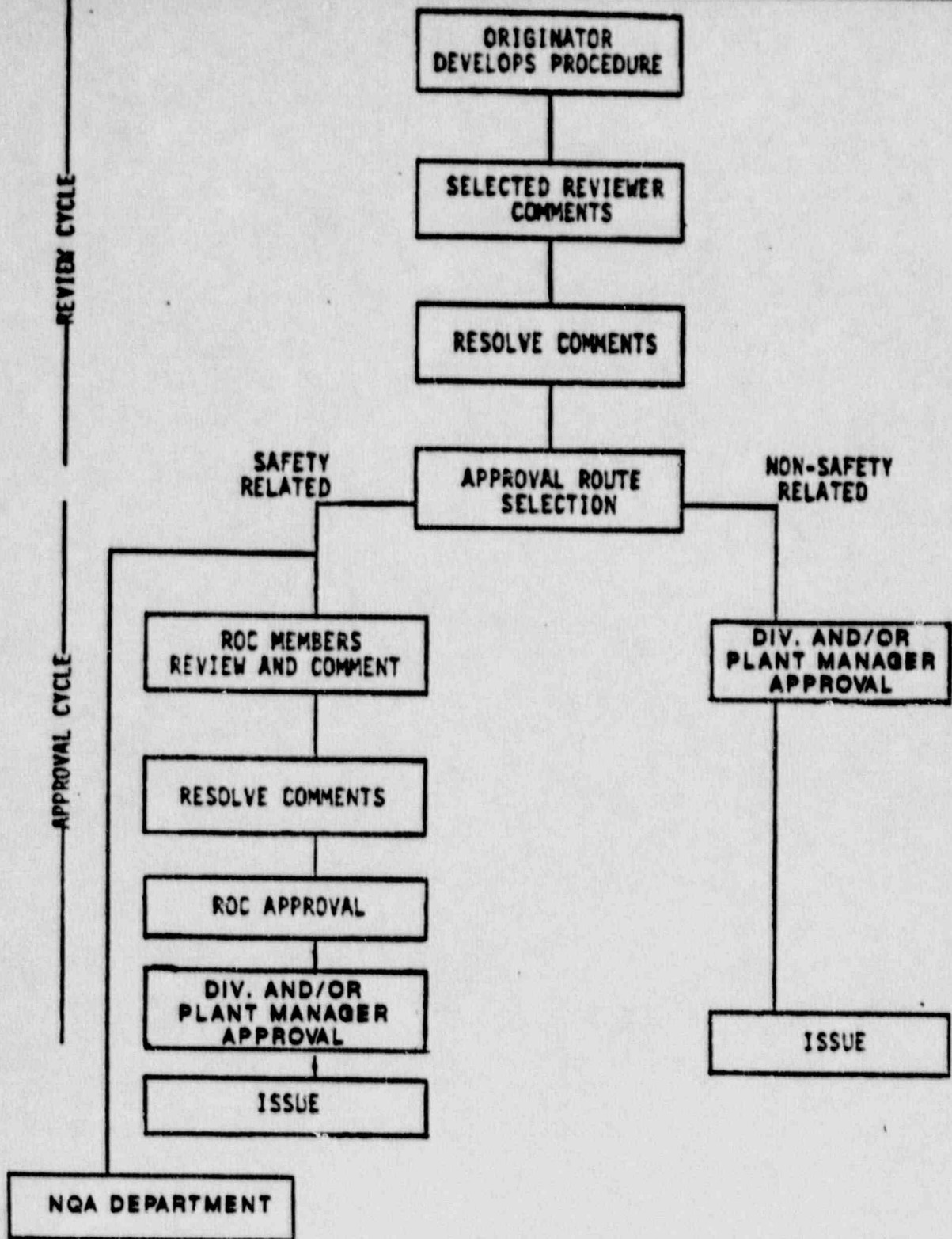


FIGURE 13.5.1-1
PROCEDURE FLOW DIAGRAM
SHOREHAM NUCLEAR POWER STATION
DEFUELED SAFETY ANALYSIS REPORT

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CHAPTER 14

INITIAL TESTS AND OPERATIONS

This chapter is not needed due to the defueled condition of the plant.

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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.

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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

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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 (550 watts) large thermal capacity of the pool active fuel pool cooling is not required. Loss of normal cooling and makeup systems will result only in a very slow evaporation 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

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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:

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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.

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 scenario postulated in the USAR Sections 11.2.3.4.2 through 11.2.3.4.4 is considered here:

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1. A conservative partition factor of $1.0E-03$ is assumed for all isotopic activities listed in Section 11.1 with the exception of H-3, for which it is assumed all activity is evolved into the atmosphere.
2. A two hour release duration is assumed.
3. A ground release atmospheric dispersion factor is assumed, as given in Table 15.1.36-1 for the EAB. Note that the Exclusion Area Boundary (EAB) is limiting insofar as 10CFR100 dose limits are concerned, because the release duration is two hours.
4. The breathing rate of adults 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.

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

<u>Source</u>	<u>Dose, millirem</u>		
	<u>Whole Body Gamma*</u>	<u>Beta Skin</u>	<u>Maximum Organ**</u>
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	<u>$1.8E-05$</u>	<u>$2.8E-06$</u>	<u>$1.3E-03$</u>

The consequences of the above postulated accident are negligible. The whole body gamma, skin, and thyroid doses are $7.2E-08\%$, $9.3E-10\%$, and $4.3E-07\%$, respectively, of the 10CFR100 dose guidelines.

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

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

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15.1.38 Failure of Air Ejector Lines

As the main condenser is not operating, this accident is no longer a design basis event.

6.C Events with Fuel Damage

15.1.36 Fuel Handling Accident

15.1.36.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in the dropping of a raised fuel assembly onto the top of the spent fuel racks.

15.1.36.2 Starting Conditions and Assumptions

Accidents that result in the release of radioactive materials directly to the secondary containment can occur when fuel is being handled. In this case, radioactive material released as a result of fuel damage is available for transport directly to the secondary containment. Table 15.1.36-1 presents the parameters used in this analysis.

15.1.36.3 Accident Description

The most severe fuel handling accident from a radiological viewpoint is the dropping of a fuel assembly onto other fuel assemblies. The sequence of events is as follows:

<u>Event</u>	<u>Approximate Elapsed Time</u>
1. Fuel assembly is being handled by refueling equipment. The assembly drops.	0+
2. Some of the fuel rods in both the dropped assembly and another assembly are damaged, resulting in the release of gaseous fission products to the fuel pool and eventually to the secondary containment atmosphere.	1 min.
3. The reactor building refueling floor ventilation exhaust radiation monitoring system may alarm to alert plant personnel.	1 min.
4. Operator actions begin	5 min.

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15.1.36.4 Identification of Operator Actions

1. The operator will initiate the evacuation of the secondary containment and securing of Secondary Containment doors, if necessary.
2. The fuel handling foreman will instruct personnel to go immediately to the radiation protection personnel decontamination area, if necessary.
3. The fuel handling foreman will make the operator aware of the accident.
4. The operator will initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the secondary containment.
5. An HP technician will post the appropriate radiological control signs at the entrance to the secondary containment.
6. Before entry to the secondary containment is made, a careful study of conditions, radiation levels, etc., will be performed.

15.1.36.5 HVAC Scenarios Considered

As set forth in Section 15.1.36.6, the quantity of gaseous fission products in the fuel's gap which is released will not be large (2.52 Ci of Kr-85 only). Calculations indicate that the reactor building refueling floor exhaust radiation monitoring system would not alarm and consequently the RBSVS will not be actuated (i.e., the RBNVS continues to operate). As a result, analyses were performed assuming either RBSVS or RBNVS system operation. Secondary containment discharge rates are 167 and 6580 percent/day for the RBSVS and RBNVS cases, respectively. As a comparison case, a "puff" release over a short period of time (2 hours, as suggested by Regulatory Guide 1.25), has been analyzed. Although this is not a design basis case, it is useful to compare it with the two HVAC cases. Results for all three cases (RBSVS, RBNVS, and puff release) are given in the following sections.

15.1.36.6 Analysis of Effects and Consequences

15.1.36.6.1 Evaluation Methods

The analytical methods and associated assumptions used to evaluate the consequences of this accident are consistent with Regulatory Guide 1.25. The assumptions and parameters are given in Table 15.1.36-1.

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15.1.36.6.1.1 Methods, Assumptions, and Conditions

The assumptions used in the analysis of this accident are listed below:

1. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.
2. The entire amount of potential energy, referenced to the top of the spent fuel racks, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the racks and requires the complete detachment of the assembly from the fuel hoisting equipment. This is possible if the fuel assembly handle, the fuel grapple, or the grapple cable breaks.
3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

15.1.36.6.1.2 Results and Consequences

15.1.36.6.1.2.1 Fuel Damage

The analysis of USAR Section 15.1.36.5.1.2.1 applies to this accident. In that section of the USAR, it was assumed that 125 fuel rods would fail as a result of dropping the fuel assembly into the reactor vessel. The same assumption is applied here.

15.1.36.6.1.2.2 Fission Product Release From Fuel

Fission product releases for the fuel handling accident are determined from the inventory in Table 12.2-1. Specifically, it is seen that only Kr-85 is of any significance with respect to gaseous releases. The only other gaseous isotope in this table is H-3, which would add, at most, 0.1% to the skin dose from Kr-85. Using the above number of failed rods, and the assumptions given in Regulatory Guide 1.25, the quantity of Kr-85 released, is as follows:

$$\text{Release} = 1.56\text{E}+03\text{Ci} \times \frac{125 \text{ damaged rods}}{62 \text{ rods/bundle} \times 560 \text{ bundles in core}}$$

$$\times 1.5 \text{ radial peaking factor} \times 30\% \text{ in gap} = 2.52 \text{ Ci}$$

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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, $9.59\text{E-}08$ and $2.08\text{E-}04$ 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, the 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 ($1.56\text{E+}03\text{ Ci Kr-85}$). 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.

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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 scenario are above current limits, but the duration of the high dose rates in the RBNVS and puff release cases is quite short (two hours or less).

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TABLE 15.1.36-1FUEL HANDLING ACCIDENT - PARAMETERS
FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) <u>Assumptions</u>
I. Data and assumptions used to estimate radioactive source from postulated accidents	
A. Power level	See this Chapter
B. Peaking factor	1.5
C. Fuel damaged	125 rods
D. Release of activity from fuel	30% Kr-85
E. Iodine fractions	
(1) Organic	N/A
(2) Elemental	N/A
(3) Particulate	N/A
II. Data and assumptions used to estimate activity released	
A. Secondary containment discharge rate (%/day)	See Section 15.1.36.5
B. Adsorption and filtration efficiencies	
(1) Elemental iodine	N/A
C. Recirculation system parameters	
(1) Flow rate	N/A
(2) Mixing efficiency	N/A
III. Dispersion data	
A. EAB and LPZ distances (meters)	311/3,220
B. X/Qs (sec/m ³)	
EAB (0-2 hr)	1.36E-03
LPZ (0-8 hr)	2.50E-05
(8-24 hr)	1.75E-05
(1-4 days)	7.80E-06
(4-30 days)	2.45E-06
IV. Dose data	
A. Method of dose calculation	Regulatory Guide 1.25
B. Dose conversion assumptions	Regulatory Guide 1.25
C. Doses and Dose Rates	Table 15.1.36-2

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TABLE 15.1.36-2

FUEL HANDLING ACCIDENT
RADIOLOGICAL CONSEQUENCES

HVAC Scenario	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit*
RBSVS Operates	1.14E-07	1.22E-08	2.50E+01	9.90E-06	1.06E-06	3.00E+02
Maximum (t = 0) Dose Rates, mrem/hr						
	Whole Body Gamma			Skin		
	EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
	6.10E-05	1.12E-06	5.70E-02	5.30E-03	9.74E-05	3.42E-01
RBNVS Operates	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit*
	1.74E-06	3.22E-08	2.50E+01	1.52E-04	2.80E-06	3.00E+02
Maximum (t = 0) Dose Rates, mrem/hr						
	Whole Body Gamma			Skin		
	EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
	4.79E-03	8.81E-05	5.70E-02	4.17E-01	7.66E-03	3.42E-01
Puff Release	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit*
	1.75E-06	3.22E-08	2.50E+01	1.52E-04	2.80E-06	3.00E+02
Maximum (t = 0) Dose Rates, mrem/hr						
	Whole Body Gamma			Skin		
	EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
	8.75E-04	1.61E-05	5.70E-02	7.61E-02	1.40E-03	3.42E-01

* The skin dose limit is set equal to the thyroid limit.

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TABLE 15.1.36A-1

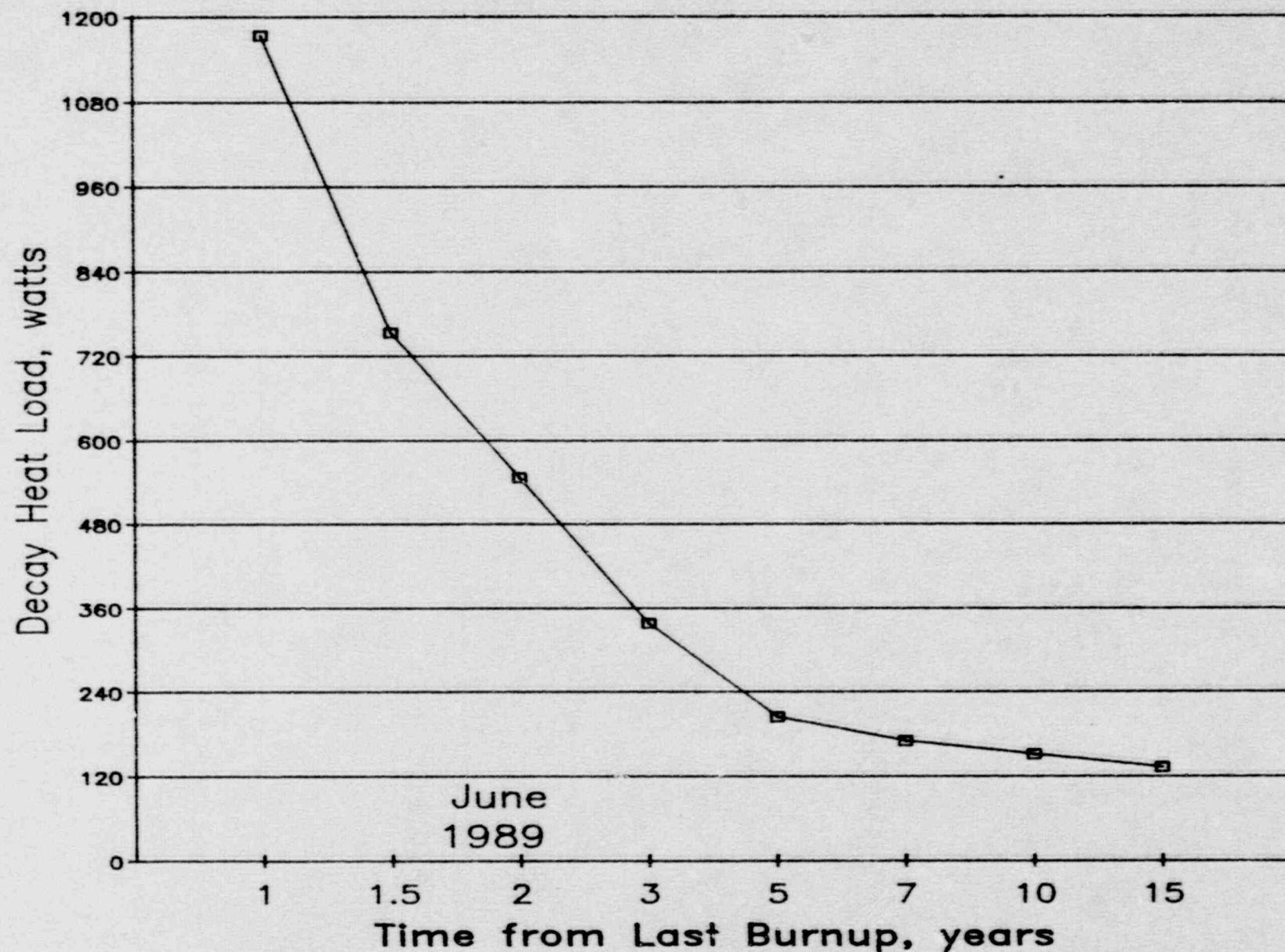
"WORST CASE" FUEL DAMAGE ACCIDENT
RADIOLOGICAL CONSEQUENCES

HVAC Scenario	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit
RBSVS Operates	7.03E-05	7.50E-06	2.50E+01	6.11E-03	6.52E-04	3.00E+02
	Maximum (t = 0 Dose Rates, mrem/hr					
	Whole Body Gamma			Skin		
	EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
	3.76E-02	6.92E-04	5.70E-02	3.27E+00	6.01E-2	3.42E-01
RBNVS Operates	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit
	1.08E-03	1.99E-05	2.50E+01	9.35E-02	1.73E-03	3.00E+02
	Maximum (t = 0) Dose Rates, mrem/hr					
	Whole Body Gamma			Skin		
	EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
	2.96E-00	5.44E-02	5.70E-02	2.57E+02	4.73E+00	3.42E-01
Puff Release	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit
	1.08E-03	1.99E-05	2.50E+01	9.39E-02	1.73E-03	3.00E+02
	Maximum (t = 0) Dose Rates, mrem/hr					
	Whole Body Gamma			Skin		
	EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
	5.40E-01	9.93E-03	5.70E-02	4.70E+01	8.63E-01	3.42E-01

* Skin dose limit set equal to thyroid limit

DSAR FIGURE 15.1-1

SNPS Spent Fuel Decay Heat Load

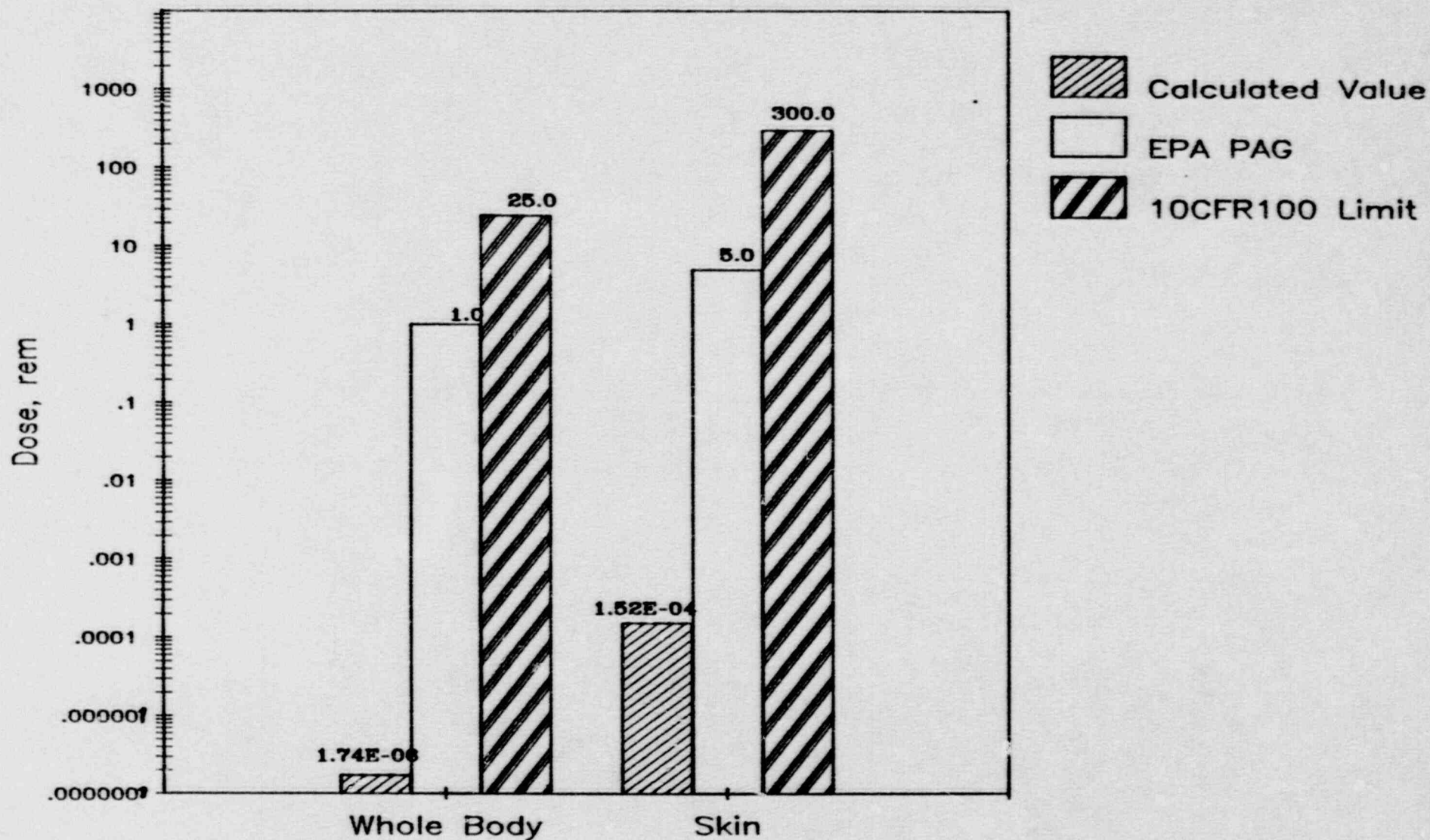


DSAR Figure 15.1.36-1

Design Basis Fuel Handling Accident

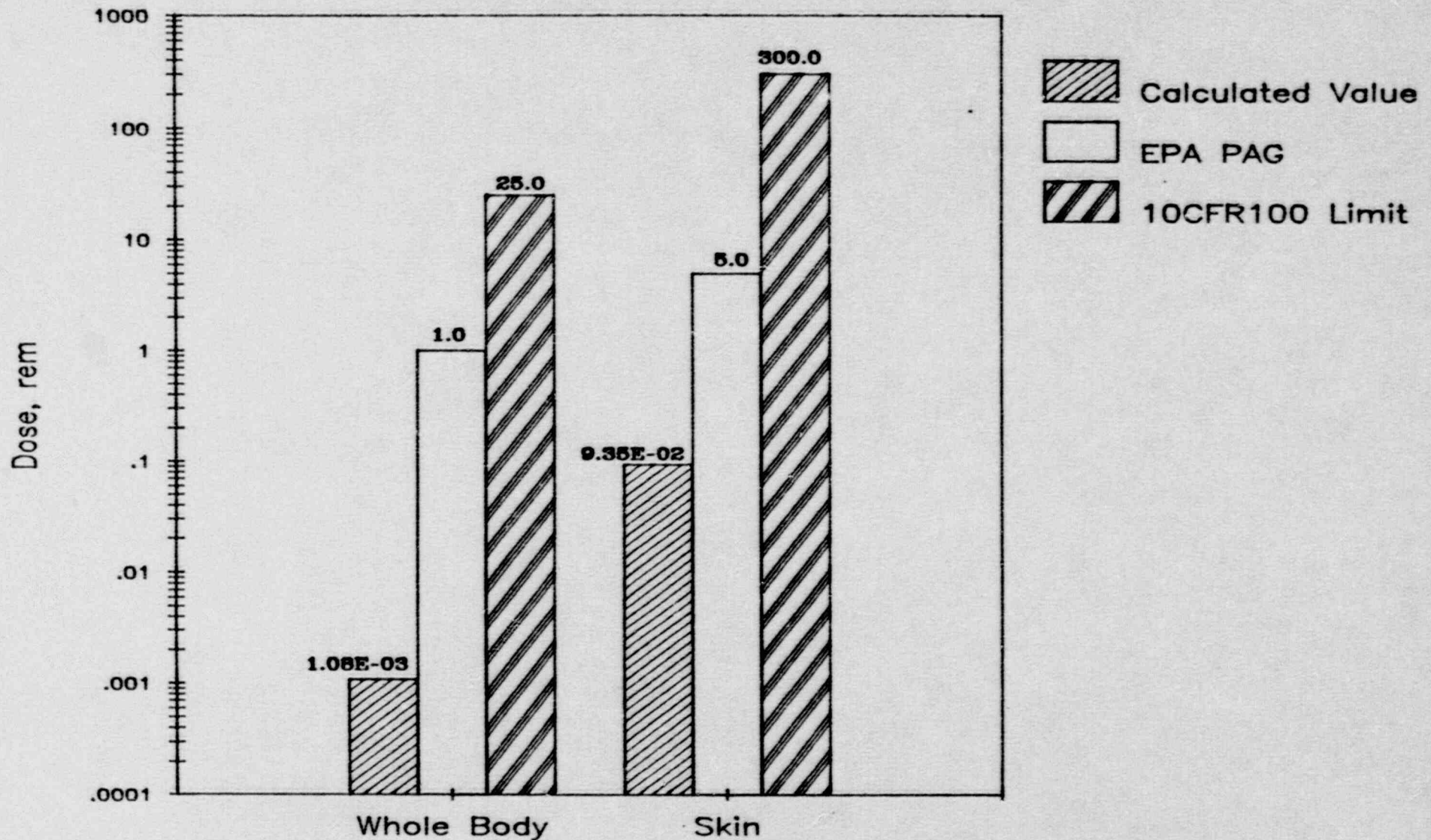
Exclusion Area Boundary Results

RBNVS HVAC System In Operation



DSAR FIGURE 15.1.36A-1

Worst Case Fuel Damage Accident
Exclusion Area Boundary Results
RBNVS HVAC System In Operation



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CHAPTER 16

TECHNICAL SPECIFICATIONS

The SNPS Technical Specifications are found in Appendix 1 of NPF-82.

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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 OPERATIONAL 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 essentially unchanged. 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 IIA 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 3.2.1-1 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) | - Category IIA systems or components which were previously designated as QA Category I but no longer have a safety function. These systems or components are considered non-safety related and will no longer comply with QA Appendix B or Nuclear Codes and Standards. Deviations from previously defined QA Category I requirements will be documented and filed for retrievability. |
| QA Category II
(non safety related) | - Appropriate measures are applied to these structures, systems, and components in accordance with QA corporate policy to assure that the safety significance given to them in the USAR, Technical Specifications, and Emergency Operating Procedures are maintained. |

The specific modifications of the USAR Section 17.2 applicable to the Shoreham DSAR phase are as follows:

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17.2.1 Organization

- A) The Assistant Vice President, Nuclear Operations has assumed all the responsibilities of the Vice President, Nuclear Operations and the position of Vice President, Nuclear Operations was eliminated.
- B) The Manager, Nuclear Quality Assurance Department (NQAD), reports directly to the Assistant Vice President, Nuclear Operations, but has direct access to the President of the Company as he deems necessary.
- C) The Safety Engineering and Reliability organization that reported to the Manager, NQAD, has been eliminated. This organization included the Independent Safety Engineering Group (ISEG) and Reliability Section. The Nuclear Review Board continues under the responsibility of the NQA Manager as Chairman. The Quality Control and Quality Systems organizations continue to be jointly responsible for assuring full implementation of the LILCO QA Program.
- D) The Quality Systems (QS) Manager is located within the protected area.
- E) The title of Senior Vice President is changed to Group Vice President.
- F) The calibration services of Gas Operations and Electric Operations are no longer utilized by SNPS. These services are now performed by SNPS personnel or qualified suppliers.
- G) The Vice President, Corporate Services, reports to the President and provides SNPS with training, emergency preparedness and nondestructive examination services. The primary responsibility for providing these services previously were the Assistant Vice President and Director of the Office of Training, Nuclear Operations Support Department and Nuclear Quality Assurance Department, respectively. These services continue to be subject to the policies and requirements of the QA Program.
- H) The LILCO organization structure for the Shoreham DSAR Phase is shown in Figure 17.2.1-1.
- I) The Nuclear Engineering Department (now Engineering and Technical Support) has been reorganized and reports to the Manager, Engineering and Administrative Support.
- J) The Nuclear Operations Support Department and the Nuclear Engineering Department have been combined in the Engineering and Administrative Support Organization.

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17.2.2 Quality Assurance Program

- A) The Reliability and ISEG organization are eliminated from the NQAD.
- B) The structures, systems, and components designated as QA Category I (safety related) in USAR Table 3.2.1-1 which no longer have a safety function are designated as QA Category IIA. These structures, systems, and components will no longer require a QA Appendix B program and need not comply with Nuclear Codes and Standards. These structures, systems, and components will be afforded the requirements associated with non-safety systems. Deviations from QA Category I requirements will be documented and filed for traceability.

17.2.3 Design Control

No change.

17.2.4 Procurement Document Control

No change.

17.2.5 Instructions, Procedures, and Drawings

No change.

17.2.6 Document Control

No change.

17.2.7 Control of Purchased Material, Equipment, and Services

No change.

17.2.8 Identification and Control of Material, Parts and Components

No change.

17.2.9 Control of Special Processes

No change.

17.2.10 Inspection

No change.

17.2.11 Test Control

No change.

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17.2.12 Control of Measuring and Test Equipment

No change.

17.2.13 Inspection, Test, and Operating Status

No change.

17.2.14 Inspection, Test, and Operating Status

No change.

17.2.15 Nonconforming Materials, Parts, or Components

No change.

17.2.16 Corrective Action

No change.

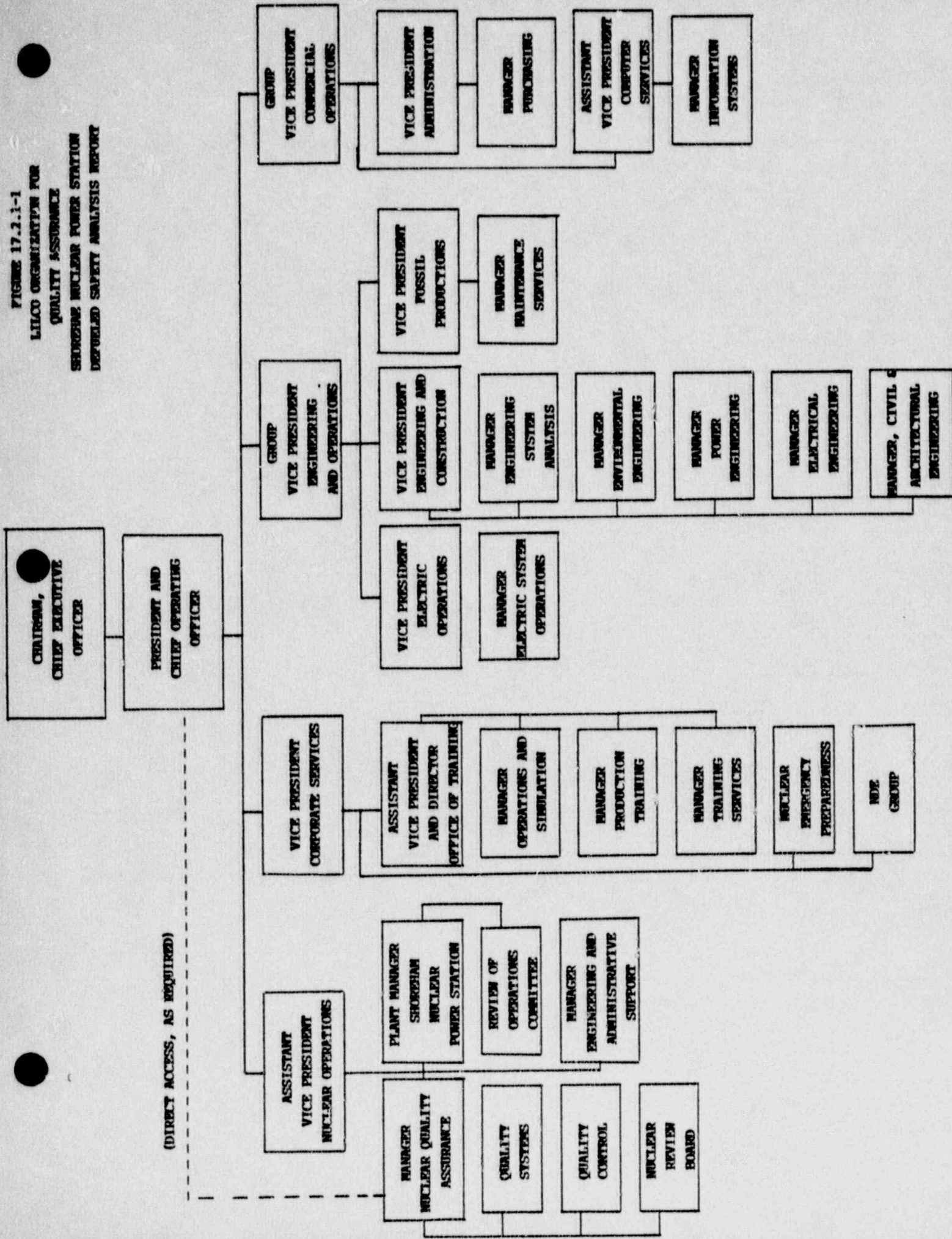
17.2.17 Quality Assurance Records

No change.

17.2.18 Audits

No change.

FIGURE 17.2.1-1
 LILCO ORGANIZATION FOR
 QUALITY ASSURANCE
 SHOREHAM NUCLEAR POWER STATION
 DEFUELED SAFETY ANALYSIS REPORT



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NUREG-0737 TMI ACTION PLAN REQUIREMENTS

I. OPERATIONAL SAFETY

I.A.1.1 Shift Technical Advisor

NRC Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

LILCO Position

Under the conditions of the LILCO and New York State Shoreham settlement the plant is shutdown and defueled. STA staffing is therefore not required.

I.A.1.2 Shift Supervisor Administrative Duties

NRC Position

Review the administrative duties of the shift supervisor and delegate functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant to other personnel not on duty in the control room.

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.
2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

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- a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The principle shall be reinforced that the shift supervisor should not become totally involved in any single operations in times of emergency when multiple operations are required in the control room.
 - b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.
 - c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.
3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function that the shift supervisor is to provide for assuring safety.
 4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

LILCO Position

In lieu of a Shift Supervisor SNPS uses a Watch Engineer. The administrative duties are minimized by the plant being shutdown and defueled with some systems being preserved/protected. Additionally, the administrative duties have little effect on safe operation of the plant. See the USAR for additional information.

A Corporate Management Directive that emphasizes the primary management responsibility of the Watch Engineer for the safe operation of the plant under all conditions is issued by the Assistant Vice President, Nuclear Operations annually.

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I.A.1.3 Shift Manning

NRC Position

Assure that the necessary number and availability of personnel to man the operations shifts have been designated by the licensee. Administrative procedures should be written to govern the movement of key individuals about the plant to assure that qualified individuals are readily available in the event of an abnormal or emergency situation. This should consider the recommendations on overtime in NUREG-0578. Provisions should be made for an aide to the shift supervisor to assure that, over the long term, the shift supervisor is free of routine administrative duties.

At any time a licensed nuclear unit is being operated in Modes 1-4 for a PWR (Power Operation, Startup, Hot Standby, or Hot Shutdown, respectively) or in Modes 1-3 for a BWR (Power Operation, Startup, or Hot Shutdown, respectively), the minimum shift crew shall include two licensed senior reactor operators (SRO), one of whom shall be designated as the shift supervisor, two licensed reactor operators (RO), and two unlicensed auxiliary operators (AO). For a multi-unit station, depending upon the station configuration, shift staffing may be adjusted to allow credit for licensed senior reactor operators and licensed reactor operators to serve as relief operators on more than one unit; however, these individuals must be properly licensed on each such unit. At all other times, for a unit loaded with fuel, the minimum shift crew shall include one shift supervisor who shall be a licensed senior reactor operator (SRO), one licensed reactor operator (RO), and one unlicensed auxiliary operator (AO).

Adjunct requirements to the shift staffing criteria stated above are as follows:

- a. A shift supervisor with a senior reactor operator's license, who is also a member of the station supervisory staff, shall be onsite at all times when at least one unit is loaded with fuel.
- b. A licensed senior reactor operator (SRO) shall, at all times, be in the control room from which a reactor is being operated. The shift supervisor may from time-to-time act as relief operator for the licensed senior reactor operator assigned to the control room.
- c. For any station with more than one reactor containing fuel, the number of licensed senior reactor operators onsite shall, at all times, be at least one more than the number of control rooms from which the reactors are being operated.
- d. In addition to the licensed senior reactor operators specified in a., b., and c. above, for each reactor

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containing fuel, a licensed reactor operator (RO) shall be in the control room at all times.

- e. In addition to the operators specified in a., b., c., and d. above, for each control room from which a reactor is being operated, an additional licensed reactor operator (RO) shall be onsite at all times and available to serve as relief operator for that control room. As noted above, this individual may serve as relief operator for each unit being operated from that control room, provided he holds a current license for each unit.
- f. Auxiliary (non-licensed) operators shall be properly qualified to support the unit to which assigned.
- g. In addition to the staffing requirements stated above, shift crew assignments during periods of core alterations shall include a licensed senior reactor operator (SRO) to directly supervise the core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties.

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," dated February 1, 1980 discusses the concern of overtime work for members of the plant staff who perform safety-related functions.

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hour working day, the effects of shift rotation, and other factors. NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02.

In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance or major

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plant modifications), the following overtime restrictions should be followed:

- (1) An individual shall not be permitted to work more than 12 hours straight (not including shift turnover time).
- (2) There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.
- (3) An individual shall not work more than 72 hours in any 7-day period.
- (4) An individual shall not work more than 14 consecutive days without having 2 consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation may be authorized by the plant manager or his deputy, or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individual shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

We encourage the development of a staffing policy that would permit the licensed reactor operators and senior reactor operators to be periodically assigned to other duties away from the control board during their normal tours of duty.

If a reactor operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about 4 hours at a time.

The guidelines on overtime do not apply to the shift technical advisor provided he or she is provided sleeping accommodations and a 10-minute availability is assured.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after July 31, 1980.

LILCO Position

The Shoreham Station Procedures implement the following:

1. The minimum shift complement consists of three operators and a sufficient number of extra people in order to meet the Emergency Plan and Fire Protection Plan requirements.

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2. The shift schedule conforms to the guidelines provided in the Shoreham Station Procedure entitled Station Operations - Overtime Selection as it applies to the scheduling and use of overtime.
3. The movement in the plant by members of the shift complement are such that they may be easily and rapidly informed and/or contacted and dispatched by the operators in the event an emergency situation arises.

The above items ensure that qualified plant personnel are available to man operational shifts.

The Shoreham Station Procedure entitled Station Operations - Overtime Selection, implements the requirements of the Technical Specifications (Chapter 16).

The Operators are trained and qualified as outlined in the Shoreham Station Procedure entitled Operations Section Training and Qualification Program. Since SNPS contains only one unit and since no other units are operated by LILCO, the requirement that Auxiliary (nonlicensed) operators be properly qualified to support the unit to which assigned is not a problem at Shoreham.

The Shoreham Nuclear Power Station is in complete compliance with the portions of this Task Action Item that apply to a shutdown and defueled plant.

I.A.2.1 Immediate Upgrade of Reactor Operator and Senior Reactor Operator Training and Qualification

NRC Position

Effective May 1, 1980, an applicant for a senior reactor operator (SRO) license shall have four years of responsible power plant experience of which at least two years shall be nuclear power plant experience. Six months of the nuclear power plant experience shall be at the plant on which the applicant is licensing. A maximum of two years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis.

Effective December 1, 1980, an applicant for a senior reactor operator (SRO) license shall have held an operator's license for one year.

Effective August 1, 1980, an applicant for a senior reactor operator (SRO) license shall have three months of shift training as an extra man on shift. An applicant for a reactor (RO) license shall have three months training on shift as an extra person in the control room.

Effective August 1, 1980, training programs shall be modified to provide:

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1. Training in heat transfer, fluid flow, and thermodynamics,
2. Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged,
3. Increased emphasis on reactor and plant transients.

Effective May 1, 1980, certifications that operator license applicants have learned to operate the controls shall be signed by the highest of corporate management for plant operation.

LILCO Position

There will not be any applicants for SRO or RO licenses.

The SNPS operator training program will provide training for subjects applicable to a shutdown and defueled plant and also for fuel handling operations.

I.A.2.3 Administration of Training Programs

NRC Position

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations shall be walked.

Simulator examinations will be included as part of the licensing examinations.

LILCO Position

It is LILCO's position that permanent members of the training staff who teach systems, integrated responses, or transients be qualified or certified to teach in the appropriate subject area.

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LILCO does not intend to require either guest lecturers who are experts in particular subjects (reactor theory, instrumentation, thermodynamics, health physics, chemistry, etc.) to successfully complete an NRC SRO examination; or system experts, such as an instrument and control supervisor teaching the control rod drive system to successfully complete an NRC SRO examination.

The degree of training provided will be commensurate with the tasks required to be performed.

I.A.3.1 Revise Scope and Criteria for Licensing Examinations

NRC Position

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

Contents of the licenses operator requalification program shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations shall be walked.

Simulator examinations will be included as part of the licensing examinations.

LILCO Position

LILCO will not have applicants for operator licenses. Licenses will be limited to fuel handling operations.

I.B.1.2 Evaluation of Organization and Management

NRC Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent

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review and audits of plant activities including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verifications that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

The new ISEG shall not replace the plant operations review committee (PORC) and the utility's independent review and audit group as specified by current staff guidelines (Standard Review Plan, Regulatory Guide 1.33, Standard Technical Specifications). Rather, it is an additional independent group of a minimum of five dedicated, full-time engineers, located onsite, but reporting offsite to a corporate official who holds a high-level, technically oriented position that is not in the management chain for power production. The ISEG will increase the available technical expertise located onsite and will provide continuing, systematic, and independent assessment of plant activities. Integrating the Shift Technical Advisors into the ISEG in some way would be desirable in that it could enhance the group's contact with and knowledge of day-to-day plant operations and provide additional expertise. However, the shift technical advisor on shift is necessarily a member of the operating staff and cannot be independent of it.

It is expected that the ISEG may interface with the quality assurance (QA) organization, but preferably should not be an integral part of the QA organization.

The functions of the ISEG require daily contact with the operating personnel and continued access to plant facilities and records. The ISEG review functions can, therefore, best be carried out by a group physically located onsite. However, for utilities with multiple sites, it may be possible to perform portions of the independent, safety assessment function in a centralized location for all the utility's plants. In such cases, an onsite group still is required, but it may be slightly smaller than would be the case if it were performing the entire independent safety assessment function. Such cases will be reviewed on a case-by-case basis.

At this time, the requirements for establishing an ISEG is being applied only to applicants for operating licenses in accordance

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with Action Plan item I.B.1.2. The staff intends to review this activity in about a year to determine its effectiveness and to see whether changes are required. Applicability to operating plants will be considered in implementing long-term improvements in organization and management for operating plants (Action Plan item I.B.1.1).

LILCO Position

LILCO does not have an ISEG. The functions of an ISEG do not apply to shutdown and defueled reactor which will not be operated.

I.C.1 Guidance for the Evaluation and Development of Procedures for Transients and Accidents

NRC Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operating retraining (see also Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (see Table C.1, items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, items 4, 12, 17, 18, 19, 20; and Table C.3, items 6, 35, 37, 38, 39, 41, 47, 55, 57).

Based on staff reviews to date, there appear to be some recurring deficiencies in the guidelines being developed. Specifically, the staff has found a lack of justification for the approach used (i.e., symptom-, event-, or function-oriented) in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail.

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The analysis conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concluded that the single-failure criterion was not considered appropriate for guideline development and called for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients and accidents and inadequate core cooling analyzed should postulate multiple failures such that, if the failures were unmitigated, conditions of inadequate core cooling would result.

Examples of multiple failure events include:

1. Multiple tube rupture in more than one steam generator and tube rupture in more than one steam generator;
2. Failure of main and auxiliary feedwater;
3. Failure of high-pressure reactor coolant makeup system;
4. An anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open relief valve or safety relief valve, or loss of main feedwater;
5. Operator errors of omission or commission.

The analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/neutronic phenomena are identified (e.g., upper head voiding due to rapid cooldown, steam generator stratification). Failures and operator errors during the long term cooldown period should also be addressed.

The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions. Rationale for this transition should be discussed. Additional information that should be submitted includes:

1. A detailed description of the methodology used to develop the guidelines;
2. Associated control function diagrams, sequence-of-event diagrams, or others, if used;
3. The bases for multiple and consequential failure considerations;
4. Supporting analysis, including a description of any computer codes used;

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5. A description of the applicability of any generic results to plant-specific applications.

Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in task action plan Item I.C.8 (NUREG-0660).

For PWRs, this will involve review of the loss-of-coolant, steam-generator-tube rupture, loss of main feedwater, and inadequate core cooling procedures. The adequacy of each PWR vendor's guidelines will be identified to each NTOL during the emergency procedure review. Since the analysis and guidelines submitted by the General Electric Company (GE) owners' group that comply with the requirements stated above have been reviewed and approved for trial implementation of six plants with applications for operating licenses pending, the interim program for BWRs will consist of trial implementation on these six plants.

Following approval of analysis and guidelines and the pilot monitoring of emergency procedures, the staff will advise all licensees of the adequacy of the guidelines for application to their plants. Consideration will be given to human factors engineering and system operational characteristics, such as information transfer under stress, compatibility with operator training and control room design, the time required for component and system response, clarity of procedural actions, and control room personnel interactions. When this determination has been made by the staff, a long-term plan for emergency procedure review, as described in task action plan Item I.C.8, will be made available. At that time, the reviews currently being conducted on NTOLs under Item I.C.8 will be discontinued, and the review required or applicants for operating licenses will be as described in the long-term plan. Depending on the information submitted to support development of emergency procedures for each reactor type or vendor, this transition may take place at different times. For example, if the GE guidelines are shown to be effective on the six plants chosen for pilot monitoring, the long-term plan for BWRs may be complete in early 1981. Operating plants and applicants will then have the option of implementing the long-term plan in a manner consistent with their operating schedule, provided they meet the final date required for implementation. This may require a plant that was reviewed for an operating license under Item I.C.9 to revise its emergency procedures again prior to the final implementation date for Item I.C.8. The extent to which the long-term program will include review and approval of plant-specific procedures for operating plants has not been established. Our objective, however, is to

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minimize the amount of plant-specific procedure review and approval required. The staff believes this objective can be acceptably accomplished by concentrating the staff review and approval on generic guidelines. A key element in meeting this objective is use of staff-approved generic guidelines and guideline revisions by licensees to develop procedures. For this approach to be effective, it is imperative that, once the staff has issued approval of a guideline, subsequent revisions of the guideline should not be implemented by licensees until reviewed and approved by the staff. Any changes in plant-specific procedures based on unapproved guidelines could constitute an unreviewed safety issue under 10CFR50.59. Deviations from this approach on a plant-specific basis would be acceptable provided the basis is submitted by the licensee for staff review and approval. In this case, deviations from generic guidelines should not be implemented until staff approval is formally received in writing. Interim implementation of analysis and procedures for small-break loss-of-coolant accident and inadequate core cooling should remain on the schedule contained in NUREG-0578, Recommendation 2.1.9.

LILCO Position

The description contained under this heading in the latest revision of Shoreham USAR remains unchanged with the following exception: LILCO will not continue to participate in the BWR Owners' Group program to develop Emergency Procedure Guidelines for GE Boiling Water Reactors. Refer to USAR for information on this subject.

I.C.2 Shift and Relief Turnover Procedures

NRC Position

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:
 - a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).
 - b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable states shall be included on the checklist).

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- c. Identification of systems and components that are in a degraded mode if operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).
2. Checklists or logs shall be provided for completion by the ongoing and oncoming auxiliary operators and technicians. such checklists and logs shall include any equipment under maintenance or test that by themselves could upgrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transients (what to check and criteria for acceptable status shall be included on the checklist), and
3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

LILCO Position

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.

I.C.3 Shift Supervisor Responsibilities

NRC Position

In the letters of September 13 and 27, October 10 and 30, and November 9, 1979, NRC required licensees and applicants to review and revise as necessary plant procedures and directives to assure that the duties, responsibilities, and authority were properly defined to establish a definite line of command and clear delineation of the command decision authority of the supervisor in the control room relative to other plant management personnel. These letters also emphasized the primary management responsibility of the shift supervisor for safe operation of the plant. Training programs for the shift supervisor were required to emphasize and reinforce the responsibility for safe operation and management function of the shift supervisor to assure safe operation of the plant.

LILCO Position

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

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I.C.4 Control Room Access

NRC Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.
2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

LILCO Position

A Shoreham Station Procedure on Control Room Conduct establishes the authority and responsibility of the person in charge of the control room to limit access to the control room.

The same procedure establishes the line of authority and responsibility in the control room in the event of an emergency.

I.C.5 Procedures for Feedback of Operating Experience to Plant Staff

NRC Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

- (1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

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- (2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures, operating orders);
- (3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job function of the recipients;
- (4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;
- (5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;
- (6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,
- (7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to assure that important information on operating experience originating both within and outside the organization is continually provided to operators and other personnel, and that it is incorporated into plant operating procedures and training and retraining programs.

Those involved in the assessment of operating experience will review information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE Bulletins, Circulars, and Notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient importance that it must be dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special changes to operating and emergency procedures, issuance of special precautions, etc.) and must be handled in such a manner to assure that operations management personnel would be directly involved in the process. In many other cases, however, important information will become available which would be brought to the attention of operators and other personnel for their general

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information to assure continued safe plant operations. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealt with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. It is important, also, that technical review be conducted to preclude premature dissemination of conflicting or contradictory information.

LILCO Position

The description contained under this heading in the latest revision of the USAR remains unchanged with the following exceptions:

1. The membership and quorum requirements of the ROC are given in the Technical Specifications.
2. The Nuclear Review Board and ISEG have been eliminated. Refer to Section 13.4.2 and 13.4.3 for justification.
3. The Training Division Manager has been changed to Office of Training.
4. The responsibilities assigned to the Plant Manager, Division Manager and Section Heads have been reassigned to plant management personnel (Section Heads and above).
5. The Training Administrative Supervisor administers the circulation of required reading lists.

I.C.6 Procedures for Verification of Correct Performance of Operating Activities

NRC Position

It is required from NUREG-0660 that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification or operations, and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5) or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the

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licensees may consist of two phases - one before and one after installation of automatic status monitoring equipment, if required, in accordance with item I.D.3 of NUREG-0660.

An acceptable program for verification of operating activities is described below.

The American Nuclear Society has prepared a draft revision to ANS Standard N 18.7-1972 (ANS). Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants. A second proposed revision to Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), which is to be issued for public comment in the near future, will endorse the latest draft revision to ANS 3.2 subject to the following supplemental provisions:

- (1) Applicability of the guidance of Section 5.2.6 should be extended to cover surveillance testing in addition to maintenance.
- (2) In lieu of any designated senior reactor operator (SRO), the authority to release systems and equipment for maintenance or surveillance testing or return-to-service may be delegated to an on-shift SRO, provided provisions are made to ensure that the shift supervisor is kept fully informed of system status.
- (3) Except in cases of significant radiation exposure, a second qualified person should verify correct implementation or equipment control measures such as tagging of equipment.
- (4) Equipment control procedures should include assurance that control-room operators are informed of changes in equipment status and the effects of such changes.
- (5) For the return-to-service of equipment important to safety, a second qualified operator should verify proper systems alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

NOTE: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a qualified person. The staff is investigating the level of qualification necessary for other operators to perform these functions.

For plants that have or will have automatic system status monitoring as discussed in Task Action Plan Item I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities will be reduced. However, the need for such verification will not be eliminated in all instances.

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LILCO Position

The description contained under this heading in the latest revision of the USAR remains unchanged with the following exceptions:

1. The Watch Engineer;
 - a. gives permission to release plant systems or equipment for maintenance, surveillance tests or return to service.
 - b. must be informed of changed in equipment status and the effect of such changes.
 - c. provides final acceptance for return to service.
2. Considerations for shutdown margin and decay heat removal are not applicable.

I.C.7 NSSS Vendor Review of Procedures

NRC Position

Applicants for near-term operating licenses will be required to obtain NSSS vendor review of low-power and power-ascension test and emergency procedures (see Regulatory Guide 1.33, Appendix A, Section 6) as a further verification of the adequacy of the procedures. After trial use of this requirement on a few pending operating license applications, the staff will decide whether its further use or expansion to include procedure review by the A-E is desirable. This decision will be made in light of the long-term program described in Item I.C.9. See also Table C.1, Item 4a and Table C.3, Item 50 of NUREG-0660.

LILCO Position

Emergency procedures were prepared using the Emergency Procedures Guidelines developed by the Emergency Procedures Subgroup of the BWR Owners' Group and the General Electric Company.

Low-power and power-ascension test and emergency procedures will not be used at Shoreham.

I.C.8 Pilot Monitoring of Selected Emergency Procedures For Near-Term Operating License Applicants

NRC Position

Correct emergency procedures as necessary based on the NRC audit of selected plant emergency operating procedures (e.g., small break loss-of-coolant accident, loss of feedwater, restart of engineered safety features following a loss of ac power and steam-line break).

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LILCO Position

The generic guidelines prepared by the BWR Owners' Group and approved by the NRC for trial implementation at the Shoreham Nuclear Power Station have been incorporated into the Shoreham Emergency Operating Procedures (refer to NUREG-0737 Item I.C.1). The completed procedures have received an extensive in-house review, and were subjected to simulator verification. The verified procedures were submitted for NRC review. Any comments or corrections found necessary as a result of the NRC audit were evaluated and implemented, as appropriate.

This item was closed in Section 3.1.1 of Inspection Report 83-09.

I.D.1 Control Room Design Reviews

NRC Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed control-room design review to identify and correct design deficiencies. This detailed control-room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

NRR is presently developing human engineering guidelines to assist each licensee and applicant in performing detailed control-room review. A draft of the guidelines has been published for public comment as NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation." The due date for comments on this draft document was September 29, 1980. NRR will issue the final version of the guidelines as NUREG-0700, by February 1981, after receiving, reviewing, and incorporating substantive public comments from operating reactor licensees, applicants for operating licenses, human factors engineering experts, and other interested parties. NRR will issue evaluation criteria, by July 1981, which will be used to judge the acceptability of the detailed reviews performed and the design modifications implemented. Applicants for operating licenses who will be unable to complete the detailed control-room design review prior to issuance of a license are required to perform a preliminary control-room design assessment to identify significant human factors problems. Applicants will find it of value to refer to the draft document NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation," in performing the preliminary assessment. NRR will evaluate the applicants' preliminary assessments including the performance by NRR of onsite review/

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audit. The NRR onsite review/audit will be on a schedule consistent with licensing needs and will emphasize the following aspects to the control room:

1. The adequacy of information presented to the operator to reflect plant status for normal operation, anticipated operational occurrences, and accident conditions;
2. The groupings of displays and the layout of panels;
3. Improvements in the safety monitoring and human factors enhancement of controls and control displays;
4. The communications from the control room to points outside the control room, such as the onsite technical support center, remote shutdown panel, offsite telephone lines, and to other areas within the plant for normal and emergency operation.
5. The use of direct rather than derived signals for the presentation of process and safety information to the operator;
6. The operability of the plant from the control room with multiple failures of nonsafety-grade and nonseismic systems;
7. The adequacy of operating procedures and operator training with respect to limitations of the instrumentation displays in the control room;
8. The categorization of alarms, with unique definition of safety alarms.
9. The physical location of the shift supervisor's office either adjacent to or within the control-room complex.

Prior to the onsite review/audit, NRR will require a copy of the applicant's preliminary assessment and additional information which will be used in formulating the details of the onsite review/audit.

LILCO Position

LILCO has performed the required preliminary design assessment of the Shoreham control room and remote shutdown panel. The intent of the review is to identify significant human factors and instrument problems and to establish a schedule for correcting any deficiencies.

LILCO retained the General Physics Corporation, as a human factors consultant, to perform the preliminary assessment. General Physics has been involved in similar audits at the following near-term operating license BWR's: LaSalle, Susquehanna, and Zimmer.

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The criteria and checklists used at Shoreham considered the draft NUREG/CR-1580 "Human Engineering Guide to Control Room Evaluation" and the BWR Owners' Group Control Room Human Factor Review guidelines.

The preliminary assessment report detailing the resulting findings and a schedule for correcting deficiencies was submitted to the NRC March 12, 1981, SNRC-536.

NRC performed a five day onsite review/audit of the Shoreham control room beginning March 30, 1981. A final report of their findings (CRDR/A report) was issued May 10, 1981.

A Detailed Control Room Design Review (DCRDR) has been performed. The items identified by this review have been placed on hold due to the LILCO and New York State Shoreham settlement.

I.D.2 Plant Safety Parameter Display Console

NRC Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

These requirements are defined in NUREG-0696 which was published in February 1981.

LILCO Position

Since the Shoreham Nuclear Power Station will not be operating and because its reactor is defueled, the SPDS is not needed.

I.G.1 Training Requirements During Low Power Testing

NRC Position

Define and commit to a special low power testing program approved by the NRC to be conducted at power levels no greater than five percent to obtain additional technical information and supplemental training.

LILCO Position

A low power test program will not be conducted. This item is not applicable to LILCO.

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II.B.1 Reactor Coolant System Vents

This system is specific to the reactor coolant system and is not needed to support the storage of the fuel in the fuel pool.

II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation

NRC Position

The NRC position on the above is as given in the Shoreham USAR.

LILCO Position

In a fashion similar to the Shoreham USAR, LILCO has determined that the only areas after an accident where access may be needed are the Radwaste and Main Control Rooms and the Technical Support Center (TSC) full-time, and the Reactor Building refueling deck part-time. The basis of this is that, as seen in Chapters 11 and 15 of the DSAR, the only design basis events which remain credible are the Fuel Handling Accident (FHA) as described in Regulatory Guide 1.25, and the Liquid Radwaste Tank Rupture Accident. As discussed in Chapters 15 and 11, respectively, neither of these involve the release of large quantities of radioactivity, and thus there is no need for the Post Accident Sampling and Analysis Facility (PASF). The other areas suggested as vital post accident in NUREG-0737 do not apply for Shoreham or are not needed, for the reasons given in Section II.B.2 of the USAR. Furthermore, systems such as the hydrogen recombiner are clearly no longer required in the defueled condition, as documented in Chapter 6 of the DSAR.

Source Term and Results

The radioactive source terms (quantities and source distributions) for the design basis accidents are described in Section 11.2 for the Liquid Radwaste Tank Rupture Accident, and in Chapter 15.1.36 for the FHA (and Worst Case Fuel Damage Accident).

Because the cubicles where a Liquid Radwaste Tank Rupture Accident could occur are exhausted to a process air header, which is then processed through a filter train before release through the station vent, the postulated accident would have no affect on general Radwaste Building habitability. Specifically, the Radwaste Control Room would be unaffected, since it is well isolated from the cubicles where the accident could occur. For details, see Section 9.4.3 of the USAR.

For the FHA and Worst Case Fuel Damage Accidents, the habitability criteria for full occupancy of the Main Control Room and Technical Support Center are clearly not challenged. The 30-day integrated doses are:

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Control Room

	<u>Dose, mrem</u>	
	<u>Whole Body</u>	<u>Skin</u>
Fuel Handling Accident	9.59E-05	2.08E-01
Worst Case Fuel Damage Accident	5.92E-02	1.28E+02

Technical Support Center

	<u>Dose, mrem</u>	
	<u>Whole Body</u>	<u>Skin</u>
Fuel Handling Accident	5.02E-05	1.04E-01
Worst Case Fuel Damage	3.10E-02	6.42E+01

For any fuel damage accident with the spent fuel in the pool, clearly the highest dose rate area in the plant will be the refueling floor. With the Reactor Building Normal Ventilation System (RBNVS) operating throughout the event, airborne concentrations and associated dose rates are quickly dissipated in the Reactor Building. The time-dependent dose rates are as follows:

Fuel Handling Accident

	<u>Dose Rate, mrem/hr</u>	
<u>Time After Accident, hrs</u>	<u>Whole Body</u>	<u>Skin</u>
0	7.53E-02	7.11E+00
2	3.16E-04	2.98E-02
8	2.32E-11	2.20E-09

Worst Case Fuel Damage Accident

	<u>Dose Rate, mrem/hr</u>	
<u>Time After Accident, hrs</u>	<u>Whole Body</u>	<u>Skin</u>
0	4.65E+01	4.39E+03
2	1.95E-01	1.84E+01
8	1.43E-08	1.36E-06

Integrated 30-day doses, even assuming full occupancy for the Reactor Building refueling deck area, are as follows:

	<u>Dose, mrem</u>	
	<u>Whole Body</u>	<u>Skin</u>
Fuel Handling Accident	2.75E-02	2.60E+00
Worst Case Fuel Damage	1.70E+01	1.60E+03

The above are clearly of no concern for post-accident plant habitability.

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Harsh Environment

Due to the lack of safety-related equipment in the Radwaste Building, there are no harsh environment concerns there. Neither the Fuel Handling nor the Worst Case Fuel Damage Accident involve the release of any meaningful quantity of heat energy or chemically hazardous material. Furthermore, the gamma and beta doses given above are insignificant insofar as environmental qualification is concerned. As such, the credible accidents considered do not result in harsh environment in the Reactor Building or elsewhere.

II.B.3 Post-Accident Sampling

This system samples the reactor and containment and is not needed to support the safe storage of the fuel in the fuel pool.

II.B.4 Training for Mitigating Core Damage

This training is not needed to support the storage of the fuel in the fuel pool because it applies to an operating reactor.

II.B.7 Analysis of Hydrogen Control

This system is not needed to support the storage of the fuel in the fuel pool because the primary containment is not required for a defueled reactor.

II.D.1 Performance Testing of BWR and PWR Relief and Safety Valves

This system is not needed to support the storage of the fuel in the fuel pool because the reactor is defueled and unpressurized.

II.D.3 Relief and Safety Valves Position Indication

This system is not needed to support the storage of the fuel in the fuel pool. (See II.D.1)

II.E.4.1 Containment Dedicated Penetrations

This system is not needed to support the storage of the fuel in the fuel pool because the primary containment is not required.

II.E.4.2 Containment Isolation Dependability

This system is not needed to support the storage of the fuel in the fuel pool. (See item above)

II.F.1 Additional Accident Monitoring Instrumentation

Refer to Chapters 12 and 15 regarding instrumentation requirements for a defueled reactor.

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II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling

This item is specific to an operating reactor and is not needed to support the safe storage of the fuel in the fuel pool.

II.K.1.5 Safety Related Valve Position

These valves are not needed to support the storage of the fuel in the fuel pool.

II.K.1.10 Operating Status

This section applies without change.

(Plant Staff to confirm)

II.K.1.22 Auxiliary Heat Removal System Procedure

This procedure refers to an operating reactor and is not needed to support the storage of the fuel in the fuel pool.

II.K.1.23 RV Level, Procedures

The reactor vessel level procedures are not required.

II.K.3.3 Failure of Power Operated Relief Valve or Safety Valve to Close

This system is not needed because the reactor is not pressurized.

II.K.3.13 Separation of HPCI and RCIC System Initiation Levels - Analysis and Implementation

The HPCI and RCIC systems are not needed to support the storage of the fuel in the fuel pool.

II.K.3.15 Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems

This system is not needed to support the storage of the fuel in the fuel pool (See item above).

II.K.3.16 Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification

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

[NSSS will no longer challenge SRV's.]

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II.K.3.17 Report on Outage of ECC Systems -Licensee Report and Proposed Technical Specification Changes

This system is not needed to support the storage of the fuel in the fuel pool. Refer to Chapters 9 and 15 for system requirements.

II.K.3.18 Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences

This system is not needed to support the storage of the fuel in the fuel pool because the reactor is not pressurized.

II.K.3.21 Restart of Core Spray and LPCI Systems on Low Level

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

[CS and LPCI not required for defueled reactor.]

II.K.3.22 Automatic Switchover of Reactor Core Isolation Cooling System Suction - Verify Procedures and Modify Design

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

II.K.3.24 Confirm Adequacy of Space Cooling for High-Pressure Coolant Injection and Reactor Core Isolation Cooling Systems

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

II.K.3.25 Effect of Loss of AC Power on Pump Seals

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

[This issue deals with Reactor Recirculation Pumps which are no longer needed.]

II.K.3.27 Provide Common Reference Level for Vessel level Instrumentation

This system is not required for a defueled reactor.

II.K.3.28 Study and Verify Qualification of Accumulators on AD's Valves

This system is not needed because the reactor is unpressurized.

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II.K.3.30 Revised Small-Break LOCA Methods to Show Compliance with 10CFR50.46, Apperdix K

LOCA's are not possible for a defueled reactor.

II.K.3.31 Plant-Specific Calculations to Show Compliance with 10CFR50.46

This item is not needed to support the storage of the fuel in the fuel pool (See item above).

II.K.3.44 Evaluation or Anticipated Transients with Single Failure to Verify No Fuel Failure

Only the loss of Normal AC Power is applicable to the defueled plant configuration. See Table I. Without any inventory makeup to the pool the evaporation rate would be approximately .6 gpm. Chapter 15 demonstrates that there are no radiological consequences associated with this event.

II.K.3.45 Evaluation of Depressurization with Other Than Full ADS

This system is specific to operating reactor conditions and is not needed to support the storage of the fuel in the fuel pool.

II.K.3.46 Response to List of Concerns from ACRS Consultant (Mr. C. Michelson)

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

[These questions address concerns with protecting the fuel in the Vessel Core.]

III. EMERGENCY PREPAREDNESS

(Refer to LILCO Defueled Emergency Plan)

III.A.1.1 Upgrade Emergency Preparedness

NRC Position

The overall state of emergency preparedness for nuclear power plant accidents will be upgraded, including the integration of emergency preparedness onsite and offsite, according to the NRC/FEMA Memorandum of Understanding (item III.B). Approval of the overall state of preparedness will be required prior to issuance of an operating license. The review and upgrading for operating reactors is under way.

Six NRC teams were formed in September 1979 to implement the "Action Plan for Promptly Improving Emergency Preparedness" (SECY 79-450). That Action Plan identifies the elements required for

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promptly improving licensee emergency preparedness and for ensuring the capability of offsite agencies to take appropriate emergency actions. In the short term, the teams are making an integrated assessment of licensee, local, and State capabilities and interfaces based on: (a) a review of existing plans and a meeting in the site area to communicate upgraded criteria and to identify to licensees the areas requiring improvements. This includes an opportunity for expression of concerns by the public through an open meeting. An objective of the teams is to help improve working relationships and communications concerning emergency plan development among all parties. The criteria being used by the NRC teams reflect a number of the recommendations made as a result of the TMI-2 accident by the President's Commission and the NRC Special Inquiry Group; and (b) a review of upgraded licensee, local, and State plans submitted by the licensee after the site visit is summarized in a safety evaluation report. This includes an identification of areas requiring improvement, a schedule for implementation of the improvements, and a specification of any required interim measures. The review of upgraded plans encompasses the points in SECY-79-450 and reflects any input from the Federal Regional Advisory Committees (RAC). Items on local or State plans requiring improvement to meet the upgraded criteria of NUREG-0654 but which are adequate to meet the essential planning elements of "NRC Guide and Checklist," NUREG-75/111, and Supplement 1 thereto, are not being required for issuance of licenses for low-power testing.

The above actions are in progress and will be completed in FY 1980. In the longer term, beginning in FY 1981, an integrated assessment of the implementation of the plans will be performed. This assessment will take into account comments and reviews by the RAC as a result of State plan concurrence efforts, including critiques of emergency exercises. The results of the Office of Inspection and Enforcement (IE) special team efforts to evaluate Licensee health physics programs during 1980-81 will be factored into the review. This longer term review of emergency preparedness will consist of three parts: (a) a review of implementing procedures, including inplant and offsite personnel and equipment. The review of these procedures will be done by the team. Subsequently, periodic reviews and inspections will be performed by IE; (b) observing and critiquing exercises involving licensee, local, and State capabilities; and (c) observing and critiquing exercises involving licensee, local, State and Federal capabilities. For new operating license applicants, this must be completed before full-power licensing and within about five years for operating reactors.

NRR has sent letters to operating reactors, operating license applicants, and holders of construction permits requesting information regarding time estimates for evacuation of areas around plants to determine the difficulty of implementing protective measures for the public.

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LILCO Position

Refer to the emergency plan for the Shoreham site which is being submitted for review and approval as a separate document entitled, Defueled Emergency Preparedness Plan, via letter SNRC-1651. The information contained in this document supersedes in its entirety the information originally submitted as part of the FSAR.

III.A.1.2 Upgrade License Emergency Response Facilities

NRC Position

Each operating nuclear power plant shall maintain an onsite Technical Support Center (TSC) separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and locations of the TSC. Records that pertain to the asbuilt conditions and layout of structures, systems, and components shall be readily available to personnel in the TSC.

An Operational Support Center (OSC) shall be established separate from the control room and other emergency response facilities as a place where operations support personnel can assemble and report in an emergency situation to receive instructions from the operating staff. Communications shall be provided between the OSC, TSC, EOF, and control room.

An Emergency Operations Facility (EOF) (Near-Site) will be operated by the licensee for continued evaluation and coordination of all licensee activities related to an emergency having or potentially having environmental consequences. The EOF shall be located within 20 miles of the TSC to permit periodic face-to-face communication between management personnel in the TSC and the EOF. The EOF structure shall be well engineered for the design life of the plant. If the EOF is located within 20 miles of the TSC it shall have an isolatable ventilation system with HEPA filters and a backup EOF shall be located within from 10 to 20 miles of the TSC. If the EOF is located between 10 and 20 miles of the TSC, no isolatable ventilation system or backup EOF is required. The facility will have sufficient space to accommodate representatives from Federal, State and local governments as appropriate. In addition, the major State and local response agencies may provide for data analysis jointly with the operator at this location. The EOF will provide information needed by Federal, State, and local authorities for implementation of offsite emergency plans in addition to a centralized meeting location for key representatives from the agencies. Recovery operations shall be managed from this facility. Press facilities also may be available at the EOF.

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LILCO Position

The "Permanent Emergency Response Facilities Design Criteria and Description" applicable to this DSAR can be found in the Defueled Emergency Preparedness Plan which is being submitted separately via SNRC-1651.

The facilities changes principally involve:

1. The elimination of the EOF.
2. Moving the ENC to the Corporate Information Department in Hicksville.
3. The following information regarding TSC habitability:

As presently designed, the SNPS TSC meets the habitability criteria of GDC 19, for all credible design basis accidents (DBAs) envisioned with the fuel in the pool. The DBA on which this conclusion is based is the fuel handling accident (FHA), as described in detail in DSAR Section 15.1.36. Because the accident releases are so low, it is (conservatively) assumed that the TSC's HVAC system is not isolated (i.e., 7000 cfm of unfiltered intake and exhaust continues throughout the accident). A conservative, ground-level X/Q to the TSC intake is assumed, $7.86\text{E-}04$ seconds per cubic meter.

Whole body gamma and beta doses are due to Kr-85. The gamma doses are computed based on a finite cloud model in the TSC, plus a semi-infinite cloud surrounding the building, which has 18 inches of concrete shielding all around. The beta doses are based on the semi-infinite cloud model suggested by the NRC, Reg. Guide 1.3. The only radioiodine determined to be in SNPS' core is I-129, with an inventory of approximately 4 millicuries. Thyroid doses are computed using a conversion factor for I-129 derived in a fashion consistent with Reg. Guide 1.109 rev. 1, and a breathing rate of $3.47\text{E-}04$ cubic meters per second (1.25 cubic meters per hour).

The resulting doses, and the associated GDC 19 Criteria, are as follows:

	Dose, rem		
	Whole Body Gamma	Beta	Thyroid
Results	$5.02\text{E-}08$	$1.04\text{E-}04$	$1.21\text{E-}07$
GDC 19 Criteria	25	300	300

III.A.2 Improving Licensee Emergency Preparedness --Long-Term

NRC Position

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures

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can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement are delineated in NUREG-0654 (FEMA-REP-1). "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants".

In accordance with Task Action Plan item III.A.1.1, "Upgrade Emergency Preparedness," each nuclear power facility was required to immediately upgrade its emergency plans with criteria provided October 10, 1979, as revised by NUREG-0654 (FEMA-REP-1, issued for interim use and comment, January 1980). New plans were submitted by January 1, 1980, using the October 10, 1979 criteria. Reviews were started on the upgraded plans using NUREG-0654. Concomitant to these actions, amendments, were developed to 10CFR part 50 and Appendix E to 10CFR Part 50, to provide the long-term implementation requirements. These new rules were issued in the Federal Register on August 19, 1980, with an effective date of November 3, 1980. The revised rules delineate requirements for emergency preparedness at nuclear reactor facilities.

NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," provides detailed items to be included in the upgraded emergency plans and, along with the revised rules, provides the meteorological criteria, means for providing for a prompt notification to the population, and the need for emergency response facilities (see Item III.A.1.2).

Implementation of the new rules levied the requirement for the licensee to provide procedures implementing the upgraded emergency plans to the NRC for review. Publication of Revision 1 to NUREG-0654 (FEMA-REP-1), which incorporates the many public comments received is expected in October 1980. This is the document that will be used by NRC and FEMA in their evaluation of emergency plans submitted in accordance with the new NRC rules.

NUREG-0654, Revision 1; NUREG-0696, "Functional Criteria for Emergency Response Facilities;" and the amendments to 10CFR Part 50 and Appendix E to 10CFR Part 50 regarding emergency preparedness, provide more detailed criteria for emergency plans, design, and functional criteria for emergency response facilities and establish firm dates for submission of upgraded emergency plans for installation of prompt notification systems. These revised criteria and rules supersede previous Commission guidance for the upgrading of emergency preparedness at nuclear power facilities.

LILCO Position

Refer to the emergency plan for the Shoreham Site which is being submitted as a separate document entitled, "Defueled Emergency Preparedness Plan", via letter SNRC-1651. The information contained in this document supersedes in its entirety the information originally submitted as part of the USAR.

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III.D.1.1 Primary Coolant Sources Outside the Containment Structure

NRC Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

Immediate leak reduction

- (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.
- (b) Measure actual leakage rates with system in operation and report them to the NRC.

Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

Systems that should be leak tested are as follows (any other plants system which has similar functions or postaccident characteristics even though not specified herein, should be included):

Residual heat removal (RHR)

Containment spray recirculation

High-pressure injection recirculation

Containment and primary coolant sampling

Reactor core isolation cooling

Makeup and letdown (PWR's only)

Waste gas (includes headers and cover gas system outside of containment in addition to decay or storage system)

Include a list of systems containing radioactive materials which are excluded from program and provide justification for exclusion.

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Testing of gaseous systems should include helium leak detection or equivalent testing methods.

Should consider program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to all operating nuclear power plants regarding North Anna and related incidents, dated October 17, 1979.

LILCO Position

The purpose of this program is to minimize leakage of primary coolant sources outside of the primary containment. Since in the FIPS condition there is no "primary coolant" per se, the leakage prevention program becomes irrelevant and unnecessary.

III.D.3.3 In-plant Radiation Monitoring

NRC Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Effective monitoring or increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

- a. The physical size of the auxiliary and/or fuel handling building precludes locating stationery monitoring instrumentation at all areas where airborne iodine concentration data might be required.
- b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.
- c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.
- d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of

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noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

LILCO Position

Revise to state "The source term calculations for spent fuel (discussed in DSAR Section 12.2.1) indicate only a very small amount of iodine is present in the fuel, about 4.0 millicuries of I-129 (total core). There is no measurable quantity of radioiodine elsewhere in the Reactor, Radwaste, or Turbine Buildings. As such, inplant measurement of radioiodine during and after an accident is unnecessary, and no provisions are made to perform such analyses."

III.D.3.4 Control Room Habitability

NRC Position

In accordance with the Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50).

All licensees must make submittal to the NRC regardless of whether or not they meet the criteria of the referenced Standard Review Plans (SRP) sections. The new clarification specifies that licensees that meet the criteria of the SRP's should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan 2.2.1.2.2.2 Identification of Potential Hazards in Site Vicinity, 2.2.3 Evaluation of Potential Accidents, and 6.4 Habitability Systems, shall report their findings regarding the specific SRP Sections as explained below.

The following documents should be used for guidance:

1. Regulatory Guide 1.78, "Assumptions for evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release", June 1974.
2. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and,

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3. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19, "13th AEC Air Cleaning Conference, August 1974.
4. NUREG- 0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release.

In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should however ensure that submittals reflect the current facility design and that the information requested in Attachment 1 is provided. All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references shall perform the necessary evaluations and identify appropriate modifications.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive, material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident (LOCA) containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i.e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems. Other DBAS should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA.

In addition to the accident-analysis results, which should either identify the possible need for control room modifications or provide assurance that habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

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ATTACHMENT 1 INFORMATION REQUIRED FOR CONTROL-ROOM HABITABILITY EVALUATION

- (1) Control-room mode of operation, i.e., pressurization and filter recirculation for radiological accident isolation or chlorine release
- (2) Control-room characteristics
 - (a) air volume control room
 - (b) control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.)
 - (c) control-room ventilation system schematic with normal and emergency air-flow rates
 - (d) infiltration leakage rate
 - (e) high efficiency particulate air (HEPA) filter and charcoal adsorber efficiencies
 - (f) closest distance between containment and air intake
 - (g) layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions
 - (h) control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.
 - (i) automatic isolation capability-damper closing time, damper leakage and area
 - (j) chlorine detectors or toxic gas (local or remote)
 - (k) self-contained breathing apparatus availability (number)
 - (l) bottled air supply (hours supply)
 - (m) emergency food and potable water supply (how many days and how many people)
 - (n) control-room personnel capacity (normal and emergency)
 - (o) potassium iodide drug supply
- (3) Onsite storage of chlorine and other hazardous chemicals
 - (a) total amount and size of container
 - (b) closest distance from control-room air intake

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- (4) Offsite manufacturing, storage, or transportation facilities, of hazardous chemicals
 - (a) identify facilities within a 5-mile radius
 - (b) distance from control room
 - (c) quantity of hazardous chemicals in one container
 - (d) frequency of hazardous chemical transportation traffic (truck, rail, and barge)
- (5) Technical specifications (refer to standard technical specifications).
 - (a) chlorine detection system
 - (b) control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8 in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

LILCO Position

Habitability Systems Final Decision

Design Bases

The original plant design bases of the control room's habitability systems, as described in USAR Section III.D.3.4, still apply in general. However, due to the small quantity of radioactivity released during the design basis accident (the fuel handling accident), the control room's remote intakes and standby charcoal filtration system are no longer required to meet General Design Criteria 19. Doses, assuming the control room's HVAC system continues to function as during normal operation, are indicated in Chapter 15 of the DSAR.

System Design

The design of the control room HVAC system is as described in DSAR Section 9.4.1. As stated above, the remote intakes and the standby filtration system are no longer required. Inlet duct instrumentation is no longer required as well. As such, discussion of these items in USAR Section III.D.3.4 no longer applies. During the design basis accident, the control room HVAC system will continue to function as during normal plant operations.

Design Evaluation

This section is as described in USAR Section III.D.3.4, except that the remote intakes and standby filtration systems are no

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longer required. Also, as per DSAR Section 9.2.9, only one chilled water system is required with the spent fuel in the pool.

Tests and Inspections

Tests and inspections of the control room HVAC system are as described in USAR Section III.D.3.4, except that the standby filtration systems are no longer required.

Standby Charcoal Filtration Trains

This equipment is no longer required.