UNITED STATES NUCLEAR REGULATORY COMMISSION

LONG ISLAND POWER AUTHORITY

SHOREHAM NUCLEAR POWER STATION, UNIT 1 (SNPS)

DOCKET NO. 50-322

NOTICE OF ISSUANCE OF ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering the issuance of an order authorizing decommissioning for the Long Island Power Authority's (LIPA) Shoreham Nuclear Power Station, Unit 1 (SNPS). The Decommissioning Plan involves immediate dismantlement of the reactor pressure vessel and internals, contaminated systems, and plant structures (DECON).

Description of Proposed Action

On June 28, 1989, the Long Island Lighting Company (LILCO), the previous owner of SNPS, entered into an agreement with the state of New York. This settlement agreement between LILCO and New York State specifies that LILCO transfer ownership of SNPS to LIPA, an entity of New York State. LIPA will decommission SNPS. All spent fuel has been removed from the reactor and is stored in the SNPS Spent Fuel Pool. Approval of the Decommissioning Plan will allow immediate dismantlement of the reactor pressure vessel and internals, contaminated systems, and plant structures.

Environmental Impacts

The NRC staff has reviewed LIPA's proposed Decommissioning Plan, and Supplemental Environmental Report prepared in accordance with 10 CFR 51.53(b). To document its review, the staff has prepared an Environmental Assessment (EA). The proposed DECON of SNPS will allow unrestricted use of the remaining portion of SNPS sooner than SAFSTOR, but will result in some additional exposure for workers because of exposure during the immediate dismantlement operations.

Finding of No Significant Impact

The staff has reviewed the proposed decommissioning relative to the requirements set forth in 10 CFR Part 51. Based upon the Environmental Assessment, the staff concluded that there are no significant environmental impacts associated with the proposed action and that the proposed action will not have a significant effect on the quality of the human environment. Therefore, the Commission has determined, pursuant to 10 CFR 51.31, not to prepare an environmental impact statement.

For further details with respect to this action, see: (1) the licensee's application for authorization to decommission the facility, dated January 2, 1991, as supplemented August 26, November 27, and December 6, 1991; and (2) the Environmental Assessment and Finding of No Significant Impact. These documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555, and at the Shoreham-Wading River Public Library, Route 25A, Shoreham, New York 11786-9697. Copies may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects - III/IV/V.

Dated at Rockville, Maryland this 5th day of June 1992.

FOR THE NUCLEAR REGULATORY COMMISSION

Seymour H. Weiss

Seymour H. Weiss, Director
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT
BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
RELATED TO THE ORDER AUTHORIZING FACILITY DECOMMISSIONING
OF
LONG ISLAND POWER AUTHORITY (LIPA)
SHOREHAM NUCLEAR POWER STATION, UNIT 1
DOCKET NO. 50-322

JUNE 1992

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1.0 INTRODUCTION

1.1 Background

The Shoreham Nuclear Power Station is located in the Town of Brookhaven, Suffolk County, New York, about 50 miles east of New York City on the north shore of Long Island, as shown in Figure 1. The Shoreham plant as described in the Updated Safety Analysis Report (USAR) [Ref. 1], consists of a boiling water reactor (BWR) Nuclear Steam Supply System (NSSS), and a turbine generator, both supplied by General Electric (GE). The reactor has a design core thermal power rating of 2436 megawatts. The U.S. Nuclear Regulatory Commission (NRC) on July 3, 1985, granted the Long Island Lighting Company (LILCO) a low-power operating license to operate the Shoreham plant at power levels not to exceed 5 percent of rated power. A full power operating license (NPF-82) authorizing LILCO to operate the Shoreham plant at full power was granted on April 21, 1989.

The Shoreham Nuclear Power Plant operated intermittently at low power levels during the period July 1985 through June 1987. At the time of the plant's final shutdown (June 1987), the average fuel burnup was calculated to be approximately 2 effective full-power days.

The State of New York and LILCO entered into a Settlement Agreement (February 28, 1989) with the Long Island Power Authority (LIPA), under which LILCO agreed not to operate the Shoreham plant as a nuclear facility. The settlement also transferred the plant and specific areas and buildings on the Shoreham site to LIPA. LIPA is a corporate and political subdivision of the State of New York, created by New York State statute. LIPA was authorized by the LIPA Act to acquire the Shoreham Nuclear Power Plant, and upon acquisition of the plant, LIPA is required to close and decommission the Shoreham Nuclear Power Plant.

LILCO reiterated its agreement not to operate the Shoreham plant as a nuclear facility in an Assets Transfer Agreement with LIPA on April 14, 1989. The LILCO and LIPA Settlement became final on June 28, 1989, when LILCO shareholders voted to approve the Assets Transfer Agreement. Fuel removal from the reactor was completed in August 1989, and by Confirmatory Order, dated March 29, 1990, the Shoreham license was modified such that fuel could not be reloaded in the reactor without prior NRC approval.

LIPA and LILCO on June 28, 1990, jointly submitted a license transfer amendment request to NRC, requesting that LILCO's Shoreham license be amended to authorize the Shoreham license to be transferred to LIPA. By letter (R.M. Kessel to T.E. Murley) [Ref. 2] dated December 29, 1990, LIPA submitted to NRC the Shoreham Decommissioning Plan (DP), and the Supplement to Environmental Report (Decommissioning) [Refs. 3 and 4].

On July 19, 1991, license Amendment No. 7 to Facility Operating License No. NPF-82 became effective. This amendment removed the licensee's authority to operate the Shoreham facility, and modified the license from a full-power

SHOREHAM NUCLEAR POWER STATION, UNIT
GENERAL LOCATION

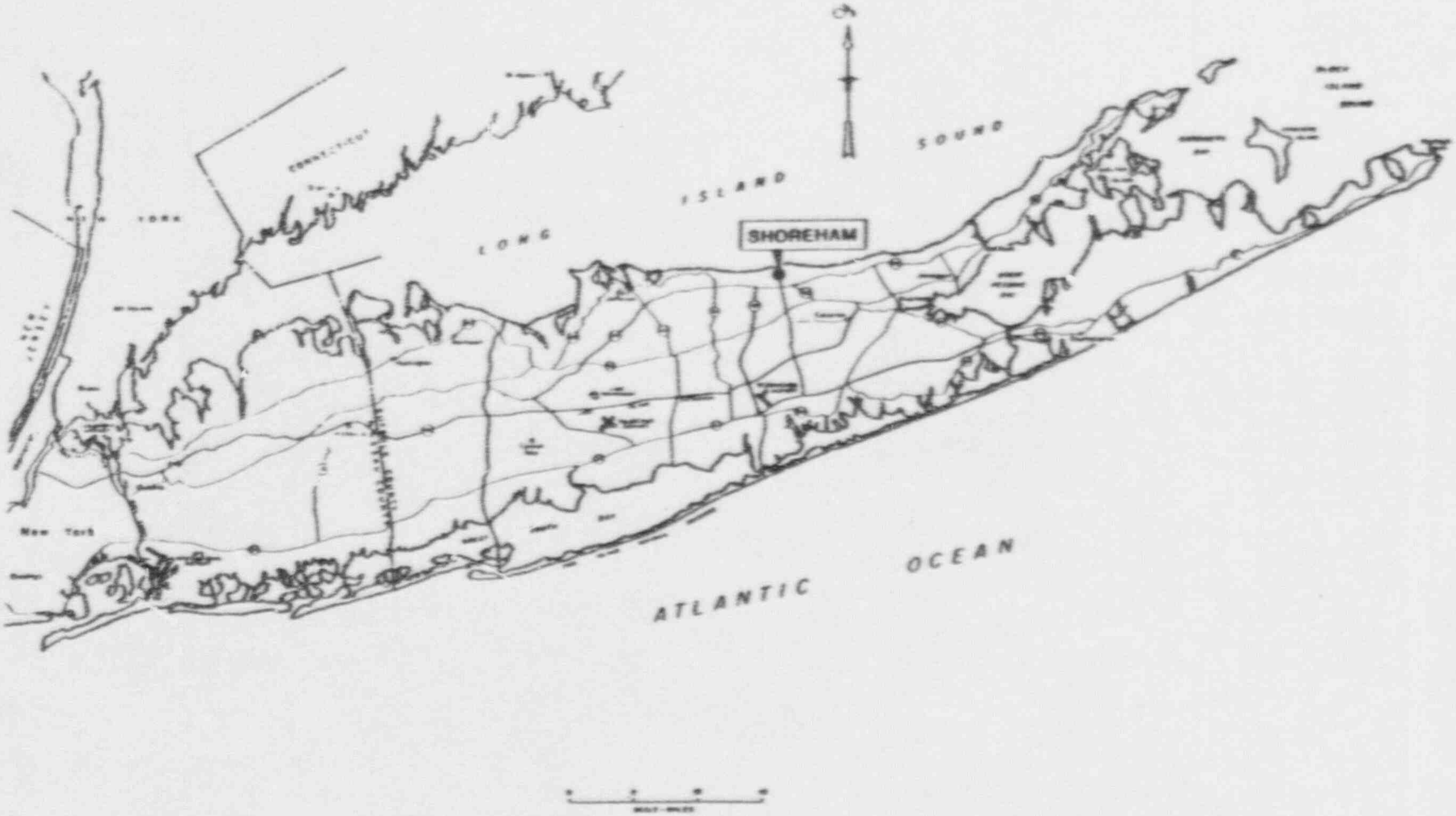


Figure 1

operating license to a Possession Only License (POL). On February 29, 1992, the NRC issued an order authorizing the transfer of the POL from LILCO to LIPA in accordance with the Commission decision in CLI-92-04, dated February 26, 1992.

The analyses in this Environmental Assessment (EA) are based on the Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, NUREG-0586, August 1988 (GEIS), and provides additional information and evaluations particularly relevant to the decommissioning of Shoreham.

1.2 Proposed Action

This EA addresses the proposed final decommissioning of the Shoreham Nuclear Power Station, Unit 1, providing for dismantlement and decontamination of the facility (DECON). LIPA's intentions are to dismantle systems and decontaminate structures to the extent necessary to ensure the removal of radioactive materials, and to irrevocably remove the plant from service as a nuclear generating facility.

Because of the short operating history of Shoreham, the contamination and activation levels are low when compared with a reactor that operated for the entire licensed period (i.e., the reference BWR in NUREG/CR-0672, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station" [Ref. 5]). Because of the limited contamination levels at the plant, the DECON decommissioning alternative was selected by the licensee because it offers a number of advantages (i.e., early elimination of existing radiation hazards, near-term release of the site and equipment for unrestricted use, and added flexibility in selecting future use of the site).

Complete release of the Shoreham site for unrestricted use will depend on the disposal of all radioactive materials from the site at the completion of decommissioning. Because of the pending closure of the two offsite disposal sites available to the State of New York, and the loss of access to the third existing disposal site on December 31, 1992, there is a high probability that some decommissioning waste will require interim onsite storage at the Shoreham site.

Fuel disposal is not part of the decommissioning action at Shoreham; however, continued onsite storage of fuel will affect the DECON decommissioning alternative selected by the licensee. The licensee is proposing two fuel disposal options. The first and preferred option is to ship the fuel to another utility for use and/or storage (Nine Mile Point, Unit 2), and the second option is to ship the fuel for reprocessing in Europe (Additional Information in Support of the Decommissioning Plan for Shoreham (Working Meeting November 7-8, 1991)) [Ref. 6]. The DECON decommissioning alternative is not compatible with long term storage of irradiated fuel. Therefore, a time period equivalent to the 5 to 6 years for the DECON alternative noted in NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" [Ref. 7], will be imposed on the licensee, to have the fuel shipped offsite. If the irradiated fuel remains onsite beyond the period noted above, the licensee will be required to submit a modified decommissioning plan.

1.3 Needed Action

The proposed action (decommissioning) is necessary because the LILCO determined not to operate Shoreham. See GEIS; see also 10 CFR Part 50.82. Dismantlement and decontamination of plant systems and structures to conditions suitable for unrestricted release are the required results of the decommissioning action that the licensee is undertaking. The licensee has determined that it is to its advantage to proceed as soon as possible with decommissioning, using the DECON alternative. The advantages the licensee cited for proceeding with its proposed method of decommissioning are as follows:

a) maximization of the licensee's flexibility in selecting near future uses of the Shoreham site;

b) use of personnel who are knowledgeable about the Shoreham Nuclear Power Station and its operating history;

c) accomplishment of DECON decommissioning without significant impact from radiation exposure (because of the limited plant operation and limited contamination levels);

d) elimination of the need for long-term monitoring, security, surveillance and maintenance; and

e) the fact that the option causes no significant environmental impact.

2.0 DESCRIPTION OF THE SHOREHAM NUCLEAR POWER STATION

This chapter provides descriptive information on the physical plant, including relevant Shoreham plant history. A description of Shoreham has also been provided in the Final Environmental Impact Statement, NUREG-0285, October 1977, and should be included in connection with this description.

2.1 General Plant Description

The Shoreham Nuclear Power Station is located in the Town of Brookhaven, Suffolk County, New York, about 50 miles east of New York City on the north shore of Long Island. The entire Shoreham site consists of approximately 500 acres. Figure 2 depicts the Shoreham Nuclear Power Station site plan, showing the reactor building and other major structures. The decommissioning "Project Area" consists of approximately 18 acres, 11 of which were transferred to LIPA by LILCO. The "Project Area" is the property to be used by LIPA during the decommissioning, and its boundaries are shown in Figure 3. Areas outside the property transferred to LIPA for decommissioning activities include existing parking lots, laydown areas, warehouses, and other ancillary structures.

SHOREHAM NUCLEAR POWER STATION
SITE PLAN

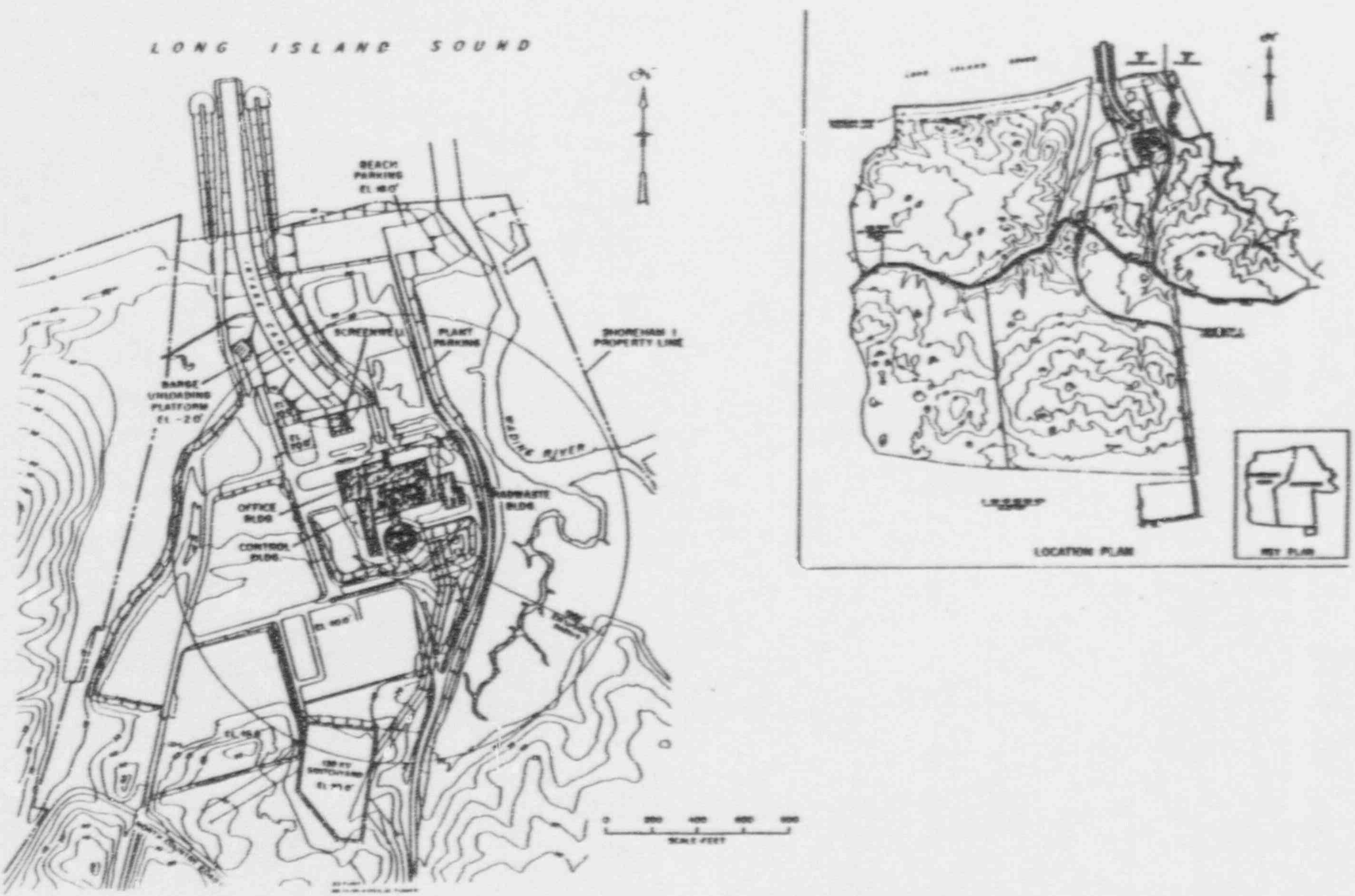


Figure 2

SHOREHAM NUCLEAR POWER STATION
PROJECT AREA

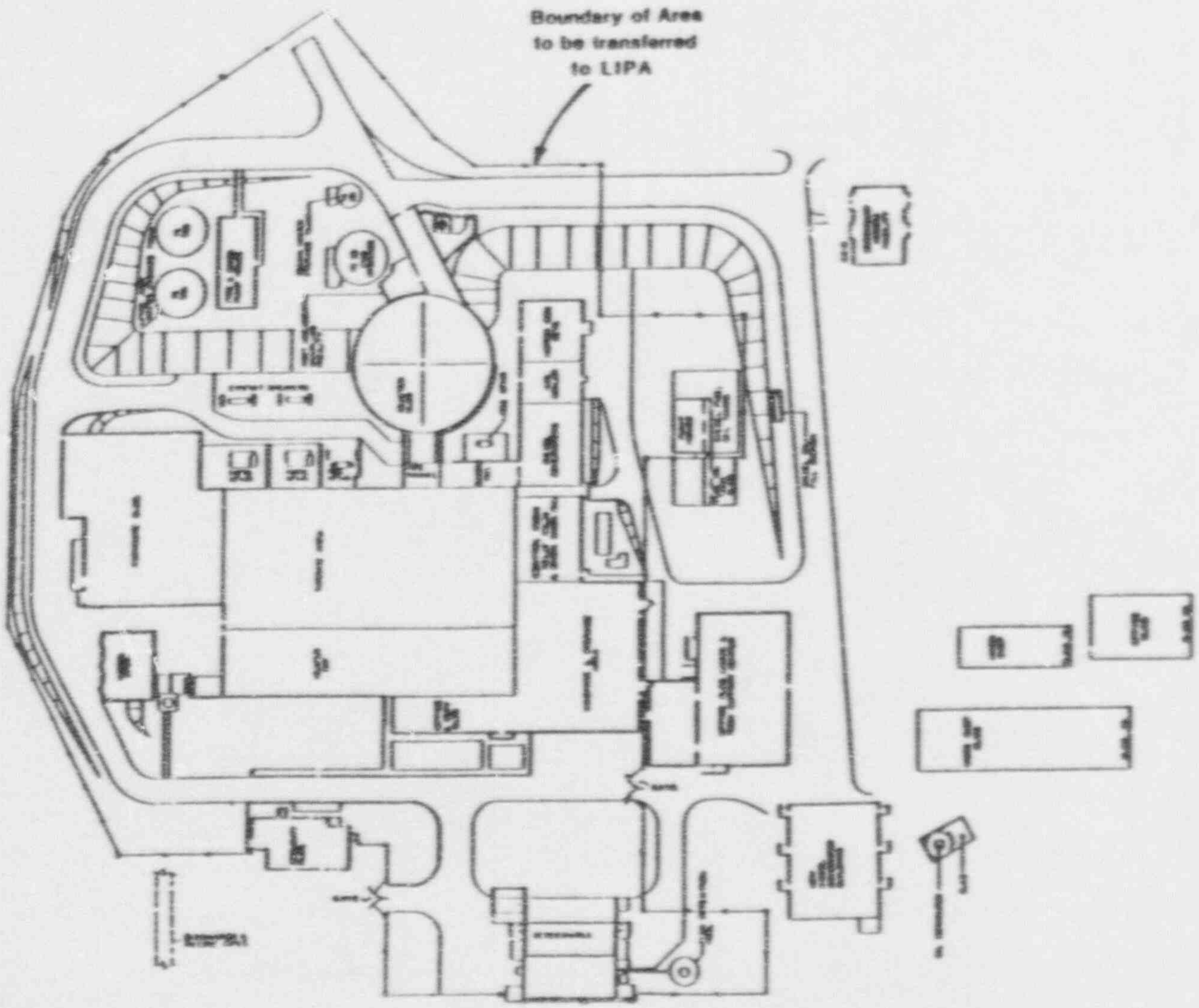


Figure 3

REACTOR BUILDING

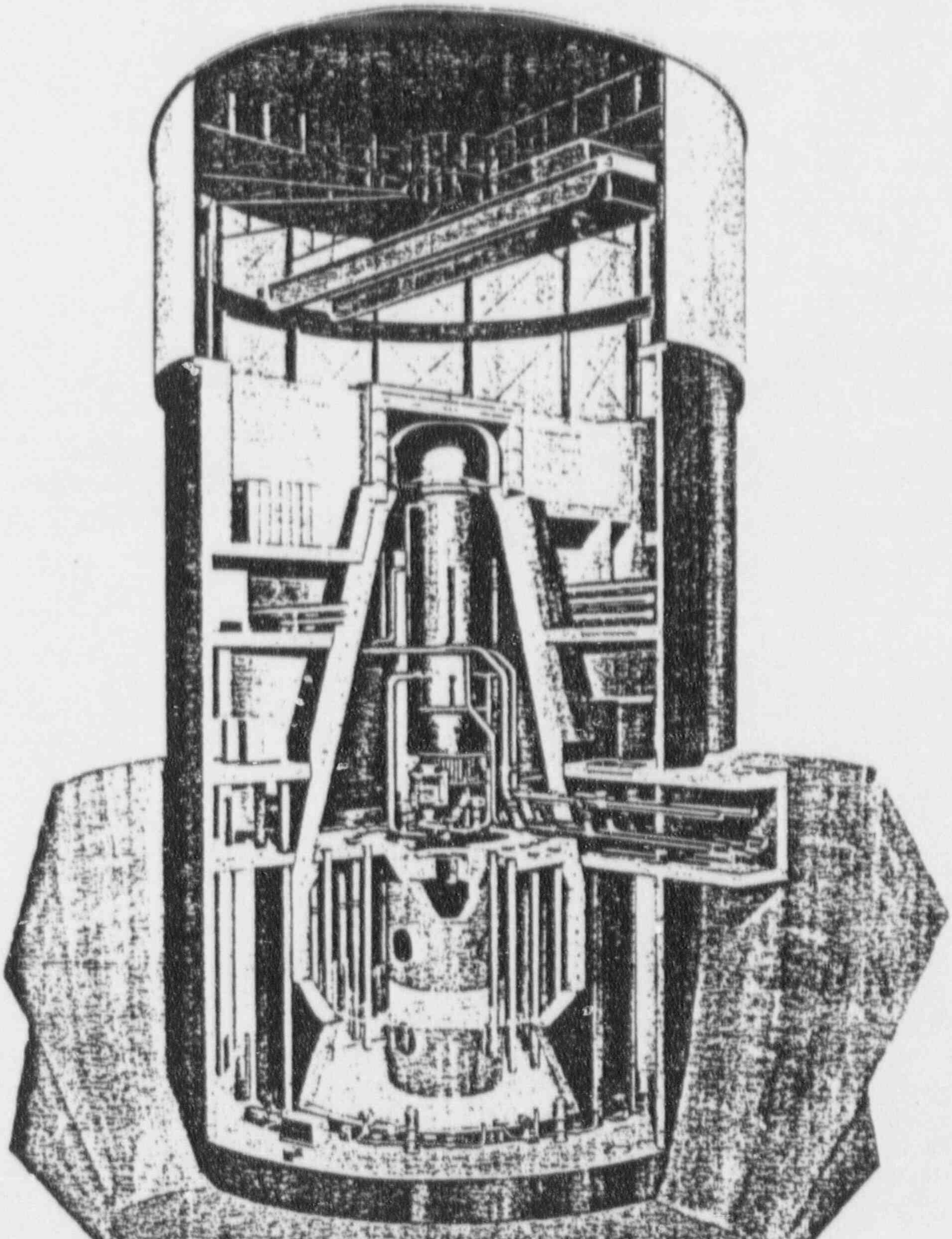


Figure 4

SHOREHAM NUCLEAR POWER STATION REACTOR PRESSURE VESSEL AND INTENALS

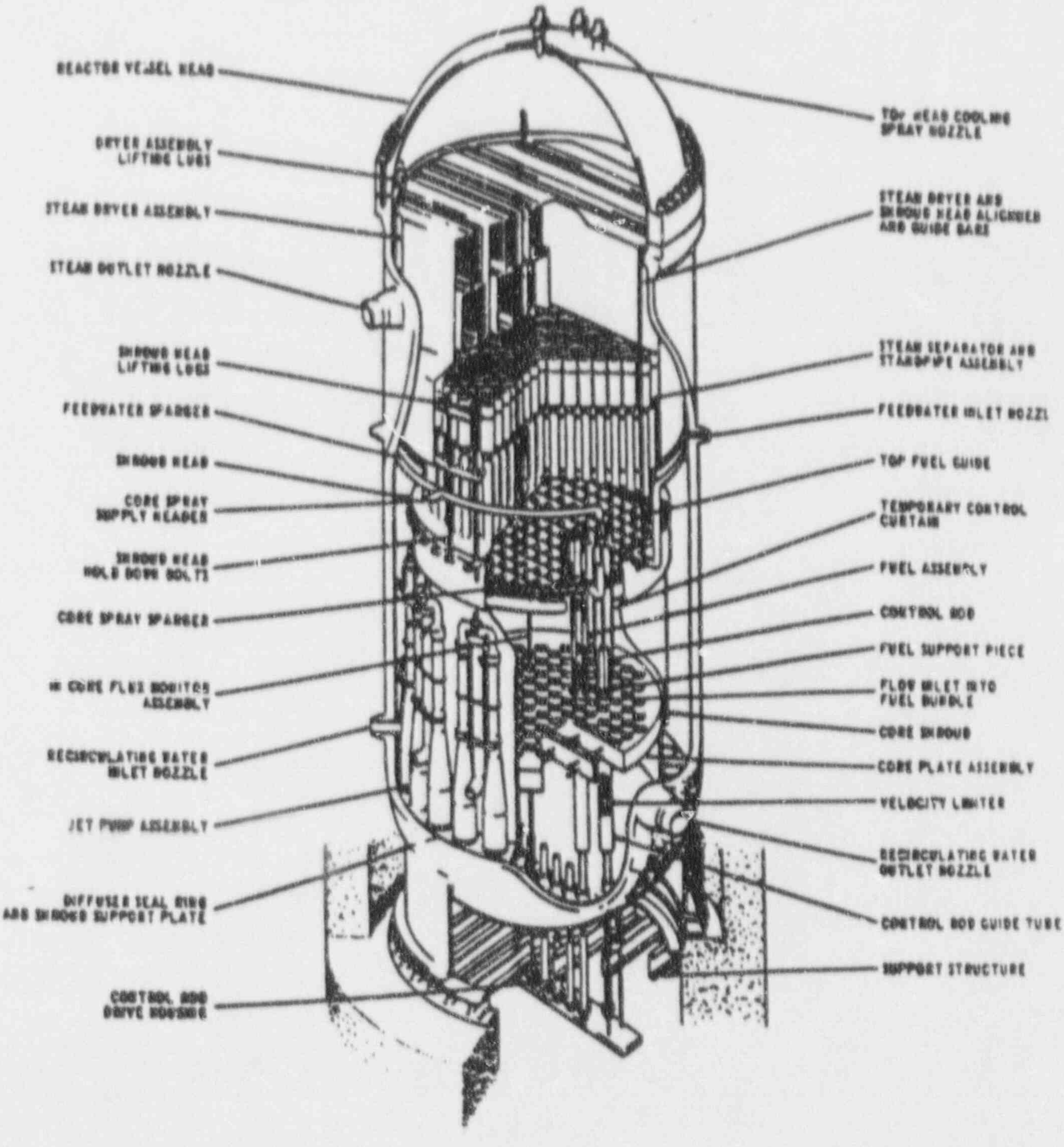


Figure 5

decommissioning and its boundaries are shown in Figure 3. Areas outside the property transferred to LIPA for decommissioning activities include existing parking lots, laydown areas, warehouses, and other ancillary structures.

As described in the Shoreham USAR [Ref. 1], the plant is a BWR design. Figure 4 displays the primary and secondary containments for the Shoreham Mark II containment design. The Shoreham Reactor Pressure Vessel (RPV) is shown in Figure 5. The Reactor Building contains nearly all of the contaminated systems and structures that are to be dismantled and decontaminated during the decommissioning. The contaminated systems and structures include the RPV, system piping, floor drains, and sumps.

Balance of Plant (BOP) systems, the main turbine, and main generator are housed in the Turbine Building. In the Turbine Building, only the drain sump is known to be contaminated.

A number of plant support systems such as the condensate demineralizers, the liquid radwaste system, the solid radwaste storage area, and portion of the gaseous radwaste system are housed in the Radwaste Building. The Radwaste Building contains only slightly contaminated structures and systems, such as floor drains and radwaste tanks.

2.2 Shoreham Nuclear Power Station Operating History

The Shoreham plant operated intermittently over a 2 year period (July 1985 through June 1987) in a startup testing mode. Final plant shutdown occurred in June 1987. The plant completed defueling in August 1989, when all 560 fuel assemblies were removed from the reactor and stored in the Spent Fuel Storage Pool. By Confirmatory Order dated March 29, 1990, the Shoreham license was modified so that LILCO could not reload fuel into the reactor without prior NRC approval.

The plant's fuel burnup was calculated to be approximately 2 effective full-power days, as of June 1990. It was estimated, at that time that the fuel contained approximately 176,000 curies of radioactivity [Ref. 3].

Table 1 summarizes relevant plant operating information for the Shoreham Nuclear Power Station.

Table 1

Shoreham Nuclear Power Station Descriptive and Licensing Information

Nuclear Unit	Shoreham Nuclear Power Station
NRC Docket Number	50-322
Location	Approximately 50 miles east of New York City on the north shore of Long Island Sound.
Reactor Type	BWR

Table 1 (Cont'd)

Capacity, MWt	2436
Capacity, MWe	849
Containment Type	Mark II
Cooling Source	Long Island Sound
NSSS	GE
BOP	Stone & Webster
Construction Permit	04-15-68
Low Power License	12-07-84 (0.001 percent)
Initial Fuel Loading	
o Began	12-21-84
o Completed	01-19-85
Initial Criticality	01-15-85
Operating License	07-03-85
Full Power License	04-21-89
POL	07-19-91
License Transfer to LIPA from LILCO	02-29-92

Liquid and gaseous releases of radioactive materials during plant operations were within regulatory limits (10 CFR Part 20, Appendix B, Table II, Columns 1 and 2), and releases to the environment were reported in the Shoreham Semi-Annual Effluent Reports (1985-1987) [Ref. 8]. During plant operations, there were no indications of fuel leakage, and there were no spills, releases of radioactive materials, or operational events that would have resulted in residual radioactive contamination that would adversely affect decommissioning.

On July 19, 1991, license Amendment No. 7 to Facility Operating License No. NPF-82 became effective. This amendment removed the licensee's authority to operate the Shoreham facility, and modified the license from a full-power operating license to a POL. On February 29, 1992, transfer of the POL from LILCO to LIPA became effective.

2.3 Current Radiological Conditions

The licensee conducted numerous surveys at Shoreham, in accordance with the requirements of 10 CFR Part 20.201. Radiological conditions at Shoreham are summarized by the licensee in the Shoreham DP [Ref. 3], and are based on data documented in the "Shoreham Nuclear Power Station Characterization Program, Final Report," May 1990; Addendum 1, June 1990; and Addendum 2, August 1990 [Ref. 9]. The data and analysis methods used are documented in the "Shoreham Site Characterization Program (SSCP)" [Ref. 10]. The SSCP divided the plant into the following four principal areas: a) RPV, b) systems, c) structures, and d) soil.

2.3.1 Plant Radioactivity Inventory

A summary of the plant's radioactive inventory and discussion, by area, follow. The calculated quantities of radioactivity in and on equipment at the Shoreham plant are summarized in Table 2.

Table 2

<u>Radioactive Inventory at Shoreham</u>

1)	RPV and its Components and Biological Shield (as of July 1990)		602 Ci
2)	Plant Systems	approx.	0.0033 Ci
3)	Plant Structures		<0.001 Ci
4)	Soil a)		b)
5)	Irradiated Fuel		176,000 Ci
6)	Control Rod Blades		960 Ci
7)	Local Power Range Monitors ("hot ends")		356 Ci
8)	Local Power Range Monitors ("cold end")		3.6 Ci
9)	Antimony Pins		30 Ci
10)	Beryllium Sleeves		20 Ci

a) Only 1 of 61 samples contained radioactivity in excess of background.

b) Measurable quantities of Cs-137 found in samples were determined to be consistent with background concentrations, which lead to the conclusion that no releases of radioactivity from Shoreham occurred.

Based on the record keeping requirements of 10 CFR 50.75(g)(1) that pertain to a safe and effective decommissioning, the licensee noted that there were no spills, releases of radioactive materials, or operational events reported that would result in residual radioactive contamination that could affect the decommissioning.

As noted in its response to NRC questions "Response to Request for Additional Information for Shoreham Decommissioning Plan," August 26, 1991 [Ref. 11], and based on the requirements of 10 CFR Part 50.75(g)(2), there are two inaccessible areas that were not surveyed during site characterization. The inaccessible areas identified are the drains in the Reactor Building, and the Spent Fuel Storage Pool. Based on plant history, there are no additional inaccessible areas that are suspected of being contaminated.

2.3.1.1 RPV

The licensee analyzed the RPV, reactor vessel internals, mirror insulation, and the biological shield wall for radioactive contamination. Radioactivity associated with the RPV is the result of direct neutron activation, and deposition of neutron activated contamination during plant operation. The licensee used the computer codes ORIGEN2 and RADCOR to determine the level of radioactivity due to activation. Samples taken inside the RPV did not show any fission product contamination.

The total radionuclide inventory for the RPV was calculated to be 602 Ci [Ref. 3]. Over 97 percent of the radioactive inventory is attributable to Fe-55 and Co-60. The remaining radioactivity is due to Ni-63. The core shroud (170 Ci total activity), top guide plate (334 Ci total activity), and Source Range Monitors and Intermediate Range Monitors (SRM/IRM) dry tubes (76 Ci total activity) contained over 96 percent of the radioactivity inventory [Ref. 9].

The RADCOR computer code was used to provide a curie estimate for the control rod blades. Based on the licensee's estimate, there are 960 Ci associated with the 138 control rod blades. An estimated 356 Ci were related to the hot end of the Local Power Range Monitors (LPRMs), and approximately 3.6 Ci were estimated distributed over the cold end of the LPRMs. Thirty Ci were estimated for the 20 antimony pins, and approximately 20 Ci for the 6 beryllium sleeves [Ref. 6].

2.3.1.2 Plant Systems

All systems within the Reactor, Turbine, and Radwaste buildings that could conceivably contain radioactivity were selected for radiological characterization. The systems listed in Table 3 have been identified as the systems containing radioactive materials.

Table 3

Contaminated Systems

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

The licensee estimated that the total radioactivity in the nine contaminated systems was approximately 33 millicuries [Ref. 3].

2.3.1.3 Plant Structures

The quantities of radioactivity on plant structures are based on radioactive materials detected on surfaces of the Reactor, Turbine, and Radwaste buildings. The total radioactivity on surfaces of contaminated plant structures was estimated to be less than 1 millicurie [Ref. 4]. Most of this radioactivity is located in the reactor head cavity, and in 6 sumps in the Reactor, Radwaste, and Turbine buildings.

2.3.1.4 Soil

The areas covered during soil sampling at Shoreham are located within a 1000-foot radius of the Reactor Building center line. The 1000-foot radius encompasses the entire protected area of the site. Only 1 of the 61 total samples analyzed contained radioactivity above background. This sample was taken from a marsh area, and the analysis revealed a small concentration of Cs-137. The marsh sample contained roots that concentrate elements such as cesium. The Cs-137 concentration of 909 pCi/kg-dry found in the marsh were attributed to fallout from atmospheric weapon testing and the Chernobyl accident. No detectable fission products were found in or on Shoreham structures, systems, or plant components [Ref 9]. Based on sample results, it was determined that there were no radionuclides in the Shoreham environ soil that are attributable to plant operations.

2.3.2 Plant Radiation and Contamination Levels

When compared with the reference BWR described in NUREG/CR-0672 [Ref. 5], the dose rates throughout the Shoreham plant are very low. General area dose rates in the Reactor Building (on the 175-foot elevation) are less than 0.5 mrem/hr [Ref. 3]. Inside the RPV, dose rates are in the range of 0.5 to 20 mrem/hr [Ref. 3] (dose rates at the vessel flange after shielding or after removal of activated intervals). The unshielded dose rates inside the RPV are expected to vary, based upon location, from 100 mrem/hr to 100R/hr [Ref. 3]. In the remainder of the Reactor, Turbine, and Control buildings, the dose rate is < 0.1 mrem/hr [Ref. 3]. The dose rate in the general areas of the Radwaste Building is 0.01 mrem/hr [Ref. 3].

The SSCP [Ref. 10] identifies systems, structures (interior and external surfaces of the Reactor, Radwaste, and Turbine buildings), RPV and Internals, and soil that is contaminated or has the potential to be contaminated. Contamination in systems was found to be Co-60 and Mn-54. The contamination found on structural surfaces tends to be greater than 95 percent Co-60, with the remainder Mn-54. Structural surface contamination tends to be fixed rather than easily removable [Ref. 9].

2.3.2.1 RPV and Internals

The RPV internals, and most of the vessel itself, the mirror insulation, and the biological shield are contaminated as a result of direct neutron activation, or deposition of neutron activated contamination. Based on isotopic analysis [Ref. 9] 97 percent of the contamination is due to Co-60 and to Mn-54. Table 4 lists each RPV component and its contamination level.

Table 4
Reactor Component Surface Contamination[a]

Component	Specific contamination (uCi/ft^2)
RPV	5.76×10^{-2}
Core Shroud	5.76×10^{-2}

Table 4 (Cont'd)

Jet Pump Assemblies	5.76×10^{-2}
Top Guide Plate	5.76×10^{-2}
Lower Core Support Plate	5.76×10^{-2}
Moisture Separators	3.12
Steam Dryers	0.996
SRM/IRM Dry Tubes	5.76×10^{-2}
Control Rod Drive Guide Tubes	5.76×10^{-2}

(a) The values in the table represent the average specific surface contamination levels.

2.3.2.2 Plant Systems

A total of 127 systems at Shoreham were initially considered for characterization. Of these 127 systems, 44 were characterized. Of the 44 systems characterized, 27 were included as a part of the site characterization program. The remaining 17 systems were either included as a part of the 27 systems characterized because they received fluids, gases, or solids from the 27 systems, or they were categorized as contaminated surfaces such as, local electrical control panels [Ref. 9].

Plant systems were considered contaminated if they met the following criteria:

1. systems or equipment that carried fluids or gases that may have circulated through the reactor vessel, radioactive waste systems, or the spent fuel storage system;

2. systems and equipment used to collect, circulate, or discharge air from the Reactor, Radwaste, or Turbine buildings;

3. systems or equipment used to collect or drain fluids from the Reactor, Radwaste, or Turbine buildings systems or floor drains; and

4. systems or equipment with indications of contamination, as determined from previous Radiation Work Permit (RWP) surveys and maintenance records, as well as gamma external exposure rate scans performed for the characterization program.

Nine of the 27 potentially contaminated systems were found to have contamination levels that range from non-detectable to approximately 255,000 beta dpm/100-sq-cm total surface contamination, and approximately 42,000 dpm/100-sq-cm removable contamination [Ref. 9]. The nine systems listed in Table 3, were found to have internal contamination levels in excess of the 1000 dpm/100-sq-cm removable contamination level, and the 5000 dpm/100-sq-cm

[Ref. 9] total surface contamination level specified in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" [Ref. 12].

The radionuclide contamination was determine to be primarily Co-60, with traces of other radionuclides. Table 5 provides a summary of the average surface contamination levels for the nine systems found to be contaminated.

Table 5

System Contamination Levels

Systems	Average Total Surface Contamination dpm/ 100 sq. cm	Regulatory Guide 1.86 Criterion dpm/100 sq. cm.
Reactor Recirculation	14,000	5,000
Control Rod Drive	8,000	
Residual Heat Removal	12,000	
Core Spray	47,000	
Liquid Radwaste	2,400	
Reactor Water Cleanup	28,000	
Fuel Pool Cleanup	26,000	
Condensate Demineralizer	6,000	
Process Sampling	12,000	

2.3.2.3 Plant Structures

The licensee conducted radiological surveys of the drywell, and the Reactor, Radwaste, Turbine buildings, and external surfaces immediately adjacent to these buildings. Structural characterization has shown that the majority of surfaces having contamination levels were well below the release limits in Regulatory Guide 1.86 [Ref. 12], and were found in general to be at natural background levels [Ref. 3]. The contamination that was found was determined to be highly localized. The contamination found on characterized structural surfaces tended to be greater than 95 percent Co-60, and the remainder was determined to be Mn-54 [Ref. 9]. Structural contamination tended to be fixed rather than removable. Table 6 is based on data provided in the Shoreham DP [Ref. 3], and summarizes the structural contamination levels.

Table 6

Structural Contamination

Structures	Average Total Surface Contamination (dpm/100 sq cm)	Maximum Total Surface Contamination (dpm/100 sq cm)
Primary Containment	<1,000	3,000
Contaminated Equipment/ Floor Drains and Sumps	5,000	11,000
Dryer/Separator Storage Pool (a)	<1,000	2,000
Reactor Head Cavity	9,000	78,000
Spent Fuel Storage Pool (b) Radwaste Laydown Area	11,000	55,000

Table 6 (Cont'd)

(a) Values shown are for pool walls. There were several inches of water in the pool at the time of site characterization.

(b) The Spent Fuel Storage Pool and Spent Fuel Storage Racks were assumed to be contaminated in excess of Regulatory Guide 1.86 limits [Ref. 12].

2.3.2.4 Soil

Soil samples were taken from areas within a 1000-foot radius of the Reactor Building center line. Within the 1000-foot radius, two types of sampling distributions were considered, unbiased and biased. "Unbiased" sample areas are somewhat randomly distributed within the area to be sampled, without regard to potential contamination spread, whereas "biased" sample locations are those for which there is at least a reasonable scenario or basis for suspecting that radiological contamination of soil exists.

Soil samples were characterized for 38 locations within a radius of 1000-feet of the Reactor Building, and for 23 locations inside the plant's protected area near building exits and outside tanks. Detailed selection criteria for the sampling locations selected are defined in the Site Characterization Program description, Revision 2 [Ref. 10]. No soil contamination attributable to Shoreham was found in soil samples.

3.0 DESCRIPTION OF THE SHOREHAM NUCLEAR POWER STATION ENVIRONMENT

3.1 Shoreham Site Description

The Shoreham Nuclear Power Station consist of a GE BWR-4, which has a rated thermal output of 2436 MWt. The reactor is designed to supply saturated steam to a GE tandem compound turbine/generator. A once-through cooling system supplied by water from Long Island Sound is used to condense turbine steam. The Shoreham Nuclear Power Plant environment is further described in the Final Environmental Statement on this facility, NUREG-0285, October 1977.

3.2 Site Layout

The Shoreham Nuclear Power Station is located on an approximately 500 acre site in the Town of Brookhaven, Suffolk County, New York, on the north shore of Long Island. The developed area includes the station and its structures, and consists of approximately 80 acres in the site's northern section. The developed area is bounded by Long Island Sound on the north, by wetlands on the east, by North County Road on the south, and by the Shoreham West property on the west [Ref. 3].

The nearest location accessible to the public is approximately 600-feet NE of the Reactor Building, and along Wading River Creek, east of the plant access road. The nearest accessible location on property not controlled by LILCO is a nature conservatory that adjoins the Shoreham site to the east, about 1100-feet from the Reactor Building. The nearest residence is located on the beach, about 1500-feet NE of the Reactor Building. Details related to the physical layout of the Shoreham site are provided in the DP [Ref. 3].

3.3 Climate

The description of the climate conditions on Long Island Sound are based on data provided in the Shoreham USAR [Ref. 1]. Despite the proximity of Long Island Sound and the Atlantic Ocean, the Shoreham site climatology is more continental than maritime in character. The surrounding water bodies exert a significant, although not a dominant, influence on the site climatology.

Temperatures above 90-degrees F occur occasionally during the late spring and summer, with an extreme maximum temperature of 101-degrees F. Normal daily summer temperatures average about 69-degrees F, with daily maximum temperatures averaging 79-degrees F. Winters are generally cold but the relatively warm ocean water tends to modify outbreaks of cold Arctic air. Subzero temperatures may occur on a few days during a given winter, whereas normal daily winter temperatures and normal daily maximum winter temperatures average about 31-degrees F and 40-degrees F, respectively.

Precipitation in the site area averages about 45-inches per year and is fairly uniformly distributed throughout all months. Hurricanes occasionally threaten Long Island during the summer and early fall. Several severe hurricanes have struck Long Island, causing damaging winds, heavy rains, and tidal flooding.

3.4 Demography and Socioeconomics

The area within 10 miles of the site includes parts of the three Suffolk County towns; Brookhaven, Riverhead, and Southampton. The current population within the 10-mile radius of the site is 148,040. Population data are based on a 1990 LILCO survey, and referenced in the Supplement To Environmental Report (Decommissioning) [Ref. 4].

Decommissioning is anticipated to take 27 months, and employ at its peak approximately 650 people. The temporary nature of the decommissioning effort and the limited size of the work force support the conclusion that no significant demographic shifts will occur as a result of decommissioning [Ref. 4].

3.5 Land

With the exception of radioactive waste disposal, the vast majority of all decommissioning activities will take place on 18 acres of the Shoreham Project Area. There will be no disturbance of undisturbed land on or off the site. Existing recreational, conservation and residential areas adjacent to the Shoreham site will be unaffected. There will be no impact on public access to unused portions of the site. As a result of the Shoreham Decommissioning effort, there will be no changes in the rate of residential or industrial development.

The Shoreham site is within the established boundary of the New York State Coastal Zone Management Program. The licensee will submit the appropriate documentation related to DECON decommissioning to the proper State agencies.

When the spent fuel is removed from the plant, it may be necessary to dredge the intake canal. Any dredging will be carried out in accordance with appropriate Federal and State regulations.

No impact to archaeological or historic sites will result from the decommissioning of the Shoreham plant. According to the the historic resources listed in the National Register of Historic Places [Ref. 13] for the Town of Brookhaven, neither the project site nor any site within a one-half mile radius of the project site contains any currently listed historic site [Ref. 4].

3.6 Surface Water

The waters of Long Island Sound and the Wading River Marsh will not be affected by the Shoreham decommissioning activities. Removal of the spent fuel from the Reactor Building may have a minor impact on water resources if barge transportation is chosen to ship the fuel. The impact would be due to dredging the intake canal to accommodate an ocean-going barge. The impact would be minimal, and the licensee would get the appropriate authorizations from Federal and State regulatory agencies [Ref. 4].

3.7 Ground Water

A ground-water monitoring program has been in place at Shoreham since 1986; to assess the effects of Shoreham's operation on ground-water quality (Shoreham's Facility Operating License No. NPF-82, Appendix B, - "Environmental Protection Plan (EPP))" [Ref. 14]. The plant has three station supply wells used for plant make-up and domestic use. LIPA will adapt the EPP for use during decommissioning [Ref. 4].

3.8 Biota

The Wading River Marsh is a State listed significant wildlife habitat. Decommissioning will occur on developed areas of the site, and no indigenous vegetation will be removed from the site. Decommissioning activities are not expected to have an impact on any listed endangered, threatened, and special-concern species, none of which was recorded at the site or surrounding environs, during the original construction of the facility LILCO, Application Environmental Report, Docket No. 50-322, Revision 4, October 1979 [Ref. 15].

No marine wildlife will suffer significant impacts as a result of any necessary grading and filling in the vicinity of the cooling water intake canal. Any temporary filling will be in accordance with applicable Federal regulatory agencies [Ref. 4].

4.0 PROPOSED DECOMMISSIONING ACTION

The licensee selected the DECON decommissioning alternative. The DECON decommissioning scope of work at the Shoreham Nuclear Power Station includes the removal of contaminated and activated systems, system components, and structures. The licensee proposes to dismantle and remove system piping and system components, and to decontaminate plant structures [Ref. 6].

4.1 Major Decommissioning Activities, Tasks, and Schedules

Specific work efforts necessary to safely decommission the Shoreham plant are identified and discussed in this section. The principal activities and tasks necessary to meet the decommissioning objectives of dismantlement of systems and portions of systems, and decontamination of structures, to meet the existing criteria for release for unrestricted use, as defined in Regulatory Guide 1.86 [Ref. 12], and described in the licensee's DP [Ref. 3] are as follow:

Decommissioning Major Activities and Tasks

- System Dismantlement and Structural Decontamination
- Segmentation of the RPV and Internals
- Radwaste Management
- Area Cleanup and Decontamination
- Final Radiation Survey

In the sections that follow, each of the major activities and tasks necessary to decommission Shoreham is discussed.

4.1.1 System Dismantlement Structural Decontamination

A site characterization study was performed by the licensee, and is documented in the "Shoreham Site Characterization Program, Final Report (SSCR)" [Ref. 10]. Based on the findings documented in the SSCR, the following systems and structures were identified as being contaminated and/or activated.

Systems

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

Structures

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

In its detailed engineering planning for the decommissioning, the licensee has committed to follow the guidelines set forth in the applicable regulatory guides, and to strive to maintain as low as is reasonably achievable (ALARA) occupational radiation exposure during all decommissioning activities.

The licensee noted in response to staff questions [Ref. 6], that its intentions are to dismantle and remove all contaminated systems, and to mechanically decontaminate plant structures, using only demineralized water.

If local decontamination of some areas requires the use of more aggressive measures, then aggressive chemical reagents such as gels and sols will be used. The licensee has not identified the specific chemicals that may be used; however, several chemicals such as nitric acid and phosphoric acid are being considered. The site Chemical Control Program will be used to address the potential creation of mixed waste.

4.1.2 Systems Removal

During systems removal the licensee will ensure that health physics preparations are carried out to control contamination in areas where work will be conducted. System low points will be identified, to ensure the collection of potentially contaminated liquids. For system piping 3-inches and less in diameter, pipe sections will be cut to lengths that will ensure that the sections fit into waste containers. The licensee will use hand held power band saws to remove the 3-inch and smaller piping. Once removed from the system, piping will be transported to either the Reactor Building or Radwaste Building truck bays, where it will be loaded onto transport vehicles, and transported to either an offsite volume-reduction facility (the licensee retains title to waste) or an offsite disposal facility (offsite disposal depends on the availability of such a facility to Shoreham) [Ref. 6].

System piping greater than 3-inches in diameter will be cut with an outside diameter (O.D) mounted milling machine. Once removed from the system, this piping will be handled in a fashion similar (large bore pipe will not be put into waste containers) to the 3 inch or less piping. System components, pumps, valves, instrumentation, motors, and heat exchangers will be mechanically removed from systems. and packaged and loaded onto transport vehicles, for shipment to volume reduction or disposal. The same removal procedures described previously for small bore piping (less than 3-inches diameter) will be used [Ref. 6].

Airborne and liquid radioactive waste will be generated during the dismantlement, and when using high pressure and ultrahigh pressure water. Airborne contamination will be controlled using tents, steel enclosures, glove boxes, and glove bags with high efficiencey particulate air (HEPA) filter exhaust will be used. Liquid waste will be collected in the existing sumps, and routed to the Radwaste Building with the existing floor drain system, and processed with the existing liquid radwaste system [Ref. 6].

The following sections describe specific system dismantlement techniques, as noted by the licensee in response to staff questions [Ref. 6]. During the performance of the specific system dismantlements, the licensee will verify radiological conditions. Contamination control will be exercised at each localized system breach. Controls such as draping the area with herculite and taking airborne grab samples will be used, and if, required, continuous airborne monitoring will be carried out. Piping and components left in place after dismantlement and decontamination will be surveyed in accordance with the requirement of the Final Termination Survey requirements.

4.1.2.1 Control Rod Drive (CRD)

The CRD system will be completely dismantled. Piping will be cut using band saws (piping less than or equal to 3-inches in diameter), and with O.D.

milling machines (piping greater than 3-inches in diameter). Pumps, motors, hydraulic control units (HCUs), and motor operators will be mechanically removed.

4.1.2.2 Process Sampling

This Process Sampling system consists entirely of small-diameter tubing (piping less than 3-inches in diameter) and sampling panel. The tubing will be cut using band saws, and the panel will be mechanically removed.

4.1.2.3 Core Spray

The Core Spray system will be completely dismantled. The piping will be cut using band saws for piping less than 3-inches in diameter, and with O.D. milling machines for piping greater than 3-inches in diameter. Pumps, motors, and valve operators will be mechanically removed.

4.1.2.4 Reactor Water Cleanup (RWCU)

The RWCU system was chemically decontaminated (Soft Decon). Characterizations after the decontamination effort were inconclusive. The system will be completely dismantled. Piping less than 3-inches in diameter will be cut using band saws, and piping greater than 3-inches in diameter will be removed using O.D. milling machines. Pumps, motors, and valve operators will be mechanically removed. The Phase Separator Tank will be mechanically decontaminated, using ultrahigh pressure water.

4.1.2.5 Residual Heat Removal (RHR) System

The RHR system will be completely dismantled. Piping less than 3-inches in diameter will be cut, using band saws, and piping greater than 3-inches in diameter will be removed using O.D. milling machines. Pumps, motors, and valve operators will be mechanically removed. The Phase Separator Tank will be mechanically decontaminated, using ultrahigh pressure water.

4.1.2.6 Transversing In-Core Probe (TIP)

The TIP will be completely dismantled. This system consists entirely of small diameter tubing less than 3-inches in diameter. The tubing will be cut using band saws. Components such as the drive units, detector shield chambers, index mechanisms, tube shear, and ball valve will be mechanically removed.

4.1.2.7 Liquid Radwaste System

The licensee will use the liquid radwaste system to process liquids generated during the decommissioning (see Section 4.3.1.2) [Ref. 6]. When feasible (after processing of the major volumes of water generated during decommissioning), piping that is contaminated and accessible will be cut, using band saws (piping less than 3-inches in diameter), or O.D. milling machines, for piping greater than 3-inches in diameter. Contaminated piping that is inaccessible, such as embedded drains will be mechanically decontaminated, using ultrahigh pressure water (>20,000 psi). The licensee intends to perform chemical decontamination only if mechanical decontamination fail to meet the decontamination objectives. Pumps, motors, valve operators, evaporators, filters, and demineralizers will be mechanically removed. Contaminated tanks, such as the Floor Drain Collection Tank, will be mechanically decontaminated, using high pressure water (<20,000 psi).

4.1.2.8 Fuel Pool Cooling Cleanup

The Spent Fuel Storage Pool is currently being used to store the Shoreham irradiated fuel, and it is not anticipated that this fuel will be removed before the end of calendar year 1992. Dismantlements and decontamination of the Spent Fuel Storage Pool and its support systems will not occur until the irradiated fuel is removed. When dismantlement and decontamination of the Spent Pool Cooling and Cleanup System are possible, the licensee intends to use band saws to cut the accessible piping less than 3-inches in diameter. Contaminated piping that is inaccessible will be mechanically decontaminated, using ultrahigh pressure water (>20,000 psi). The licensee intends to perform chemical decontamination only if mechanical decontamination fails to meet the decontamination objectives. Pumps, motors, valve operators, and filters will be mechanically removed.

4.1.2.9 Reactor Water Recirculation

The Reactor Water Recirculation system will be completely dismantled. Piping less than 3-inches in diameter will be cut, using band saws, and piping greater than 3-inches in diameter will be removed, using O.D. milling machines. Pumps, motors, and valve operators will be mechanically removed.

4.1.2.10 Embedded Piping

Embedded and inaccessible piping will be mechanically decontaminated, using ultrahigh pressure water (>20,000 psi) mole nozzles. The licensee will only perform chemical decontamination if the mechanical decontamination is unable to reduce contamination levels below the release criteria.

In accordance with the guidance in Regulatory Guide 1.86 [Ref. 12] the NRC will consider as contaminated, all surfaces that are inaccessible for purposes of measurement of radiation levels. The licensee will be required to devise suitable survey methods for embedded piping, or such piping will be required to be removed before the area will be released for unrestricted use.

4.1.3 Structures

The licensee intends to mechanically decontaminate the structural areas identified previously in Section 4.1.1, using demineralized water. The licensee intends to perform chemical decontamination only if mechanical decontamination fails to meet the decontamination objectives. The site Chemical Control Program will be used to address the potential for creating mixed waste.

The RPV inside surface will be mechanically decontaminated, using ultrahigh pressure water (>20,000 psi). This decontamination will be undertaken after the removal of the RPV internals. The licensee intends to use only demineralized water; however, if the need arises, the licensee intends to use localized applications of decontamination chemicals.

SHOREHAM NUCLEAR POWER STATION
REACTOR BUILDING – ELEVATION 175'–9"

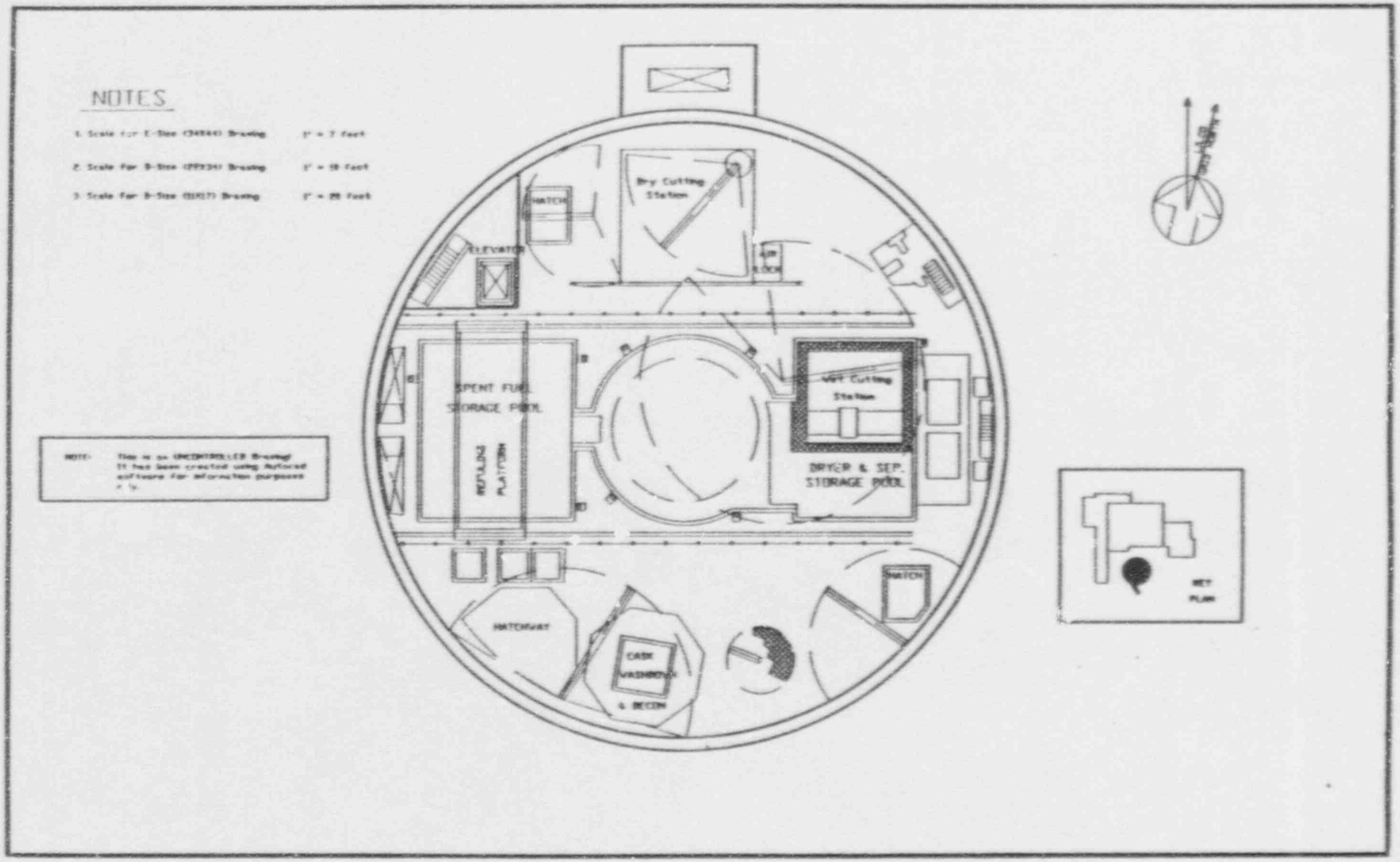


Figure 6

4.2 RPV and Internals

The RPV segmentation dismantlement will consist of cutting up the RPV and its internals into manageable pieces, decontamination of the non-activated components, rigging the components, packaging the pieces in appropriate containers, and shipping the materials offsite. RPV segmentation is expected to take place in three areas of the Reactor Building (i.e., RPV (in-situ); the Dry Cutting Station (DCS); and the Wet Cutting Station (WCS), which will be located in the Dryer Moisture Separator Storage Pool). The locations of the DCS and WCS are shown in Figure 6.

Before the start of RPV segmentation, the RPV will be decontaminated, using ultrahigh pressure water. The licensee has made the assumption that the lower portion of the RPV (recirculation nozzle and below) will not need to be removed. If additional RPV segmentation (lower portion of the RPV) is required, based on additional characterization, the cost, radwaste volume, and personnel radiation exposure estimates would all remain within the bounds defined in the Shoreham DP [Ref. 3].

Segmentation of the RPV and its internals will be accomplished using the various cutting techniques defined next, and described in Ref. 11:

- The RPV will be segmented, using diamond wire to cut heavy sections of steel, and power band saws will be used to cut small diameter RPV nozzle piping (1-inch instrument nozzles, three-quarter-inch high pressure leak detection nozzles, and one CRD return-line nozzle).

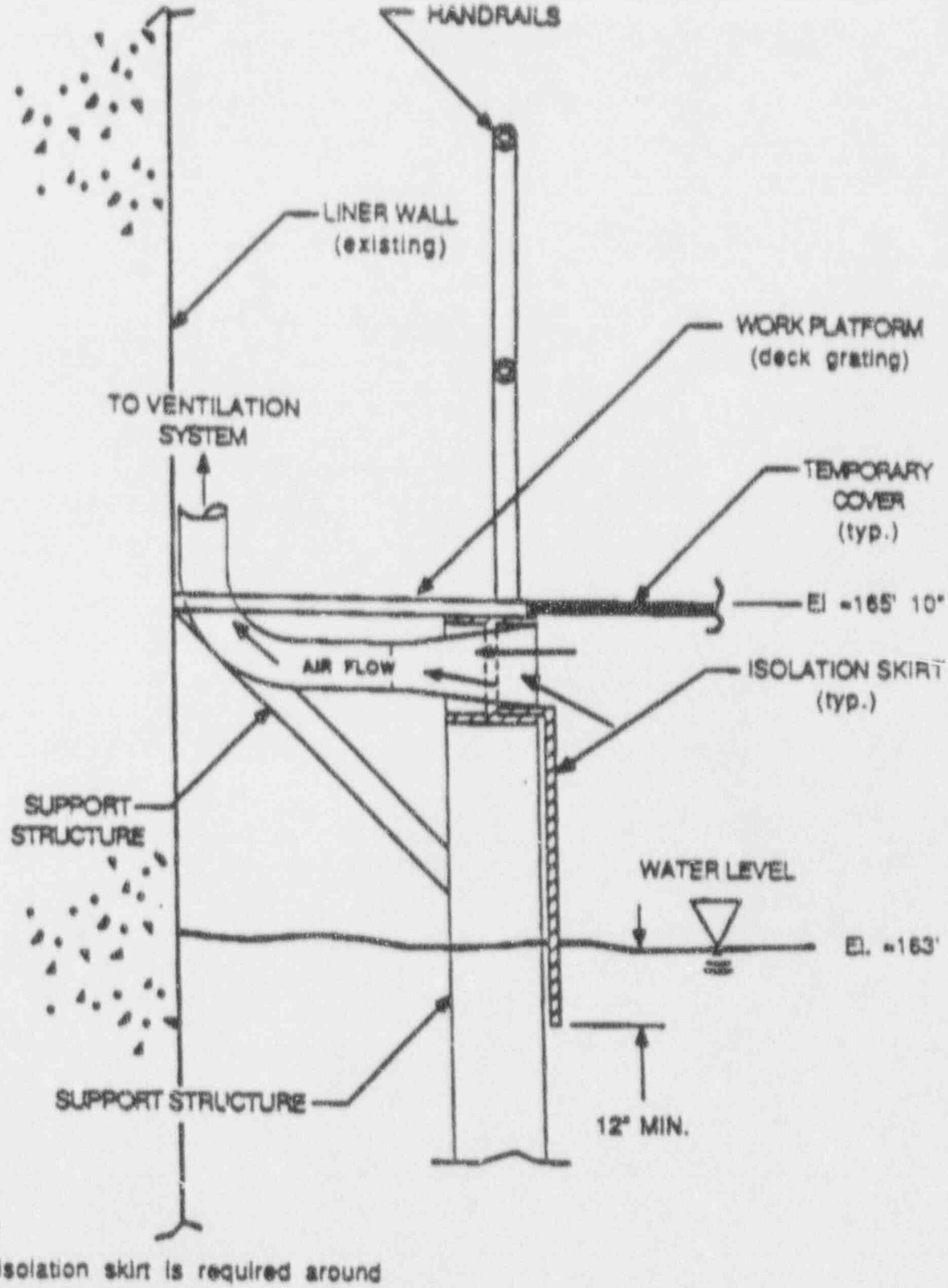
- The internals will be segmented, using plasma arc torches and metal disintegration machining.

The majority of the lifts related to segmentation will be performed using standard industry methods such as rigging slings, welded pad eyes, and plate clamps. The primary load paths will be between the RPV, DCS, WCS, and Reactor Building equipment hatch (see Figure 6). The lifts will be accomplished using the polar crane, auxiliary crane, and jib cranes. While fuel remains in the Spent Fuel Storage Pool, heavy load-handling limits and safe load paths identified in NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" [Ref. 16] will be maintained. In addition, the plant's Defueled Technical Specifications [Ref. 17] implemented by plant procedures, will be used, to preclude any load- handling over the Spent Fuel Storage Pool.

The steam dryer and the moisture separator will be lifted using special rigging, and will be transferred to the DCS. Ring sections of the RPV will be sized to allow for safe and efficient rigging, within the load limits of the polar crane. In addition, piping will be cut as close to the vessel wall as possible, to provide clearance of the reactor cavity.

Contamination-control boundaries will be used during RPV and internal segmentation, and will consist of plastic sheeting tents and HEPA filters, to control loose or airborne radioactive particulates. Cutting operations for the more highly activated reactor internals will be carried out in the WCS, using remotely operated underwater tools.

SHOREHAM NUCLEAR POWER STATION
FUME COLLECTION SYSTEM



NOTES:

1. The isolation skirt is required around the entire work area opening periphery.
2. Only one ventilation port is shown, additional ports may be required.

Figure 7

SHOREHAM NUCLEAR POWER STATION
DRY CUTTING STATION
(Flat View)

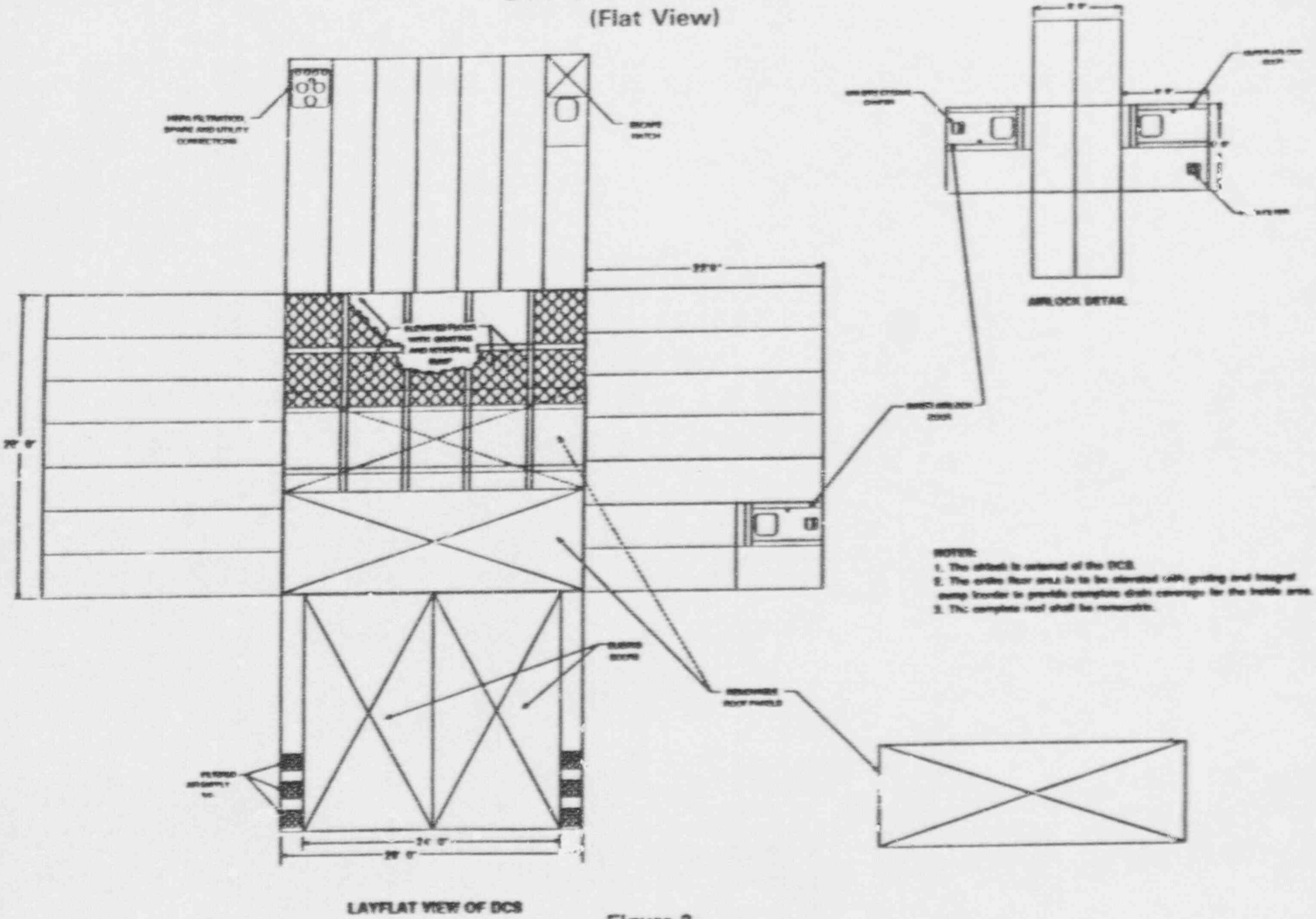


Figure 8

The WCS will be used to provide water shielding during the plasma-arc torch segmenting of the more highly activated reactor components. An enclosure above the WCS will be used for contamination control and fume collection. The fume- collection system consists of fume isolation skirts, along the platform grating as shown in Figure 7; three movable covers made of 12-gauge carbon steel sheets painted with epoxy paint (the opening in the cover is 18-ft 4-in. by 17-ft 4-in.); duct work from the isolation skirts to a HEPA filtration system; and HEPA filtering equipment with demister, bag-in/bag-out prefilter, HEPA filter, fan, and motor. The WCS will include a work platform that will be positioned above the water. The movable cover will be able to be po itioned anywhere over the 306-sq-ft WCS.

An underwater video system will be used to provide remote monitoring of WCS cutting activities. Water filtration equipment (see Section 4.3.1.2) will be used to maintain water clarity, during cutting operations.

The DCS will be installed on the refueling floor of the Reactor Building at the 175-ft level. The approximate dimensions of the DCS are 28-ft X 28-ft X 22-ft high. The DCS [illegible] be a prefabricated building that includes ventilation, lighting, [illegible]rsonnel air lock, a raised floor, and an internal sump, to ensure positive drainage of the fluids generated during decommissioning and cutting operations. The DCS enclosure will be equipped with HEPA filtration ventilation that will be used to maintain an internal negative air pressure, and that will discharge directly to the refueling floor atmosphere. Figure 8 provides a flat view of the DCS. In addition, two portable 1500 cubic feet per minute (CFM) fume collection systems with fans and pre-filters, for local fume collection, will be used inside the DCS, to minimize airborne contamination [Ref. 11].

4.2.1 RPV and Internals Segmentation

Following is a description of the RPV and internal segmentation process, as described in Ref. 11:

- The reactor vessel will be filled with water to approximately 6-feet above the top guide.
- The steam dryer assembly will be removed from the RPV and transferred to the DCS.
- In the DCS, the seal skirt ring and remaining dryer portion will be cut into pieces, using manually operated plasma arc torches.
- The moisture separator assembly will be removed from the RPV and transferred to the DCS, where it will be segmented into sections.
- In the DCS, the shroud head plates will be segmented into pieces, using track mounted plasma arc torches. The vertical plates between the stand pipes at locations just above the shroud head will be cut, using remote plasma arc torches. The plate on top of the separators will be cut into pieces, using remote plasma arc torches.
- The RPV work platform will then be lowered into the RPV, to the feedwater sparger elevation. The RPV inner walls, feedwater nozzles, and core

spray nozzle thermal liners will be decontaminated with demineralized water.

- Contamination control covers will be installed in the main steam lines.

- The feedwater sparger end pins will be disconnected and removed. The spargers will be pulled out of feedwater nozzles. Further segmentation, if necessary, will be done manually in the DCS.

- Following removal of the feedwater spargers, contamination control covers will then be installed in the feedwater nozzles.

- The work platform and water level will be lowered, to cut the core spray supply headers at the thermal liner, and core spray interface in the RPV, with manual plasma arc torches. The core spray supply header at the elbows will be cut just before the vertical runs and also just above the water level. Any further segmentation of the removed piping sections will be performed manually in the DCS.

- After removal of the internal core spray piping, contamination control covers are installed in the core spray nozzles.

- The core spray elbows will be cut at the shroud outside diameter with underwater metal disintegration machining.

- The separator and dryer guide rod attachment brackets will be cut and the guide rods removed. The guide rod brackets at the shroud head flange will be cut with underwater metal disintegration machine. The guide rod brackets at the RPV head flange will be cut with plasma arc torches.

- The LPRM and SRM/IRM guide tubes will be severed.

- The top guide and upper shroud will be cut 360 degrees below the top guide in the RPV, with underwater plasma arc torches mounted on a suspended track. Wedges will be installed, during the cutting process, to prevent the kerf from closing during cutting.

- This first ring section will then be transferred dry to the WCS, for further segmentation. This section of the shroud will also contain the core spray spargers.

- In the WCS, the top guide, core spray spargers, and upper shroud will be segmented with an underwater plasma torch.

- Contamination control covers will be placed over the control rod drive housing and in core guide tubes.

- The remainder of the core shroud will be cut from the inside diameter of the reactor vessel, with an underwater plasma arc torch mounted on a track. The shroud will be removed in segments. The bottom ring will be cut just above the core support plate.

- Each shroud ring section will then be placed in the WCS, where it will be segmented.

- In parallel with cutting a new ring segment from the shroud, the ring segment in the WCS will be cut into pieces with underwater plasma arc torches.

- The jet pump hold down bolts will be unlocked by overtorquing the keepers with long-handled tools, or the keeper will be cut with a plasma arc torch.

- The ram's head will be removed and placed in the DCS for further segmentation. If radiological conditions dictate, the ram's head will be placed in the WCS for segmentation.

- Debris and sediment will be removed from the lower vessel head, and the reactor vessel will be drained.

- The jet pump instrument lines will be cut with a hydraulic shear.

- The diffusers will be cut from the shroud support ledge by cutting the diffuser inside diameter bottom section with plasma arc torches, and will then be placed in the DCS.

- In the DCS, the diffuser will be cut into pieces with a plasma arc [illegible]

- The jet pump riser brace will be removed with plasma arc torches.

- The riser will be cut at the elbows, by the inlet nozzle thermal sleeves, with plasma arc torches, and placed in the DCS.

- The riser will be further segmented with plasma arc torches, in the DCS.

- The core support plate hold down studs will be cut with plasma arc torches.

- The in core guide tubes and stabilizers will be cut in air, with plasma arc torches, in the reactor.

- The lower portions of the in core guide tubes (below the core plate) and stabilizer assembly will be removed in one piece and transferred to the DCS, for further segmentation with plasma arc torches.

- The remaining shroud section will be cut with an underwater plasma arc torch, placed into the DCS, and segmented into three pieces.

- The RPV inner walls, recirculation inlet nozzles thermal sleeves, and shroud support will be decontaminated with ultrahigh pressure demineralized water.

- The large debris generated from the cutting operation will then be removed from the bottom head of the reactor vessel. Small debris will be vacuumed.

4.2.2 RPV Shell and Nozzle Segmentation

Before shell and nozzle segmentation, the inner bellows seal and the RPV stabilizer will be removed. The RPV and nozzles will be severed, with mechanical cutting techniques, into manageable ring sections, and transferred to the DCS for further segmentation. The cutting technique that will be used will be Diamond Wire Saw (DWS) cutting. Power band saws will be used to cut small diameter RPV nozzle piping. Nozzle lines greater than 4-inches in diameter will be cut using DWS. A step-by-step description for RPV shell and nozzle segmentation follows [Ref. 9]:

- The RPV thermocouples will be removed.

- The following large bore nozzles will be cut, using a diamond wire saw technique:

  - Four 24-inch Main Steam Outlet Nozzles
  - Four 12-inch Feedwater Nozzles
  - Two 10-inch Core-Spray Nozzles

The first cut will be made on the nozzle close to the reactor vessel, and a lifting fixture will then be attached to the piping side of the cut and secured to a lifting device. A second wire cut will be made outboard of the lifting fixture on the pipe. The nozzle and pipe assembly cut will be removed from the reactor annulus and placed in the DCS, for further segmentation, if required.

- Small bore nozzles will be cut using hand held power band saws.

- Nozzle openings will be plugged with contamination control barriers after cutting.

- The bellows seal will be cut and removed from the Reactor Building.

- Stabilizers (external struts used to aid stability of the RPV located near top of vessel) draw bars will be cut manually, with oxy-fuel torch.

- The diamond wire saw idler wheels will then be set up in the reactor vessel annulus and positioned horizontally approximately 5-feet below the top flange of the reactor vessel. The RPV shell will be segmented into six ring sections, using the diamond wire saw.

- The first ring section will be removed from the reactor cavity and placed in the DCS, for further segmentation.

- In the DCS, the ring section will be decontaminated. Decontamination of a ring in the DCS will be done in parallel with the cutting of a new ring section from the RPV.

- The ring section will be further segmented into pieces in the DCS, using a diamond wire saw.

- The steps above will be repeated until all the ring sections are cut, removed, and segmented. The last cut will be made just above the recirculation discharge nozzle.

- The debris from the lower head will be vacuumed. The vessel will be drained and decontaminated, using ultra high pressure demineralized water.

- Radiological characterization to ensure that all contamination above the release criteria has been removed will be performed.

4.3 Waste Management and Disposal

4.3.1 Waste Generation

The licensee anticipates that the decommissioning of Shoreham will generate 79,300 cu ft of radioactive waste [Ref. 3]. This volume does not include any spent fuel; fuel support castings and peripheral pieces (fuel disposition is not part of the decommissioning effort); or radioactive waste generated during plant operation. The licensee intends to dispose of activated and contaminated materials, such as traveling in core probes and other in core instrumentation; control rod blades and drives; and radioactive fluids, resins, filter media, and sludge currently contained within systems, before the start of decommissioning [Ref. 3]. During decommissioning, an estimated 620,000 gallons of water will be generated and will require processing.

The licensee intends to ship as much as possible of the radioactive materials to offsite disposal sites, before the probable closure of the available sites, on December 31, 1992.

4.3.1.1 Gaseous Radioactive Waste

Reactor operations were terminated at Shoreham in 1987, and the short lived fission products have decayed to insignificant levels. Therefore, the gaseous waste treatment system is not needed for processing fission gases. The plant's existing ventilation system will be used to prevent the release of airborne radioactive particulates generated during the decommissioning. The plant's ventilation system will be augmented by portable equipment with HEPA filtration.

The licensee's estimated releases of airborne particulates at Shoreham during decommissioning are based on the approach used in estimating releases associated with decommissioning of a Reference BWR, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station," NUREG/CR-0672 [Ref. 5]. Estimated airborne particulate releases at Shoreham are based on releases associated with the RPV segmentation, and airborne particulate releases associated with all other decommissioning activities.

During the decommissioning, potential airborne particulates will be generated during cutting activities in the DCS, WCS, and RPV. Air from the DCS, WCS, and RPV will be vented through HEPA filters, and exhausted back into the Reactor Building. The Reactor Building will be monitored by using portable air samplers, and, if necessary, by Continuous Air Monitors (CAMs). The existing plant ventilation system exhausts through HEPA filters, from the Reactor Building to the environment, and provides outside air back into the building at a rate of 2.5 air changes per hour. Noncutting activities may (note Section 4.1.2) require control of airborne particulates; in such an event, portable HEPA filtration units will be used. In addition, portable vacuum cleaners used within the radiologically controlled areas have built-in

HEPA filters for controlling airborne radioactivity [Ref. 6].

A fume collection system will be provided for work inside the RPV. This ventilation system will be appropriately sized, and will provide HEPA filtration before exhausting to the Reactor Building [Ref. 6].

The estimated releases from the DCS will comprise approximately three-quarters of the total releases for the entire Shoreham Decommissioning [Ref. 11]. The estimated quantities and types of radioactivity released to the Reactor Building as a result of cutting activities in the DCS are provided in Table 7.

Table 7

Estimated Curies Released via the DCS

H-3	1.24E-4
C-14	2.48E-5
Fe-55	3.11E-4
Co-60	1.28E-4
Ni-59	6.50E-8
Ni-63	9.61E-6

The releases just listed are based on the activated RPV components being cut in the DCS, and the fraction of activated materials from cutting components is assumed to become airborne, and is equal to the ratio of the cut metal volume to the total activated metal volume. The activated materials released into the DCS envelope will be filtered through HEPA filters with a 99.95 percent efficiency. As previously noted, releases from the DCS are discharged directly into the Reactor Building. The exhaust from the HEPA filters will be monitored using portable alarming area radiation detectors. The DCS's Heating, Ventilation, and Air Conditioning (HVAC) system will have an exchange rate of about 12 room volumes per hour. All the HEPA filter housings will have bag-in and bag-out capabilities that will be capable of being performed quickly. No leakage is assumed from the DCS envelope because of the negative pressure that will be maintained in the DCS. During the operation of the DCS, LIPA will provide for radiological surveys, in accordance with 10 CFR 20.201.

4.3.1.2 Liquid Radioactive Waste

Liquid radioactive waste will be generated as a result of decommissioning activities at Shoreham. The licensee estimates that approximately 620,000 gallons [Ref. 11] of contaminated water will require processing during the decommissioning. The volumes of water that make up the approximately 620,000 gallons of contaminated water consist of; water that will be used to flood the RPV (72,000 gallons, estimated maximum concentrations of 4.44E-5 uCi/ml); water used in the WCS (87,000 gallons, with an estimated maximum concentration of 4.44E-5 uCi/ml); a total of 360,000 gallons (estimated concentration of 2.3E-6 uCi/ml) of water contained in the Spent Fuel Storage Pool; and an estimated additional 100,000 gallons (estimated maximum concentration of 1.58E-5 uCi/ml) of water used for makeup, flushing, decontamination, and hydrolancing [Refs. 6 and 11].

The licensee intends to use the plant's installed liquid radwaste system throughout the decommissioning, to recirculate, sample, and process liquid

SHOREHAM NUCLEAR POWER STATION
LIQUID RADWASTE SYSTEM

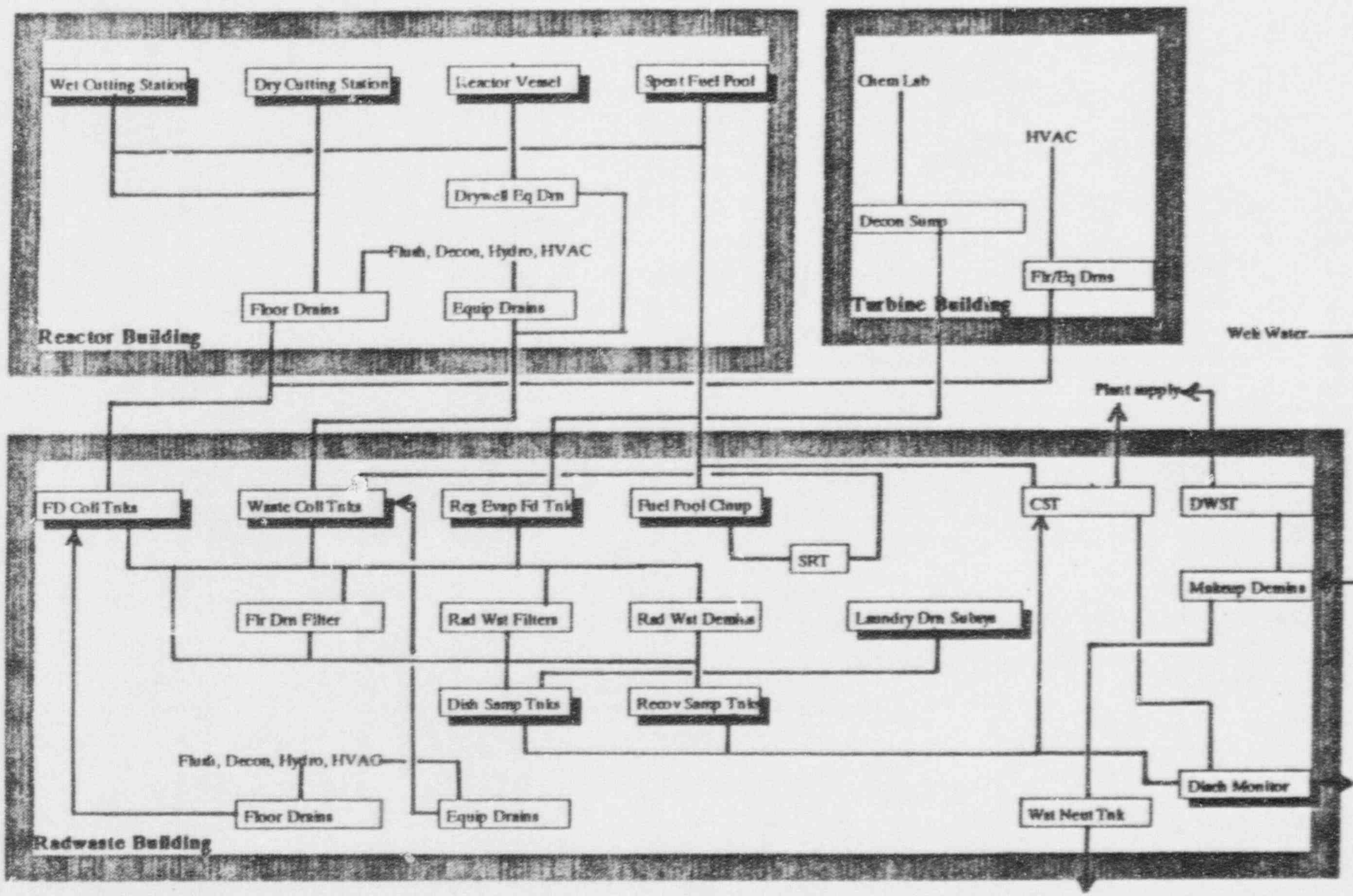


Figure 9

SHOREHAM NUCLEAR POWER STATION
WET CUTTING STATION FILTRATION SYSTEM

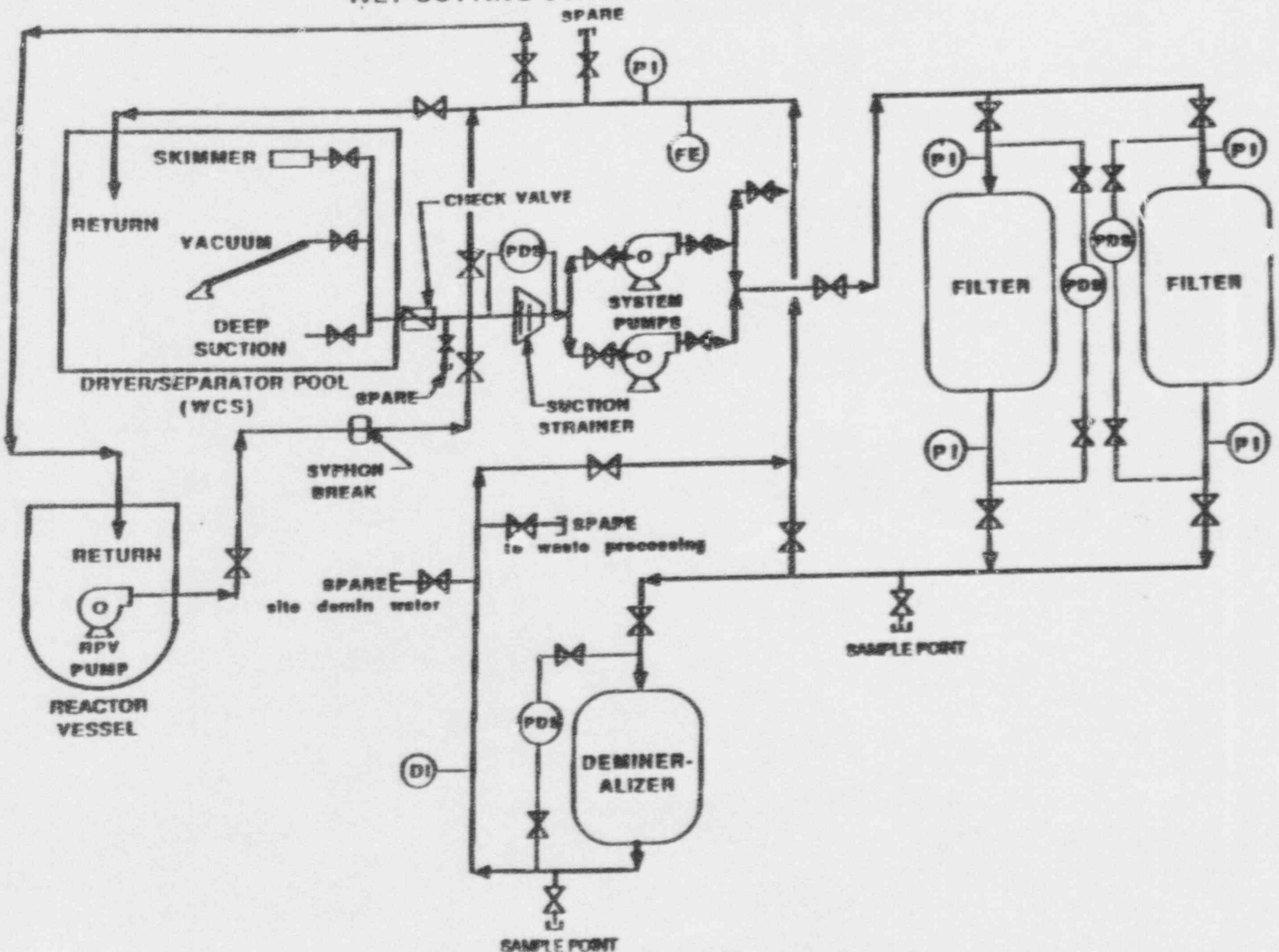


Figure 10

radioactive waste water. The liquid radwaste system has been segregated into two parts -- that is, that equipment that is not required to process the water expected to be generated during decommissioning, and equipment which will be used. Equipment items not expected to be used include: the Waste Evaporator, the Regenerant Evaporator, the Phase Separator Tanks, the Cement Storage Silo, the Cement Batch Bin, the Waste Dewatering Tank, the Radwaste Filter Body Feed Tank, and Laundry Drain Subsystem. The following equipment items are expected to be used: including the Spent Resin Tanks, the Regenerant Liquid and Evaporator Feed Tanks, the Floor Drain Filter, the Radwaste Demineralizers, the Recovery Sample Tanks, the Radwaste Filters, and the Waste Collector Tanks. Figure 9 shows the liquid radwaste system and its inputs [Ref. 6].

The equipment identified to be used during the decommissioning will process contaminated water generated during decommissioning. The plant's drains will be used to collect contaminated water generated from flushing and hydrolancing. In addition, the plant's drain system will be used as the disposal path for contaminated water from the WCS, the RPV, and the DCS [Ref. 6] .

Water in the WCS and RPV will be processed and clarified by an underwater skid mounted filter and demineralizer system, as shown in Figure 10. The system is designed so that filter elements and resin change outs can be accomplished underwater. Personnel exposure will be minimized with the implementation of the following controls: an underwater gamma-radiation monitor with remote readout, placed next to the filter housing, so that filters can be changed out at a predetermined dose; filter change outs performed underwater; and remote handling of filters, using long-handled tools for transfer ring them to shielded transfer containers. When filters are removed, they will be placed into High Integrity Containers (HICs), and dewatered [Ref. 6].

Segmentation and cutting in the RPV and WCS will generate radioactive "hot particles" on wetted surfaces. Hot particles will not be a problem until drainage of the RPV and WCS begin. The licensee plans to use the following techniques to mitigate hot particles; use high pressure water to wash the walls of the WCS and RPV as they are being drained, additional controls added to RWP, and increased radiation monitoring and surveillance of work areas.

The licensee proposes to decommission up to half the components in the liquid radwaste system, identified above, upon completion of the mechanical and chemical decontamination of the Reactor Building (not including the refueling floor). The remaining equipment will be dismantled upon draining of the WCS, RPV, and Spent Fuel Storage Pool. At the point of complete decommissioning of the liquid radwaste system, the licensee proposed to have in place the Laundry Drain Subsystem. This system will be used as an industrial waste collection and processing system.

4.3.1.3 Solid Radioactive Waste

The majority of the solid waste generated during the decommissioning will be generated because of the dismantlement of activated and contaminated systems and structures. The licensee will dismantle and segment all contaminated systems, the RPV, and its internals. Because of the imminent closure of the two available offsite disposal sites, a significant amount of this waste may require interim storage onsite at Shoreham.

The NRC staff has required the licensee to formulate contingency plans to provide interim storage onsite for all the waste generated during the decommissioning. Table 8 provides a summary of the estimated radioactive waste anticipated to be generated during decommissioning. Total burial volumes (in cubic feet), radioactivity, average concentrations, and waste classification data are provided in Table 8.

Table 8

Estimated Radioactive Waste Data for Shoreham Decommissioning[a]

Component/ System	Burial Volume(ft^3)	Total Activity(Ci)[b]	Average[c] Gross Concentration	10 CFR 61 Waste Class
RPV and Internals	16,500	601.17	1.28	A
Reactor Recirc.	6,000	2.45E-4	1.44E-6	A
Control Rod Drive (System)	500[b]	3.00E-4	2.12E-5	A
Residual Heat Removal	15,100	4.30E-4	1.01E-4	A
Core Spray	1,600	7.19E-4	1.59E-5	A
Reactor Water Cleanup	9,200	6.16E-4	2.36E-6	A
Fuel Pool Cleanup	2,500	7.86E-4	1.11E-5	A
Condensate Demineralizer	2,000	2.62E-5	4.69E-7	A
Process Sampling Sys.	300	2.29E-5	2.69E-6	A
Spent Fuel Rack and Appurtenances	8,300	5.65E-4	2.40E-6	A
Process Waste & DAW (d)	7,700	negligible	unknown, assumed negligible	A
Demineralizer Resins/Filters	3,200	negligible	unknown, assumed negligible	A
Liquid Radwaste	6,000	1.60E-4	9.14E-7	A
Mirror Insulation	400	negligible	negligible	A
TOTALS	79,300	601.17		

(a) As of March - April 1990, except for the RPV and Internals which are as of July 1990.

(b) Does not include control blades, control rod drives, LPRM (hot ends), LPRM (cold end), antimony pins, and beryllium sleeves.

(c) Units of uCi/cc.

(d) DAW is dry active waste.

The licensee, after an evaluation, concluded that the entire volume of waste (79,300 cu ft) generated during the decommissioning can be stored in the Radwaste Building. The Radwaste Building was designed with features to handle and store radioactive waste containers with high levels of radiation.

The existing overhead cranes, remote crane controls with shielded observation windows, shielded storage vaults in the floor, and a shielded storage rooms, are features that will aid in maintaining work exposure ALARA during long term interim onsite storage [Ref. 6].

The waste volume analysis for possible interim onsite storage of waste is based on the data provided in Table 8. For purposes of waste storage in the Radwaste Building the waste was grouped by container type. The container type grouping is provided in Table 9.

Table 9

Waste Container Type Grouping

Container types	Number of containers
55 Gallon Drums	6
CNS-8-120 Liners	13
CNS-14-195 Liners	3
HICs	22
B-25 Waste Containers	726
Coating & Pallets a)	

a) RPV shell segments (approximately 2060 cu ft) wrapped and palletized.

Most of the waste containers identified in Table 9 would be distributed in available areas on the 15-ft 0-in, and 50-ft 6-in elevations of the Radwaste Building [Ref. 18].

Design values for the maximum floor loading were used to limit the number of containers that could be stored in each identified location. To accommodate the waste in areas that currently contain storage tanks would require removal of tanks.

Ten CNS-8-120 liners would be stored in the existing storage vaults on Elevation 19-ft 6-in, and the three others, -- 3 CNS-14-195 liners -- and the 22 HICs would be arranged on the south side of an area on the 19-ft 6-in elevation.

Ventilation will be provided to the Radwaste Building, to make the environment safe for personnel and to control air flow. A fume exhaust will operate in the truck bay area. No permanent radiation monitors will be used; air sampling requirements will be satisfied by period surveillance, using portable air samplers. A fire detection system will be wired to the Radwaste Control Room on elevation 37-ft 6-in of the Radwaste Building, to detect heat and smoke in all areas of the building. Fire suppression in the radwaste storage areas will be by local portable dry chemical extinguishers. A manual hose station will be provided in the truck bay area.

The licensee does not expect generation of mixed waste during decommissioning. To minimize the potential for generation of mixed waste, the licensee will use the following programs: the Chemical Control Program, Radwaste Program, the Controlled Materials Program, the Solid Waste Process Control Program, and the Station ALARA Program. If a mixed waste is generated, the licensee will make the necessary arrangements to store such waste onsite, until a disposal site can accept it [Ref. 6].

The licensee estimates that a total volume of 3200 cu ft, with an estimated total activity of 24 curies, will be generated as a result of processing liquid waste with the plant's existing liquid radwaste system. The licensee stated that there are no plans to use chelating agents in any chemical decontaminating activities. However, if any chelating agents are used, such agents in radioactive waste will be kept in the range of 0.1 to 0.8 percent for disposal before 1993, and be stabilized with a disposal facility approved solidification agent. Should the need arise, concrete and dust will be packaged as low specific activity (LSA) waste, in an approved container, and shipped to an offsite disposal site, or placed in interim onsite storage. Waste that will be stored at Shoreham after 1993 will be stored in a dewatered form, in HICs, to permit future treatment to satisfy requirements that may evolve for that waste at offsite disposal facilities.

To minimize the volume of waste, the licensee intends to use the services of offsite volume reduction contractors. After onsite decontamination, materials that can be declared clean will be released for unrestricted use. Waste generated before 1992 will be packaged and shipped to an offsite disposal facility. Post 1992 waste will remain onsite, in interim storage, until a suitable disposal facility becomes available.

4.3.1.4 Spent Fuel

Spent fuel disposal is not considered part of the decommissioning process 53 FR 24018, 24019 (June 27, 1988). However, complete release of the Shoreham site for unrestricted use will not take place until spent fuel has been removed from the site. Currently, there are 560 fuel assemblies stored in the Shoreham Spent Fuel Pool. "Defueled Safety Analysis Report (DSAR)" [Ref. 19] estimates that there are approximately 176,000 curies of radioactivity contained in the fuel.

The licensee has proposed two method for fuel-disposal. The licensee will either ship Shoreham's irradiated fuel to another utility (Nine Mile Point, Unit 2) for storage or to use as fuel at the new site, or the licensee will ship the irradiated fuel to Europe for reprocessing.

Shipment of the fuel to Nine Mile Point, Unit 2, would require a combination of road and rail transportation. The fuel would be shipped in a IF-300 cask. The IF-300 cask is basically a rail cask, however; its design facilitates truck shipment on special overweight basis for short distances for facilities such as Shoreham that lack direct railroad access.

Shipment of the fuel to Europe would require a combination of road and sea transportation. The fuel would be shipped in a TN-12/2 cask. The cask would be loaded onto a heavy-haul truck in the Reactor Building bay, and moved to the Shoreham dock-site. The cask would then be transferred to a barge for transport to an ocean-going vessel for shipment to Europe for reprocessing.

Severe slippage (period greater than 6 years, defined in NUREG-0586 [Ref. 7] for completion of the DECON alternative) in the licensee's schedule for removing the fuel from the site will be cause for the decommissioning effort to stop, and the licensee will be required to submit a modified decommissioning plan.

4.3.1.5 Controls Used to Ensure the Segregation of Non-Radioactive Waste from Radioactive Waste

LIPA will employ industry methodologies to ensure the separation of contaminated and non-contaminated materials during decommissioning. The licensee will establish radiological controls consistent with the Health Physics Program, and the implementation of Institute of Nuclear Power Operations (INPO) good practices as incorporated into station procedures for housekeeping and low-level radioactive waste management.

The licensee has committed to following the recommendations for sorting and handling of materials provided in NRC Inspection and Enforcement (IE) Circular No. 81-07, "Control of Radioactively Contaminated Materials" [Ref. 20] and NRC IE Information Notice 85-92, "Surveys of Waste Before Disposal From Nuclear Reactor Facilities" [Ref. 21].

4.3.2 Waste Handling and Packaging

Table 10 summarizes Shoreham waste volume, handling, and packaging and provides data on waste volumes, number and types of containers expected to be used, and the number of shipments. The control rod blades comprise an addition 1115 cubic feet of radioactive waste. In addition, there are 31 LPRMs that will be cut for packaging. The hot ends of the LPRMs will be packaged with 20 antimony pins and 6 beryllium sleeves in a 17 cu ft liner. The 138 control rod blades are intended to be packaged four to a liner [Ref. 6].

Table 10

Shoreham Waste-Volume Analysis

Description	Volume Basis(ft³)	Typical Container Type	Container Quantity	Truck Shipments
FACILITIES				
Primary Containment	30	55-Gal	4	
Floor Drains & Floor Drain Sumps	206	B-25	2	
Radwaste Laydown Area	15	55-Gal	2	
REACTOR VESSEL				
Shell	2,060	None	Coating & Pallets	30
Internals:				
Type A LSA • Core Shroud • Top Guide Plate • SRM/IRM Dry Tubes	1,898	Nominal 120ft³ Liners	13	13
Type A LSA • Core Support Plate • Jet Pumps • Control Rod Drive Guide Tubes	711	Nominal 195ft³ Liners	3	3
Surface Contaminated				
All Remaining LSA	5,459	B-25	53	5
Internal Components	8,416	Cargo Container	6	3
SYSTEMS				
Control Rod Drive	515	B-25	5	1
Core Spray	1,545	B-25	15	2
Residual Heat Removal	15,141	B-25	147	17
Reactor Water Cleanup	9,167	B-25	89	10
Fuel Pool Cleanup	2,472	B-25	24	3
Condensate Demineralizer	1,957	B-25	19	2
Reactor Recirc.	5,974	B-25	58	7
Liquid Radwaste	5,974	B-25	58	7
Sampling	309	B-25	3	
MISCELLANEOUS				
Demineralizer Resins/ Filters	3,212	HIC	22	8
Fuel Racks & Appurtenances	8,258	Cargo Container	25	14
Process Waste and DAW	7,725	B-25	75	4

During the decommissioning of Shoreham, radioactive waste will be generated as a result of implementation of each of the major tasks and activities. Sections 4.1 and 4.2 provide discussions of the dismantlement process, and list the contaminated systems in the Reactor and Radwaste Buildings that will be dismantled. Small bore piping (less than 3-inchs diameter) will be cut and relocated either manually or by use of push cart, to staging area containers located near building hatchs. When the staging area containers reach capacity, they will be moved to the building's truck bay by crane [Ref. 6].

Large bore piping (greater than 3-inches in diameter) will be moved directly from its removal location to the truck bay, using a combination of rigging, cranes, and push carts. Components and instrumentation will be end capped or bagged, on removal from the systems, for contamination control, then handled in the same way as small bore system piping. As the staging containers and large-bore piping arrive at the truck bays, they will be loaded into "C" cargo vans for removal offsite to a volume reduction facility [Ref. 6].

The licensee intends to decontaminate the RPV head to releasable levels (decontaminated by wiping) in the cask wash down area. After decontamination, the RPV head will be rigged and removed from the building, using the polar crane and placed in the yard. The steam dryer, core spray piping, moisture separator, feed water sparagers, dryer/separator guide rods, jet pumps, "cold" portions of the in core guide tubes, and core plate will be removed from the RPV and further segmented in either the DCS or the WCS, depending on the radiological conditions [Ref. 6]. The segmented pieces will be cut and put into a "C" Van, which will be located either on the 175-ft elevation of the Reactor Building, or in the truck bay. When filled, the "C" Van will be sent to an offsite volume reduction facility [Ref. 6].

The RPV internals that are the most radioactive (i.e., portions of the in core guide tubes, and the shroud rings) will be removed from the RPV to the WCS, for segmentation. The segmented pieces will be loaded underwater directly into a liner. The licensee is still evaluating the feasibility of underwater loading of the liner into a cask. It is highly probable that the liner will be required to be loaded on the refueling floor of the Reactor Building 175-ft elevation, and that the licensee will be required to devise methods to keep radiation exposures low. At the completion of the loading of the liner into the shipping cask, the cask will be removed from the Reactor Building, using the polar crane. It is the licensee's intention to ship segmented reactor internals to an offsite disposal facility [Ref. 6]; (this option will remain viable through 1992).

It is the licensee's intention to decontaminate the RPV shell, then segment it into rings. The rings will be further segmented in the DCS into smaller pieces. The segmented pieces will be evaluated to determine the activation levels. Pieces that have activation levels below the levels specified in 49 CFR 173.425 (c) will be packaged as bulk shipments of LSA, and shipped in an exclusive use closed transport vehicle. If the activation levels are greater than the levels specified in 49 CFR 173.425 (c), the pieces will be packaged in accordance with the applicable regulations. Pieces not activated or contaminated will be removed from the building and scrapped. It should be noted that the disposal options noted above are applicable only through December 31, 1992; after that date, the licensee must make provisions to store these materials in segmented pieces, or in a volume reduced form onsite, in interim storage [Ref. 6].

Equipment used during the decommissioning such as the WCS, DCS, and work platforms will be dismantled, and attempts will be made to decontaminate. However, materials that cannot be decontaminated will be packaged for disposal [Ref. 6].

4.3.3 Waste Transportation and Disposal

The licensee intends to ship waste to the available disposal sites until December 31, 1992, when they may be closed to Shoreham. The licensee addressed the radioactive waste transportation and disposal issues in its DP [Ref. 3] and in its responses to NRC questions [Ref. 11]. The licensee anticipates that radioactive waste will be shipped either by trailer or shipping cask. Table 11 provides a summary of transportation needs. A discussion of waste transportation requirements for the major decommissioning activities and tasks follows.

4.3.3.1 RPV Internals

The core shroud, top guide plate, and SRM/IRM dry tubes will be further segmented after removal from the reactor vessel. The segmented pieces will be verified as LSA materials, placed in NRC-approved Type A containers, in accordance with 10 CFR 71.52, and shipped by exclusive use carriers. An estimated 13 casks and liners will be required to transport this waste, and the resulting burial volume will be approximately 1900 cu ft.

Based on data provided in Table 11, the licensee determined that the core support plate, jet pumps and control rod drive tubes could be shipped as Type A LSA materials, in accordance with 49 CFR 173.425. The estimated burial volume of these components is 711 cu ft.

4.3.3.2 RPV Shell

The RPV ring segments will be further segmented in the DCS into sizes consistent with shipment in exclusive use vehicles. The segments will be either wrapped or enclosed, and transported as Type A LSA bulk shipments per 49 CFR 173.425(c). The estimated burial volume is 2060 cu ft. The material is expected to be Class A waste, per 10 CFR Part 61.

4.3.3.3 Contaminated Systems

Nine systems were identified in the Site Characterization Program [Ref. 10] as being contaminated at levels in excess of Regulatory Guide 1.86 [Ref. 12]. None of these systems was found externally contaminated, when surveyed in accordance with 10 CFR 20.201. The estimated curie content indicates that all system piping external to the RPV will be classified in accordance with Part 61, as Class A, and transported as Type A LSA, per 49 CFR 173.425.

4.3.3.4 Spent Fuel Racks and Appurtenances

The spent fuel storage racks are required as long as there is fuel in the spent fuel storage pool, and the racks will not be radiologically characterized until the fuel is removed from the spent fuel pool. Upon fuel removal from the spent fuel storage pool, the internal cells will be decontaminated, using high pressure mole nozzles. The external surfaces will be decontaminated using ultrahigh pressure water. After decontamination, the racks will be surveyed to determine whether they met the release criteria.

4.3.3.5 HEPA Filter Waste and DAW

HEPA filters will be changed out and treated as radioactive waste. In addition, solid dry radioactive waste will be generated as a result of cutting operations, cleaning, maintenance activities, and use of consumable waste generated by personnel working in radioactively contaminated areas. As noted in Table 10, the total volume of dry process waste (HEPA filters and miscellaneous other dry waste) and DAW is estimated to be 7700 cu ft. The licensee expects the total radioactivity to be minimal; however, if HEPA filters used to control radioactivity, during cutting operations, during the decommissioning, are contaminated at measurable activity levels, the licensee will package and dispose of this material in accordance with applicable NRC and Department of Transportation (DOT) regulations.

4.3.3.6 Demineralizer Resins and Filters

Filtration and ion exchange processing will be required to remove radioactivity from water. The volume of spent ion resins and filters required to be used to process the estimated 620,000 gallons of water used during the decommissioning of Shoreham is approximately 3200 cu ft. Filters used to maintain the clarity of water in the WCS and the RPV have the potential to accumulate radioactivity at measurable levels. If this occurs, the licensee will package and dispose of this material in accordance with the applicable NRC and DOT regulations.

4.3.3.7 Control-Rod Blades

The control rod blades will require 35 liners, each with a volume of 32 cu ft for shipment. The control rod blades are expected to be Class A, in accordance with Part 61, and are expected to be shipped as LSA Type A materials, in an exclusive use vehicle, in accordance with 49 CFR Part 173 [Ref. 6].

4.3.3.8 LPRM Antimony Pins and Beryllium Sleeves

The LPRMs, antimony pins, and beryllium sleeves will be packaged in a liner for shipment. These components are expected to be packaged Class B waste, per Part 61, and to require shipment in a Type B container [Ref. 6].

Shoreham will have access to low level radioactive waste (LLRW) disposal at facilities in Barnwell, South Carolina; Beatty, Nevada; and Hanford, Washington, through December 31, 1992. The State of New York currently has a program to develop a New York low-level radioactive waste disposal facility. New York's facility will not be available by January 1993.

4.3.3.9 Onsite Storage of Low-Level Radioactive Waste (LLRW)

It is likely that radioactive waste generated during the Shoreham decommissioning will require onsite storage. Because of the uncertainties related to disposal alternatives for LLRW for the post 1992 period, the staff required the licensee to perform a bounding analysis for onsite storage of LLRW. The licensee's bounding conditions were based on the entire volume of LLW expected to be generated during the decommissioning, and the NRC design guidance provided in NUREG-0800. The NRC provided guidance to all licensees related to the storage of low-level radioactive materials onsite in NUREG-0800 Appendix 11.4-A, and to assure industry wide distribution. This guidance was enclosed in Generic Letter 81-38 "Storage of Low-Level Radioactive Waste at Power Reactor Sites." The licensee determined based on their analysis of space in the Radwaste Building, that the entire volume of solid radioactive waste generated (79,300 cubic feet containing approximately 600 Ci) during the decommissioning of Shoreham could fit into the Radwaste Building [Ref. 6]. Based on that volume of waste, the licensee committed to using the Radwaste Building [Ref. 6] to store LLW during the post 1992 period, if necessary.

The Radwaste Building was designed with features that support the handling and storage of radioactive materials, and is a seismic Category I structure. The staff evaluated the Radwaste Building as a storage location for solid radioactive waste in NUREG-0420 [Ref. 25]. The staff's evaluation in NUREG-0420 was based on using only 1500 square feet of storage area during power operations. The licensee's analysis included an evaluation of potential storage locations for the entire volume of waste and the identification of storage location within the building, evaluations of floor loading limits, and the identification of building modifications that would be required to accommodate the entire volume of waste.

The staff reviewed the licensee's proposed waste handling and packaging methods, the waste categories, waste transportation, the radioactive content of the waste, and onsite storage alternative, and found them to be acceptable based on the guidance provided in NUREG-0800 and applicable parts of the regulations.

The licensee assumed in its bounding analysis required by the staff that the storage of LLRW in the Reactor Building would be for a period less than 5 years [Ref. 6]. If it appears that onsite storage of low-level radioactive waste generated during decommissioning will exceed 5 years (5 year period infers short-term interim storage and is referenced in Generic Letter 81-38), the licensee will be required to make an application pursuant to 10 CFR Part 30, for a license amendment (authorizing long-term storage). Release of the site for unrestricted use will not be possible until all waste has been removed from the site, the storage area has been decontaminated, and the final survey has been completed in accordance with 10 CFR 50.82(f).

Future NRC's reviews based on the requirement of 10 CFR Part 30 will consider container integrity and retrievability, and the implications of extended onsite storage. Information needs for a 10 CFR Part 30 amendment authorizing extended onsite storage were distributed by NRC in an NRC Information Notice 90-09.

4.4 Area Cleanup and Decontamination

After completion of the decontamination and dismantlement of Shoreham's contaminated and activated systems, structures, and RPV and internals, a general radiological assessment of the work areas will be performed. A radiological survey will be conducted, and additional decontamination, using appropriate techniques, will be conducted.

4.5 Final Radiation Survey

A final radiation survey will be performed to demonstrate that the Shoreham site, or portions of the site may be released for unrestricted use. The final radiation survey will be used to demonstrate and document that contaminated materials, structures, systems, areas, and components have been successfully removed or decontaminated to levels defined in the release criteria for unrestricted use, as defined in Table 1 of Regulatory Guide 1.86 [Ref. 12], for both removal and fixed-surface contamination, and the 5 uR/hr above background limit at 1 meter, for gamma-emitting isotopes.

The licensee prepared the final radiation survey plan in accordance with guidance in NUREG/CR-2082, "Monitoring for Compliance with Decommissioning Termination Survey Criteria," [Ref. 22]; Regulatory Guide 1.86, "Termination of Operating License for Nuclear Power Reactors," [Ref. 12]; NUREG/CR-2241; "Technology and Cost of Termination Surveys Associated with Decommissioning of Nuclear Facilities" [Ref. 23]; and NUREG-0586; "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" [Ref. 7]. Data points from the Shoreham Characterization Program "Final Report" (SCPFR) [Ref. 9] are used in the final radiation survey, as much as possible.

The plan submitted by the licensee contained an outline of the final survey report, a comprehensive list of areas, in the plant, that are currently contaminated, and a list of areas with the potential for becoming contaminated, during the decommissioning. The NRC staff's review of the licensee's final termination survey will focus on these areas. The termination survey will include measurements of the direct and removable beta-gamma contamination, exposure rate measurements at 1 meter, and limited direct and removable alpha contamination measurements.

At the conclusion of the final survey, the licensee must show that the areas surveyed meet the release criteria and are suitable for release for unrestricted use. If onsite radioactive waste storage is required, these areas of the plant cannot be released for unrestricted use until the stored radioactive materials are removed, and the area can be shown to meet release criteria. NRC will conduct confirmatory surveys to verify that the licensee's final surveys are complete and accurate.

4.6 Project Schedule

The estimated duration of the decommissioning of the Shoreham plant is 27 months. Decommissioning of contaminated systems, and RPV segmentation and fuel removal operations are intended to be performed in parallel. Fuel disposition issues are in the critical path. Fuel removal from the site, although not a part of the decommissioning will affect the decommissioning schedule. Fuel storage onsite in the spent fuel pool will affect dismantlement and decontamination activities in the Reactor Building.

4.7 Radiological Control

Radiological controls at Shoreham are under the direction of the Radiological Controls Manager. Radiological controls will be implemented using applicable existing plant procedures. The Radiological Controls Manager will review all procedures to ensure that they are applicable to decommissioning activities and consistent with the licensee's ALARA considerations. The Shoreham Offsite Dose Calculation Manual (ODCM) [Ref. 24] will govern the controlled release of radioactive materials through liquid and airborne effluent pathways. The ODCM specifies the monitoring instrumentation that is required to be operable; alarm/trip-set points; required surveillances; permissible radionuclide concentrations in liquid and airborne effluents; permissible doses and dose commitments to members of the public, in unrestricted areas; Radiological Environmental Monitoring Program (REMP) provisions; and administrative controls. LIPA committed to implement the LILCO methods defined in the ODCM in Shoreham responses to NRC questions [Refs. 6 and 11].

4.7.1 Effluent Release Controls

Liquid radioactive waste generated during the decommissioning will be processed using portions of the plant's installed liquid radwaste system, as described in Section 4.3.1.2. Processed liquids will be discharged after they have been monitored and approved for release, in accordance with the Shoreham ODCM. The contaminated resins resulting from liquid radioactive waste processing will be dewatered in HICs, and shipped to licensed burial facilities, or placed in onsite interim storage.

Because short-lived radionuclides have decayed to insignificant levels, there will be no need to process gaseous fission products with the plant's existing gaseous waste treatment systems, during the decommissioning. The plant's existing Reactor Building ventilation system, as described in the Shoreham USAR [Ref. 1] and approved by the NRC staff in NUREG-0420, "Safety Evaluation Report Related to the Operation of Shoreham Nuclear Power Station, Unit No. 1" [Ref. 25], will be used for control of airborne releases. Airborne particulates generated during cutting in the Reactor Building will be controlled with the ventilation/fume collection systems (system includes HEPA filters and fans) associated with the DCS and WCS. In addition, during the decommissioning, when vacuum cleaners are used in radiologically controlled areas, they will be equipped with built-in HEPA filters. Airborne particulate releases from Shoreham will be controlled using methods defined in the ODCM.

Solid radioactive waste will be generated as a result of decontamination and dismantlement of activated and contaminated systems, components, and structures. It is the licensee's intent to ship as much radioactive waste as possible before January 1, 1993, in accordance with applicable NRC and DOT regulations, to the available offsite disposal sites, before closure of these sites to the State of New York. Radioactive waste remaining onsite or generated after December 1992 may require onsite interim storage.

4.7.2 Worker Exposure Control

During all phases of the decommissioning, ALARA engineering and administrative controls will be evaluated, to minimize radiation exposure to individual workers. The following techniques and controls are defined in the licensee's DP [Ref. 3], and are anticipated to be used during decommissioning:

- Airborne radioactivity will be monitored and controlled during all decommissioning activities. Precautions will be taken to prevent any unintentional release of airborne contamination, when any system containing radioactivity is opened.

- Decontamination and dismantlement activities will be controlled to minimize radiation exposure through the use of RWPs. All tasks will be governed by procedures and reviewed for ALARA requirements.

- Personnel will be protected against airborne contamination by using Health Physics (HP) controls and by use of portable ventilation exhaust systems containing HEPA filters. When it is impractical to apply engineering controls, respiratory protective devices will be used. Filtered ventilation systems will always be used in areas where cutting or grinding of contaminated systems is planned.

- All components scheduled for disassembly will initially be assumed to be contaminated. Radiological surveys will be performed to document the conditions found. RWPs will be written, and work controls established, if contamination is found.

- Cutting techniques will be used that have rapid setup, deployment, and easy cleanup.

- Radioactive material storage areas will be used to ensure physical protection of personnel.

- Before performing decommissioning activities in or near radioactive systems and structures, the merits of ALARA alternatives will be considered.

- All preliminary work will be performed, where possible, in areas well isolated from radioactive materials.

- Preplanning of all work activities will be performed for projected high exposure jobs. Mock-ups or dry runs will be conducted, especially when extremely complex tasks are performed. Mock-up and dry runs will be triggered when dose rates are greater than 100 mrem/hr, when the job will result in an individual exposure 2 MPC-hr or greater in one day, and when the task will expend greater than 1 man-rem [Ref. 6]. All work involving high dose rates and contamination levels will require preparatory meetings that will be attended by HP personnel, the foreman, and workers directly involved with the job. After completion of the job, debriefing sessions will be held, and experience gained will be used to plan future similar work evolutions.

4.7.2.1 Health Physics Program (HPP)

The licensee's HPP will be used to translate the licensee's commitment to ALARA objectives into action. LIPA has adopted the existing LILCO/Shoreham HPP essentially in its entirety. The HPP is described in Chapter 12, "Radiation Protection," of the Shoreham Defueled Safety Analysis Report (DSAR) [Ref. 19], Revision 3, and associated referenced sections of the Shoreham USAR [Ref. 1].

In addition to the HPP described in the DSAR, the licensee has committed to reinstitute the policy of issuing dosimetry to all personnel requiring access to the radiological controlled areas, and to re-establish the respiratory cleaning and drying facility.

As noted in its response to NRC questions [Ref. 11], LIPA has adopted LILCO's current operating procedures, including HP procedures, with only minor changes to reflect LIPA's status as Shoreham's owner and licensee. The HPP will consist of all actions and measures planned to protect workers and the environment. As a part of the HPP, steps are taken to monitor radiation and radioactive materials; to control the distribution and release of radioactive materials; and to keep radiation exposure to within the limits of 10 CFR Part 20. To control radioactive materials in various areas during the decommissioning, the licensee plans, as noted in its DP [Ref. 3], to take the following actions:

- Perform radiation and contamination surveys.
- Use RWPs to delineate controls, identify conditions, and specify protective measures to prevent inadvertent exposure of personnel to radiation or radioactive contamination, during decommissioning activities.
- Arrange the available work areas into contaminated and clean working areas.
- Institute and implement access controls to: 1) control the spread of contamination from contaminated to "clean" areas, and 2) limit RWP area access only to personnel who are directly involved in the specific task.
- Clearly identify and tag all contaminated items as they are removed, and note their place of origin and pertinent radiological information.
- Monitor work areas so as to alert personnel to any unexpected radiological conditions.
- Maintain accurate and updated records of personnel exposure, surveys, and lessons learned, to improve and revise procedures, as necessary.
- Monitor effluent waste streams with appropriate radiation monitoring equipment, surveys, and sampling.

4.7.3 Environmental Monitoring

The licensee has committed to maintain Shoreham's REMP, described in Section 11.6 of the DSAR, Revision 3 [Ref. 19] during decommissioning. The referenced REMP reports the results of comprehensive measurements of radioactivity concentrations in terrestrial, aquatic, and atmospheric media, as well as direct radiation in the vicinity of Shoreham. Environmental measurements extend to a 20-mile radius from the site. Measurements from the Shoreham Radiological Program have not detected radioactivity above background in air, precipitation, ground water, nor soil.

Radionuclides were detected in Shoreham's liquid effluents during the 1985-87 operating period. The concentrations of radionuclides in the liquid effluent streams were well below regulatory limits, as defined in Part 20.

4.7.4 Unrestricted Use Criteria

The licensee's DP [Ref. 3] calls for dismantlement and decontamination of the Shoreham plant to the levels defined in Table 1 of Regulatory Guide 1.86 [Ref. 12]. Regulatory Guide 1.86 establishes limits for both removable and fixed surface contamination. Also, the exposure rate from gamma emitting isotopes must be less than 5 uR/hr above background, at 1 meter from equipment surfaces and soil.

4.8 <u>Employee Staffing and Training</u>

4.8.1 Employee Staffing

LIPA is a corporate municipal and political subdivision of the State of New York. As referenced in Section 4.8, seven key positions vital to the decommissioning of the Shoreham plant are filled by personnel from the NYPA (i.e., "LIPA/NYPA Coemployees"). To carry out its mandate to decommission the Shoreman plant, the LIPA organization will be staffed by personnel from its prime contractor NYPA (not coemployees), and personnel from LILCO. NYPA contractor personnel will be used by LIPA to provide technical and management services.

4.8.2 Employee Training

All decommissioning personnel at Shoreham, whether employed by LIPA, NYPA, LILCO, or other contractors, will receive appropriate training commensurate with the potential hazards to which they may be exposed.

Records of training as a minimum will include the following: a) employee name; b) subject of training and brief description; c) date, time, and duration of the training; d) written examination; e) instructor's name; and f) training expiration date. These records will be kept in accordance with 10 CFR Parts 19 and 20.

General Employee Training (GET) will be provided to employees in the following subject areas, commensurate with their job duties:

- General description of the plant and facilities
- Job related policies, procedures, and instructions

- Radiological health and safety
- Emergency plan
- Industrial safety
- Fire protection
- Security
- Quality Assurance (QA)

Individuals that are required to have routine access to radiologically controlled areas, or who will routinely handle radioactive materials or contaminated equipment must satisfactorily complete GET, including appropriate radiological safety training, before being granted unescorted access to radiological controlled areas. The Radiological Safety Course will include the following training:

- ALARA practices
- Introduction to Parts 19 and 20
- Prenatal radiation exposure (Regulatory Guide 8.13, Rev.1) [Ref. 26]
- Radiological instrumentation and controls
- Decontamination and radwaste procedures
- Fire protection
- Respirator training (for appropriate employees) in accordance with
  - Regulatory Guide 8.15, Rev 1 [Ref. 27]
  - NUREG-0041 [Ref. 28]

In addition, general subjects, such as the nature of radiation, method of controlling contamination, interaction of radiation and matter, biological effects of radiation, use of monitoring equipment, and risk from occupational exposure will be covered in the Radiological Safety Course.

4.9 <u>QA</u>

The licensee will establish and implement a QA program for the decommissioning of Shoreham. In general, the licensee's proposed QA program, as presented in DSAR Section 17.2, Quality Assurance During the Operation Phase, [Ref. 19] will be adapted for use during the decommissioning. The licensee's decommissioning QA program will be implemented to address the following areas:

- Protection of the radiological health and safety of the public, project personnel, and the environment

- Exposure to radiation

- Adherence to NRC regulations

- Design, procurement, fabrication, and operation of decontamination equipment

- Design, procurement, erection, testing, and operation of specialty/engineering equipment for dismantlement and disposition of contaminated equipment

- Control of radioactive material and contamination

- Shipment of radioactive waste

- Control of activities for site characterization for decommissioning engineering

The requirements in the Shoreham QA program are mandatory and will be imposed on all personnel and organizations, including contractors, who perform Shoreham plant decommissioning activities.

4.9.1 QA Organization

The Nuclear Quality Assurance (NQA) Manager reports to the Executive Vice President-Shoreham Project and is responsible to the Resident Manager, for the development and administration of the Decommissioning QA program. The Executive Vice President-Shoreham Project, will keep the Executive Director and the LIPA Chairman and Board of trustees apprised of significant QA developments.

The Resident Manager has the overall responsibility for the implementation of the QA program. The Resident Manager will also have the responsibility for implementation of QA program requirements for maintenance and operation of safety related structures, systems, and components, as defined in DSAR Section 17.2 [Ref. 19].

The NQA Manager is responsible for directing the activities of the Quality Control (QC) and Quality Systems (QS) Managers. His principal objective is to ensure that the Shoreham plant and all support organizations establish and conform to adequate standards and procedures, in accordance with the QA manual. The NQA Manager is authorized to stop work when circumstances warrant. The NQA Manager's primary duties and responsibilities include the following [Ref. 3]:

- Development and administration of the overall QA program to be applied during the decommissioning of the Shoreham plant

- Define the content of, and changes, to the Quality Assurance Manual (QAM)

- Maintain a working interface and communications with other organizations, regulatory agencies, consultants, contractors, inspection firms, etc.

- Be responsible for the continuous implementation of the QA training program for QA and other project personnel

- Review and approve applicable documents, to ensure the inclusion of appropriate quality requirements

- Performance of audits to evaluate and implement the QA Program

- Participate on the Site Review Committee (SRC)

The NQA Manager will have at least 5 years experience, requiring technical and administrative abilities in nuclear related QA, engineering construction, or operations. At least 2 years of experience will be in nuclear QA services. The description of the licensee's QA organization is provided in the DP [Ref. 3], and supplemented in Ref. 11.

4.9.2 QA Program

The licensee's (LIPA) QA program will be derived from the existing LILCO QA program. The QA program will consist of two sets of requirements documents -- the DSAR Section 17.2 [Ref. 19] requirements for safety related structures, sys tems, and components that are needed to maintain Shoreham until the nuclear fuel is removed from the site; and the QA requirements for decommissioning activities. The QA requirements for decommissioning will be developed by the licensee, to establish the overall quality requirements that will govern the implementing documents (i.e., procedures).

The QAM will describe how compliance with appropriate quality and safety requirements will be accomplished. Procedures will be developed that provide specific controls and instructions for performing the decommissioning activities specified in Section 4.10. The process of procedure review and approval will be formally designated in a written administrative procedure.

4.10 Financial Assurance

The estimated cost to decommission the Shoreham Nuclear Power Station, Unit 1, is $186,292,000, in 1991 dollars. On November 22, 1991, the U.S. NRC issued an exemption to LILCO, exempting LILCO from the conditional requirements for the use of a surety method as financial assurance, specified in 10 CFR 50.75(e)(iii) (A), (B), and (C). The exemption was granted under the conditions that: (1) LILCO provide funds to an external account sufficient to cover, at all times, three months of projected decommissioning cost, as specified in the January 24, 1990 Site Agreement; (2) LILCO will maintain a $10-million external fund for emergency decommissioning costs; (3) notice be given to NRC, at least 90 days in advance in the event of cancellation or alteration of the $300-million line of credit; and (4) LILCO will maintain and commit an amount of its unused line of credit during decommissioning of Shoreham, sufficient to cover estimated, and yet to be incurred, decommissioning costs.

The Shoreham labor rates were compared to the labor rates for Pathfinder and escalated at 5 percent per year to 1991 dollars, and compared with cost in "1992 Means Building Construction Cost Data" [Ref. 29] using the City Cost Indexes. Based on the cost comparisons referenced, the Shoreham costs were found to be reasonable. The staff finds the Shoreham disposal cost of $240.00 per cubic foot to be conservative. The staff finds, in general, that the Shoreham decommissioning cost estimate for the proposed method of decommissioning is conservative.

4.11 Emergency Planning

The licensee's DP [Ref. 3] includes a section on accident analysis. The onsite postulated accident scenarios were included in the DP, and are listed below:

- Waste Container Drop
- Combustible Waste Fire
- Contaminated Sweeping Compound Fire
- Vacuum Filter-Bag Rupture
- Oxyacetylene Explosion

- Explosion of Liquid Propane Gas Leaked from a Front-End Loader
- Contamination Control Envelope Rupture
- Fuel Damage Accident
- Effects of Natural Catastrophes
- Breach of Physical Security Measures

The accidents listed above were all analyzed based on a dose commitment to the public or a worker.

4.12 Physical Security

The licensee's DP (Ref. 3) will use the NRC-approved versions of the "Shoreham Nuclear Power Station Security Plan for Long-Term Defueled Condition," and Fuel Storage in Spent Fuel Pool, (Ref. 30) "Guard Training and Qualification, and Shoreham Nuclear Power Station Safeguard Contingency Plan" (Ref. 31) as required by Paragraph 2.E of Facility Operating License No. NPF-82.

4.12.1 Site Security Organization

The Nuclear Security and Training Division Manager reports to the Nuclear Operations Support Department Manager and is responsible for the physical security of the site and environs. He will be responsible for establishing procedures and standards for controlling access to the site, for staff and contractor personnel, as well as vehicle access control. LILCO will retain the title to the Owner Controlled Area, except for the 11 acres which were transferred to LIPA. Patrols of these areas will jointly be managed by LIPA and LILCO.

4.12.2 Physical Security Measures

Upon license transfer the Owner Controlled Area will remain as described in the "Shoreham Nuclear Power Station Security Plan for Long-Term Defueled Condition," and Fuel Storage in Spent Fuel Pool (Ref. 30). The Protected Area consists of the Refueling Deck, elevation 175-feet, of the Reactor Building.

5.0 ENVIRONMENTAL IMPACT OF PROPOSED ACTION

5.1 Radiological Impact to the Public and Workers

Radiological impacts to members of the public will accrue from releases of radioactivity in air and water released to unrestricted areas, and direct exposure to radioactive materials during transport for disposal. Decommissioning workers will accrue radiological impacts primarily by being directly exposed to radiation sources at or near work areas. In addition to impacts from routine operations, the potential radiological consequences to the public, from potential accidents, were evaluated.

5.1.1 Radiological Impacts to the Public

Liquid waste generated during decommissioning results from the work in the WCS in the RPV from draining the spent fuel pool, and from decommissioning the nine major contaminated systems. The projected quantities of radioactivity to be released to the environment during decommissioning, assuming the liquid waste is processed once in the radwaste facility, with a 95 percent overall efficiency, are 4.4×10^{-3} μCi Co-60, 1.2×10^{-8} μCi Mn-54, and 1.1×10^{-6} μCi Fe-55.

All potentially contaminated liquids generated during decommissioning will be collected, monitored, and processed before release. Through the ODCM (Ref. 24), the licensee has committed to use the radwaste system to reduce releases of radioactivity when the projected doses, due to the liquid effluent, to unrestricted areas would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ in a 31-day period. Therefore, annual dose to a member of the public is limited to 0.72 mrem total body and 2.4 mrem to any organ, during decommissioning. The licensee estimated the dose from projected effluents released during decommissioning, using the methods in Shoreham's current ODCM, Section 3.1.2, "Method 2 (Backup Method)," to be 0.15 mrem to the child total body and 1.1 mrem to the adult organ (GI tract). These doses are 5.0 percent, and 11.0 percent, respectively, of the Appendix I annual design objectives for liquid effluents.

Airborne releases of radioactivity during Shoreham decommissioning have been estimated based on the estimated airborne releases due to RPV segmentation in the DCS and the estimated airborne releases due to all other Shoreham decommissioning activities. Table 11 lists the estimated radionuclide releases due to decommissioning of Shoreham.

Table 11

Airborne Radionuclide Releases from Shoreham Decommissioning

Curies Released

Nuclide	RVP Segmentation	Other Activities	Total
Co-60	1.28 E-4	2.02 E-4	3.30 E-4
Mn-54	0.00 E-0	5.48 E-6	5.48 E-6
Fe-55	3.11 E-4	8.85 E-6	3.20 E-4
Ni-63	9.61 E-6	2.92 E-7	9.90 E-6
H-3	1.24 E-4	0.00 E-0	1.24 E-4
C-14	2.48 E-5	0.00 E-0	2.48 E-5
Ni-59	6.50 E-8	0.00 E-0	6.50 E-8

The gaseous and airborne particulate radwaste systems in place at Shoreham were evaluated in NUREG/0420 [Ref. 25] and determined to be capable of maintaining releases of airborne particulate radioactive material such that the annual dose to any organ of a member of the public will be less than 15 mrem/yr, in accordance with 10 CFR Part 50, Appendix I. This evaluation assumed a release of approximately 100 times more radioactivity than is estimated to result from decommissioning. These gaseous radwaste systems and associated effluent monitoring systems will remain in place during decommissioning.

Doses from projected airborne effluents were calculated as 50-year committed dose equivalents, in accordance with Regulatory Guide 1.109, "Calculation of

Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluation Compliance With 10 CFR Part 50, Appendix I," Revision 1, October 1977 [Ref. 32]. Inhalation rates and other factors in the calculations are taken from the Shoreham Offsite Dose Calculation Manual [Ref. 24]. The calculated maximum whole-body dose is 1.82 E-5 millirem to a child. The calculated maximum organ dose is 2.03 E-4 millrem to the lung of a teenager. When the 50-year organ dose is compared to the 10 CFR Part 50 Appendix I design objectives, for the release of radioactive particulates to the atmosphere, it is shown to be less than 0.002 percent of the Appendix I annual exposure limits.

The collective whole-body dose, from planned decommissioning activities, to the population living within 50 miles of Shoreham, is conservatively estimated to be 5.34 E-2 person-rem. The estimate is based on the 1990 summer population and atmospheric dispersion factors reported in Shoreham's USAR [Ref. 1].

The licensee's estimates of radioactivity released to the environment during decommissioning are conservative compared to the reference BWR decommissioning described in NUREG/CR-0672 [Ref. 5]. This conclusion is based on the fact that the radionuclide inventory at Shoreham is 4 orders of magnitude lower than that estimated for the reference BWR. However, the estimated effluents from Shoreham decommissioning are within 1 order of magnitude of those estimated for the reference BWR.

The licensee's estimates of airborne and liquid effluent releases during decommissioning result in doses to the public well below Parts 50 and 20 limits. Thus, the impact on the public, from the release of radioactivity to the environment, and from the activities described in Shoreham's DP [Ref. 3], is acceptable.

5.1.2 Radiological Impacts on Workers

The licensee has estimated that a total of 190 person-rem of occupational exposure will be incurred. Table 12 lists the radiation exposure estimates for each of the decommissioning activities. The occupational dose was estimated using methods from NUREG-0800 [Ref. 33], Regulatory Guide 8.19 [Ref. 34], and NUREG/CR-0672 [Ref. 5]. The Station ALARA Review Committee (SAC) has reviewed the work associated with the entire decommissioning project.

Table 12

Shoreham Decommissioning Occupational Exposure Estimates

Decommissioning/ Activity	Estimated Occupational Dose (man-rem)
Control-Rod Drive System	0.3
Core-Spray System	0.8
Residual-Heat Removal System	2.3
Reactor Water Cleanup System	1.2
Fuel Pool Cooling and Cleanup System	1.7
Condensate Demineralizer System	0.0
Reactor Recirculation System	0.7

Table 12 (Cont'd)

Liquid Radwaste System	0.1
Process Sampling System	0.0
Equipment Floor Drainage System	0.0
Primary Containment	0.2
Floor Drain Sumps	0.0
Dryer and Separator Pool, Reactor Head Cavity, and Spent Fuel Storage Pool	0.1
Spent Fuel Storage Racks	0.1
Radwaste Laydown Area	0.0
Segmentation of RPV	158.0
Radwaste Handling and Packaging	14.2
Radwaste Transport	10.0
Totals	189.7

The dose assessment incorporates the expected average radioactive levels, the manpower requirements, and stay times. Expected average radioactive levels were determined by the use of measured values where available. Calculated values and engineering judgment were used where measured exposure rates were not available. The dose rates are assumed to be generally spatially uniform in the work areas and due primarily to Co-60 gamma radioactivity. Manpower requirements and stay times are consistent with those assumed for the decommissioning cost estimate.

NUREG-0586 [Ref. 7] estimates a total dose of 1845 man-rem resulting from decommissioning of the reference 1155 MWe BWR. Due to the low radionuclide inventory at Shoreham relative to the reference BWR, approximately $6x10^2$ Ci and 6x10 Ci, respectively, the decommissioning of Shoreham is estimated to result in a relatively low 190 man-rem exposure. The licensee has committed to perform a documented ALARA review of all activities expected to result in exposure in excess of 1 man-rem. The review will be performed by the LIPA ALARA Review Committee.

The relatively low radionuclide inventory and the licensee's commitment to ALARA provide assurance that the decommissioning can be accomplished while protecting worker safety and meeting the requirements of Part 20.

5.1.3 Waste Transportation Impacts

The radiation exposure, to the public, from transportation of the Shoreham plant's decommissioning waste to a low-level waste disposal facility, was estimated following methods described in NUREG/CR-0672 [Ref. 5]. The dose calculations in NUREG/CR-0672 [Ref. 5] assume that each shipment contains enough radioactive material to result in the maximum exposure rates allowable by regulation, that is 10 mR/hr at any point 2 meters from the vehicles. This assumption is conservative for the low-level waste shipments expected from Shoreham during decommissioning. The highest estimated exposure rate from

Shoreham waste material, including the shielding provided by the containers, is 10 mR/hr at contact with the shipping container. The majority of the containers to be shipped are estimated to result in contact exposure rates between 0.1 and 0.5 mR/hr.

The licensee used the assumptions from NUREG/CR-0672, [Ref. 5] which are acceptable except for travel distance, -- 1000 miles in place of 500 miles -- and number of shipments -- 80 in place of 1495. Correction for these two changes, the dose to the public, from the transportation of low-level waste was estimated as 0.7 man-rem. Since the total quantity of radioactive waste to be shipped from Shoreham is significantly lower than that for the reference plant in NUREG/CR-0672 [Ref. 5], the 0.7 person-rem estimate is considered to be conservative and is less than the dose to the public from waste transportation estimated for the reference BWR in NUREG-0586 [Ref. 7].

5.1.4 Impacts of Potential Accidents

There were ten different accident analyses described in the licensee's DP. The analyses described in the DP are listed below:

- Waste Container Drop
- Combustible Waste Fire
- Contaminated Sweeping Compound Fire
- Vacuum Filter-Bag Rupture
- Oxyacetylene Explosion
- Explosion of Liquid Propane Gas Leaked from a Front-End Loader
- Contamination Control Envelope Rupture
- Fuel Damage Accident
- Effects of Natural Catastrophes
- Breach of Physical Security Measures

All the accident scenarios described in NUREG/CR-0672 for the reference BWR decommissioning were evaluated, except those involving explosives. In addition, a fuel-damage accident was evaluated by the licensee.

The onsite accident analyses results show that no significant exposure to members of the public would result from onsite accidents. The highest calculated doses were 1.08 mrem whole body and 93.9 mrem to the skin of the maximum exposed individual located at the site exclusion area boundary (EAB) during the worst-case postulated fuel damage accident. The fuel damage accident assumes that the entire inventory of gaseous activity in the fuel, that is 1500 Ci of Kr-85, is released to the atmosphere. The licensee assumed a conservative stability class and wind speed in the dose calculations. Also, dose conversion factors from Regulatory Guide 1.109 [Ref. 32] were used. The staff independently verified the licensee's potential accident dose calculation methods and found them to be acceptable.

The 1.08-mrem whole body dose is 0.1 percent of the 1000 mrem minimum protective action guideline recommended by the Environmental Protection Agency, "A Manual of Protective Action for Nuclear Incidents," EPA 520/1-75-001 [Ref. 35] and 0.2 percent of the 10 CFR 20.105 limit of 500 mrem in unrestricted areas. The health and safety impacts of potential decommissioning accidents are considered to be acceptably low.

Low-level waste that arises from the Shoreham decommissioning will be packaged in accordance with applicable NRC and DOT requirements, and shipped by truck offsite to a low-level waste disposal facility. The licensee based its evaluations of a postulated truck accident on the criteria in NUREG/CR 0672 [Ref. 5].

The licensee applied the analysis of transportation accidents postulated in NUREG/CR-0672 [Ref. 5] directly to the waste to be shipped during Shoreham's decommissioning.

This analysis estimates a dose of 11 mrem whole body from the severe transportation accident, which involves the rupture and burning of 40 Type A packages. The assumptions used in the NUREG/CR-0672 analysis are acceptable for estimating the impact of a transportation accident involving waste from Shoreham decommissioning. The 11 mrem whole body estimated dose is 2.2 percent of the 10 CFR Part 20.105, permissible levels of radiation in unrestricted areas and is acceptable.

5.1.5 Impact on Disposal Site Operation

The total burial volume of the low-level waste generated during the decommissioning of Shoreham is estimated to be 79,300 cu. ft. Presently, Shoreham has access to the operating low-level waste disposal sites. The total available capacity at these disposal sites is in excess of 20 million cu ft. The total volume of waste expected to be generated at Shoreham would represent less than 0.4 percent of the total unused disposal sites' capacity. This is an acceptably small impact based on the relatively small volume of low-level waste to be disposed of compared to the volume available to receive this waste.

It is likely that the States of South Carolina, Nevada, and Washington will not accept New York low level waste after December 31, 1992. In that event, all remaining waste will be stored on the Shoreham site, in interim storage, with the State of New York responsible for development of a disposal alternative.

5.2 Nonradiological Impacts

5.2.1 Socioeconomic Impacts

The Shoreham decommissioning is expected to be completed in 27 months and to employ an average of 590 persons annually, with a peak work force of approximately 650, including management, support staff, and contractor personnel.

Decommissioning is not expected to have any significant effects on the regional economy, because of the limited size of the decommissioning staff and the temporary nature of the the decommissioning project. Therefore, a demographic shift is not expected.

It is anticipated that a significant portion of LIPA's decommissioning staff will be carried over from LILCO's existing-site work force. There are approximately 400 people, employed by LILCO, at the plant. The Shoreham property is not being abandoned. It is anticipated that other activities on the property will be performed after the Shoreham decommissioning.

Decommissioning is not expected to have a significant impact on the local employment and unemployment rates, whether the work force is drawn from the Long Island population or from outside the area. Decommissioning will not place any significant demands on community services. The surrounding communities have sufficient housing capacity to absorb any demands for short-term housing.

5.2.2 Air Quality Impacts

A local short-term deterioration of air quality because of construction vehicle and waste transport vehicle emissions and dust may be unavoidable. The only observable impact, of the decommissioning, on existing air quality, would be minor emissions of fugitive dust. Excavation equipment and equipment movement on unpaved areas will generate dust emissions. Much of the decommissioning project area is paved or covered with buildings or gravel. Water can be used to keep dust emissions down.

5.2.3 Transportation Impacts

All truck traffic from the site will use the licensee's access road that extends from the site to Route 25A. All radioactive waste removed from the site will comply with all packaging, transportation, and disposal regulations. The licensee estimated that approximately 80 truckloads of low-level radioactive waste will result from the decommissioning [Ref. 4]. During the construction phase, the staff evaluated the transportation impacts on the roads in the vicinity of the plant, and concluded that the impact would be minimal [Ref. 36]. The licensee assumed that the 80 truckloads of low-level waste would be removed over the entire 27 months estimated for the decommissioning. Thus, based on the number of truckloads of low-level waste over a 27 month period, the impact of the decommissioning will be less than the impact during the construction phase. Buildings, structures, and equipment that can be released for unrestricted use will not be removed from the site.

5.2.4 Land and Water Use

There are no anticipated negative impacts on land use resulting from the Shoreham decommissioning. Previously disturbed land is available for laydown within the Shoreham site; therefore, no impacts on undisturbed areas onsite or offsite are expected.

Decommissioning of Shoreham will not place any demands on plant water supplies and sewage-treatment services. Water supply and sewage-disposal systems already exist onsite.

5.2.5 Other Impacts

Decommissioning will occur on developed areas of the site; no indigenous vegetation will be removed from the site during decommissioning. Noise and other activities associated with decommissioning are not expected to have significant impacts on wildlife species in the surrounding non-developed areas.

No impact to archaeological or historic sites on or around the licensee's property will result from the decommissioning of the Shoreham plant. This assessment is based on the staff's evaluation in the "Final Environmental Statement Related to Operation of the Shoreham Nuclear Power Station," September 1972, [Ref. 37]. According to the historic resources listed in the National Register of Historic Places [Ref. 13] for the project site, there are no historical sites within a one-half mile radius of the project site [Ref. 4].

Therefore, the impact will be minimal on historical archaeological or historic sites, because the decommissioning work will take place in existing buildings and the truck traffic will be insignificant.

5.3 Conclusion

Based on the staff's review of the Shoreham DP and responses to staff questions, the staff has concluded that there are no significant environmental impacts associated with the proposed action and that the proposed action will not have a significant effect on the quality of the human environment. Therefore, the Commission has determined, pursuant to 10 CFR 51.31, not to prepare an environmental impact statement.

6.0 ALTERNATIVE TO PROPOSED ACTION

6.1 Decommissioning Alternatives

The purpose of decommissioning a nuclear facility is to take the facility safely from service and to remove the associated radioactivity effectively from the environment so that the facility can be released for unrestricted use. The Commission in its Memorandum and Order CLI-90-08, dated October 17, 1990, found that decommissioning actions are directed solely at assuring safe and environmentally sound decommissioning. Therefore, the decision not to operate a plant need not be considered under NEPA. The Commission in its Memorandum and Order CLI-91-02, dated February 22, 1991, reaffirmed this determination. Thus, the NRC is only responsible for approving and supervising the method of decommissioning, not for the decision whether to operate the facility. The licensee has decided not to operate the facility as a nuclear power generating station. Therefore, the licensee can propose to decommission a nuclear power plant using one of three methods: DECON, ENTOMB, or SAFSTOR. Each of these is addressed below, as well as the no-action alternative.

6.1.1 DECON

In the DECON method, equipment, structures, and those portions of the facility containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after operations stop. The proposed decommissioning alternative, DECON, was selected by the licensee for the decommissioning of the Shoreham Nuclear Power Station, and this proposed alternative is discussed in Section 1.3.

The Shoreham Nuclear Generating Station operated intermittently over a 2 year period in a test mode, and had a calculated fuel burnup of approximately 2 effective full power days. The licensee's estimated the occupational dose that would be required to decommission the plant using the DECON decommissioning alternative is 189.7 man-rem [Ref. 3].

Based on the plant's limit operating history, the estimated occupational dose is significantly less than the 1874 man-rem occupational dose estimated for the reference BWR employing the DECON alternative in NUREG-0586 [Ref. 7].

6.1.2 ENTOMB

As described in the GEIS (NUREG-0586), the ENTOMB alternative involves encasing radioactive contaminants in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained, and there is continued surveillance until the radioactivity is removed from the site or decays to a level that permits unrestricted use of the property. Long-lived radionuclides such as niobium-94 are likely to be present in the reactor vessel so that waiting for decay would be impractical. For entombment in some cases, there may be no need for a full-time onsite security guard force and less radiation monitoring and environment surveillance would be required because all radioactive material is contained within the entombment barrier.

If the licensee is unable to remove the fuel from the site, the ENTOMB alternative for Shoreham would require continued surveillance and security. Shoreham operated for a very short period, approximately 2 effective full power days. Other than the irradiated fuel, the primary activity at the plant is contained within the RPV (approximately 602 Ci) [Ref. 3]. Over 97 percent of the radioactivity in the RPV is attributable Fe-55 and Co-60. The decay of these radioactive isotopes would be controlled by Co-60 with a half life of 5.27-year. The ENTOMB alternative would present greater difficulty in monitoring during the entombment period.

Because of the very short operating period at Shoreham, the projected occupational exposures and exposure rates to the public would be significantly lower than the dose estimates for the ENTOMB alternative evaluated for the reference BWR in NUREG-0586 [Ref. 7]. Further, the projected occupational exposures and exposure rates to the public would also be significantly lower than the dose estimate for the reference BWR for both the DECON and SAFSTOR alternatives as referenced in NUREG-0586 [Ref. 7].

6.1.3 SAFSTOR

As described in the GEIS (NUREG-0586), the SAFSTOR alternative involves placing a nuclear facility in a safe condition and maintaining it in that state until it is dismantled and all remaining radioactive materials that would restrict use are removed. The facility may be left intact except that all fuel assemblies should be removed from the reactor and radioactive fluids and wastes should be removed from the site.

The licensee's estimated occupational exposure of 189.7 man-rem for the DECON decommissioning alternative is less than the 834 man-rem estimate for the 10 year SAFSTOR alternative, and the 361 man-rem estimate for the 30 year SAFSTOR alternative for the referenced BWR in NUREG-0586 [Ref. 7].

6.1.4 No Action

As stated in the GEIS, the objective of decommissioning, facilities such as the Shoreham Nuclear Generating Station, is to restore the radioactive facility to a condition such that there is no unreasonable impact from the decommissioned facility on the public health and safety. In order to ensure that the impact on the public health and safety is within the acceptable bounds, some action is required, even if it is as minimal as making a termination radiation survey to verify the radioactivity levels. Thus, the No Action alternative is not a viable decommissioning alternative, because the NRC's regulations will not allow a licensee to simply abandon or leave a facility after ceasing operations.

6.1.5 CONCLUSION

The low levels of contamination at the Shoreham facility and the corresponding low dose estimates based on the level of contamination make the ENTOMB, the SAFSTOR, or the DECON decommissioning alternative viable. Further, because of the low contamination levels at Shoreham, delayed action provides little advantage over the DECON decommissioning option selected by the licensee. Also, the DECON decommissioning option may be less expensive than a delayed action, given the uncertainties related to the cost of maintaining the plant over a long period, and the uncertainties related to the cost of future waste disposal.

Based on the licensee's intention of removing the fuel from the site, the low doses anticipated for the DECON alternative, and because the DECON alternative does not significantly impact on the environment, the licensee's selection of the DECON alternative is acceptable.

The reactor vessel will be segmented, and packaged for shipment as radioactive waste to an offsite disposal site. Other waste generated during the decommissioning will also be packaged in accordance with NRC and DOT regulations for shipment to an offsite disposal site. Shoreham has access to three offsite disposal sites until January 1, 1993. It is anticipated that radioactive waste (including the RPV and its internals) generated during the decommissioning will be packaged for shipment, and will be transported to available disposal sites by trailer in appropriate packages.

The transportation methods for both the fuel disposal option are acceptable, and within the bounds of NUREG/CR-0672 [Ref. 5]. The occupational dose for both options is less than 10 person-rem [Ref. 6].

7.0 AGENCIES AND PERSONS CONSULTED, AND SOURCES USED

This EA was prepared entirely by NRC staff, primarily within the Office of Nuclear Material Safety and Safeguards, Rockville, Maryland. No other agencies or persons were consulted, and no other sources of information were used beyond those that are referenced in the report.

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