HBPP UNIT 3 DEFUELED SAFETY ANALYSIS REPORT, REVISION 5

DEFUELED SAFETY ANALYSIS REPORT

FOR THE

HUMBOLDT BAY POWER PLANT, UNIT 3

EXECUTIVE SUMMARY

In 1984 PG&E submitted the Humboldt Bay Power Plant, Unit 3 (HBPP) SAFSTOR Decommissioning Plan (SDP) in support of the application to amend the HBPP Operating License to a Possession-Only License. As a result of the 1996 NRC decommissioning rule, the SDP was considered to be a Post-Shutdown Activities Report (PSDAR) because it contained information related to decommissioning activities. It was also considered to be a Final Safety Analysis Report (FSAR) because it contained information such as plant description, site characterization and accident analysis.

In compliance with the 1996 NRC decommissioning rule, PG&E submitted a PSDAR in February 1998 to provide a general overview of proposed decommissioning activities. As a result, the SDP will focus on providing the type of information contained in an FSAR and will contain less information related to decommissioning activities. Thus, the SDP has been more appropriately renamed the Defueled Safety Analysis Report (DSAR).

The 1996 NRC decommissioning rule became effective August 28,1996. This rule modified 10 CFR 50.71 to require licensees of permanently defueled plants to revise their FSARs at least every 24 months. To comply with the decommissioning rule, PG&E submitted a revised DSAR on August 28,1998, and submitted DSAR revisions at least every 24 months thereafter.

License Amendment 34, issued November 18, 2002, relocated a great deal of information from the Unit 3 Technical Specifications to the DSAR. In addition, numerous other DSAR changes have been made for various reasons, plus, reformatting for enhancement purposes has been performed. Because of these changes, HBPP has decided to submit Revision 5 of the DSAR prior to the required submittal date of August 23, 2004. Future DSAR revisions will be submitted at least every 24 months from the date of Revision 5.

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1.0 INTRODUCTION

In 1984 PG&E submitted the Humboldt Bay Power Plant, Unit 3 (HBPP) SAFSTOR Decommissioning Plan (SDP) in support of the application to amend the HBPP Operating License to a Possession-Only License. As a result of the 1996 NRC decommissioning rule, the SDP was considered to be a Post-Shutdown Decommissioning Activities Report (PSDAR) because it contained information related to decommissioning activities. It was also considered to be a Final Safety Analysis Report (FSAR) because it contained information such as plant description, site characterization and accident analysis.

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In addition to the DSAR and PSDAR, PG&E has submitted other documents to the NRC in accordance with 10 CFR 50 that constitute the licensing basis for HBPP. These other documents include: (1) License Amendment Application, (2) revised Technical Specifications, (3) Environmental Report, (4) Quality Assurance Plan, (5) Security Plan, (6) Emergency Plan.

1.1 DEFUELED SAFETY ANALYSIS REPORT

This DSAR (formerly known as the SDP) was originally prepared in support of PG&E's application to amend the Unit 3 operating license to a possession-only license. The unit was placed in a state of custodial SAFSTOR for up to 30 years, after which it is planned to dismantle the unit, remove all radioactive material from the site, and terminate the license in accordance with NRC requirements. More specific information pertaining to future decommissioning activities is contained in the PSDAR.

Section 1.0 of this plan includes an introduction to the DSAR, criteria and guidelines review, a summary of the licensing and operating history of the plant, and a site description. This section also describes the activities that were performed to establish the custodial SAFSTOR mode and the conditions that will exist during the SAFSTOR period.

Section 2.0 contains a description of the facility; including Unit 3 plant structures, plant systems, and systems common to Units 1,2, and 3. Spent fuel storage and inspection of spent nuclear fuel are discussed in a subsection of Spent Fuel Pool and Associated Systems, contained in the Plant Systems Description section.

Section 3.0 is the Radiation Protection section. This section includes a radiological characterization of the facility, monitoring and surveillance programs, radioactive waste processing and disposal, and a health physics section. The characterization describes the conditions present at the time of commencing decommissioning activities.

Certain activities such as unloading of the reactor core, processing and disposal of radioactive wastes, and decontamination of some structures, systems, and components performed in accordance with the operating license before the commencement of SAFSTOR activities, are reflected in this section. The Health Physics section includes discussions of the ALARA and Radiation Protection Programs.

Section 4.0 contains a description of the plant organization, administration, and control. Additionally, the "Industrial Health and Safety Program" is included in this section.

Section 5.0 describes plant operating and surveillance requirements, and includes a description of the fire protection program.

Appendix A contains a safety and accident analysis for Unit 3 during the SAFSTOR period.

Appendix B is a description of spent fuel heatup following loss of storage pool water.

Appendix C is a criticality analysis for SAFSTOR decommissioning.

Appendix D is the SAFSTOR baseline radiation study.

Appendix E contains the radionuclide inventory conducted in 1984.

Appendix F is a description of the certified fuel handler training and certification program used during the SAFSTOR period.

Appendix G contains descriptions of the systems and components that are abandoned in place, or that have been removed from service completely.

Appendix H contains figures developed at the time Unit 3 entered SAFSTOR. The figures are provided for general information purposes.

1.2 CRITERIA AND GUIDELINES REVIEW

In preparation of this DSAR, PG&E complied with the guidance provided in Draft Regulatory Guide DG-1067, "Decommissioning of Nuclear Power Reactors," dated July 1997.

In preparation of the original SDP, government regulations, regulatory guidelines, and technical reports were reviewed to determine their applicability to HBPP. Other decommissioning plans, reports, and relevant facility experiences were also reviewed. The following list identifies the regulations and related documents that were reviewed.

<u>CODE OF FEDERAL REGULATIONS</u>

<u>TITLE 10 - ENERGY</u>

Part 20 -	Standards for Protection Against Radiation
Part 30 -	Rules of General Applicability to Domestic Licensing of Byproduct Material

Part 40 - Domestic Licensing of Source Material
Part 50 - Domestic Licensing of Production and Utilization Facilities
Part 51 - Licensing and Regulatory Policy and Procedures for Environmental Protection
Part 61 - Licensing Requirements for Land Disposal of Radioactive Waste
Part 70 - Domestic Licensing of Special Nuclear Material
Part 71 - Packaging of Radioactive Material for Transport and Transportation of Radioactive Material
Part 73 - Physical Protection of Plants and Materials

TITLE 29 - LABOR
Part 1910 - Occupational Safety and Health

TITLE 40 - PROTECTION OF ENVIRONMENT
Part 61 - National Emission Standard for Hazardous Air Pollutant
Part 122 - National Pollutant Discharge Elimination System
Part 141 - National Interim Primary Drinking Water Regulations
Part 190 - Radiation Protection Standards for Nuclear Power Operations

TITLE 49 - TRANSPORTATION
Parts 171-179 - Hazardous Materials Regulations

REGULATORY GUIDELINES

Regulatory Guide 1.86 - Termination of Operating Licenses for Nuclear Reactors

TECHNICAL REPORTS

(1) Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, June 1980

(2) Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672 Addendum 1, July 1983

(3) Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, NUREG-0586, January 1981

(4) Final Environmental Impact Statement - Decommissioning of the Shippingport Atomic Power Station, DOE/EIS-0080, May 1982

(5) An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives, AIF/NESP 009, November 1976

DECOMMISSIONING PLANS AND EXPERIENCE

(1) Dismantling Plan - Plum Brook Reactor Facility, February 1980

(2) Decommissioning Plan and Safety Analysis Report - Peach Bottom

Atomic Power Station Unit 1, May 1975

(3) Decommissioning Peach Bottom Unit 1 Final Report, July 1978
(4) Decommissioning Plan for Indian Point Unit 1 (Proposed), October 1980
(5) Discussions with General Electric Company regarding decommissioning experience with the Vallecitos Boiling Water Reactor, ESADA Vallecitos Experimental Superheat Reactor, and General Electric Test Reactor.

1.3 HUMBOLDT BAY POWER PLANT - UNIT 3 OPERATING HISTORY

Humboldt Bay Power Plant, Unit 3 was a natural circulation boiling water reactor and associated turbine-generator operated by Pacific Gas and Electric Company (PG&E). In addition to Unit 3, the Humboldt Bay Power Plant consists of two oil and/or natural gas fueled units (Unit 1 rated at 52 MWe and Unit 2 rated at 53 MWe). Two diesel-fueled gas turbine Mobile Emergency Power Plants (MEPPs), each rated at 15 MWe, are also currently located at the plant, but may be relocated to other sites either temporarily or permanently.

1.3.1 INITIAL CONSTRUCTION AND LICENSING HISTORY

Unit 3 was granted a construction permit by the Atomic Energy Commission (AEC) on October 17, 1960, and construction began in November 1960. The AEC issued Provisional Operating License No. DPR-7 for Unit 3 in August 1962. Unit 3 achieved initial criticality on February 16, 1963, and began commercial operation in August 1963.

To simplify plant design, Unit 3 included certain features that were not typical of nuclear plants of that era. Natural circulation within the reactor vessel eliminated the need for recirculation pumps, a direct cycle design eliminated the need for heat transfer loops between the reactor and turbine-generator, and as a joint effort between PG&E and General Electric Company, the pressure suppression containment system was developed to eliminate the need for the large containment structures that had been used at earlier nuclear plants. The pressure suppression containment design permitted the reactor to be located below ground level.

On July 2, 1976, Unit 3 was shut down for annual refueling and to conduct seismic modifications. Seismic and geologic studies were in progress. In December 1980 it became apparent that the cost of completing required backfits might have made it uneconomical to restart the unit. Work was suspended at that time awaiting further guidance regarding backfitting requirements. In 1983, updated economic analyses indicated that restarting Unit 3 would probably not be economical, and in June 1983 PG&E announced its intention to decommission the unit.

1.3.2 OPERATING EVENTS WHICH AFFECT DECOMMISSIONING

During the operation of Unit 3, certain events occurred that affected plant conditions and have to be considered during SAFSTOR and decommissioning. The following section describes these events and how they relate to SAFSTOR and the decommissioning effort. None of these events caused conditions that would prevent Unit 3 from being decommissioned with current technologies and work practices.

1.3.2.1 Fuel Cladding Failures

When Unit 3 began operation, the fuel utilized stainless steel cladding. In 1964 and 1965, fuel cladding failures began to occur and it was determined that the cause of the failures was stress corrosion cracking of the stainless steel cladding. In 1965, the stainless steel-clad fuel was replaced with zircaloy-clad fuel.

The early fuel cladding failures resulted in contamination of the reactor vessel, spent fuel storage pool, and plant systems with fission products and transuranic nuclides. All stainless steel-clad fuel was shipped offsite for reprocessing during the years 1969 through 1971.

1.3.2.2 Spent Fuel Pool Leakage

In March 1966, it was discovered that a leak in the spent fuel storage pool liner had developed. Operating procedures were developed to minimize leakage and investigations were conducted to determine the magnitude of any groundwater contamination that could have occurred. Samples of groundwater from the plant wells, the reactor caisson sump, and two of three test wells did not reveal signs of contamination. One test well drilled north of the spent fuel storage pool (between the pool and the bay) revealed evidence of contamination, but the levels were a factor of 100 below allowable drinking water limits. The test wells have been monitored regularly since that time and results of the surveillance have indicated no increase in activity.

1.3.2.3 Spills of Contaminated Water

On several occasions during the operation of Unit 3, radioactively contaminated liquids were spilled in certain areas of the facility. Since access to most areas of Unit 3 is controlled for purposes of contamination and radiation exposure control, the corrective action was to clean up the spill and either decontaminate the area or fix the contamination so that exposures required either for decontamination or resulting from the contamination would be consistent with ALARA considerations. During the SAFSTOR period, any residual contamination resulting from these spills will continue to be contained. Final decontamination of these areas to levels acceptable for unrestricted use will be accomplished as part of the final dismantlement program.

1.3.2.4 Dropped Fuel Assembly

In 1975, a fuel assembly was dropped into the spent fuel pool cask loading pit, and several fuel rods separated from the assembly. A special container was fabricated to contain the assembly. The assembly and the loose rods have been retrieved and stored in the container in the spent fuel storage pool fuel storage racks.

1.3.3 OPERATING RECORD

During the period August 1963 to July 1976, Unit 3 generated over 4.7 billion kilowatt-hours of electricity and had a cumulative availability factor of 85.9 percent.

1.4 SITE DESCRIPTION

Humboldt Bay Power Plant is located about four miles true southwest of the city of Eureka, Humboldt County, California, and consists of 142.9 acres of land. A physical description of the plant is detailed in the following sections. Section 2 of the DSAR contains the facility description.

1.4.1 Topography

Terrain of the site varies from submerged and low tidal land, protected by dikes and tide gates, to a high precipitous bluff along the southwestern boundary. Elevations range from approximately -3 feet to +75 feet based on a datum of the mean lower low water (MLLW) level. The ground floor of the refueling building is at elevation +12 feet.

1.4.2 Soils and Geology

HBPP lies in the Northern California Coast Ranges geomorphic province. This province consists of a system of longitudinal mountain ranges (2000 to 4000 foot elevations with occasional 6000 feet peaks) and valleys with a trend of N 30 degrees to 40 degrees W.

The immediate vicinity of the site consists of sand and alluvial soil and strata of the Hookton and Carlotta sedimentary formations. These formations are primarily consolidated sands, gravels, and clays and conglomerates with good engineering properties. HBPP buildings have their foundations in these strata.

The principal rocks in the area range in age from late Jurassic to early Upper Cretaceous. These rocks are in two groups:

Clastic sedimentary rocks, consisting of sandstone, mudstone, and conglomerate

Volcanic and associated rocks, consisting of greenstone, basalt, chert, and minor amounts of limestone

In the site area, younger rocks overlie the volcanic strata. These rocks are dominantly marine sandstone, mudstone, and conglomerates ranging in age from the late Cretaceous to early Pleistocene. Recent alluvium forms the shallow strata in the valleys and in areas along the coast.

1.4.3 Hydrology

1.4.3.1 Surface Hydrology

The surface runoff from the site is directed into drains discharging into the plant cooling water intake canal, through the plant, and into Humboldt Bay via the discharge canal. Outside the area served by the plant drain system, surface runoff drains into Buhne Slough, the natural drainage for the area, which drains into Humboldt Bay.

The nearest streams to the site are Salmon Creek and Elk River, which are within a mile south and north of the site, respectively, and which discharge into Humboldt Bay. These streams are used for watering livestock, but are not used as a potable water supply.

The Mad River flows west approximately 13 -15 miles northeast of the site. The Ruth reservoir, the source of the city's water supply, is located on this river.

To the south, the Eel River discharges to the Pacific Ocean 8-10 miles from HBPP. This river is not used for potable water within 25 miles of HBPP.

1.4.3.2 Groundwater Hydrology

Groundwater supplies all domestic, industrial, and agricultural needs in Humboldt County except that which is supplied by the Ruth reservoir. A groundwater study made in the area of HBPP prior to Unit 3 construction (Morliave, 1960) identified the following important features of the groundwater system:

Movement of all groundwater is generally toward the bay.

Vertical rates of groundwater movement in the area of the plant are a few inches per day in the light surface alluvium.

Horizontal movement in aquifers beneath the site ranges from several feet to hundreds of feet per day.

Groundwater elevation in the area near the bay is similar to sea level and may be somewhat affected by tidal action. This elevation is approximately 12 feet below the plant floor elevation.

Both a groundwater and slight topographic divide appear to exist between HBPP and Elk River. These features reduce the probability of liquid discharges or leakage from the plant site to this stream either by surface or groundwater flow.

Southwest of the plant, an area exists which has slight landward groundwater gradients under some conditions. However, this area lies within an area that is affected by tidal action. Negligible inland flow is estimated to occur.

Any migration of materials of plant origin into the soils beneath or near the plant would move vertically quite slowly until reaching the saturation zone. Migration would then be horizontal, toward the bay.

1.4.3.3 Humboldt Bay

Humboldt Bay is a tidal bay receiving and discharging ocean water through its inlet. Very little fresh water discharges into this bay.

A study of tidal hydrology in Humboldt Bay has been made (Hazards Assessment Report, 1960). The purpose of this study was to determine the flow pattern of tidal currents in Humboldt Bay, dilution of the effluent from the plant, and the flushing action of the tides by movement in and out of the bay. The study concluded that the discharge of effluents into Humboldt Bay would result in a gradual dilution as they moved into the bay. Dilution of effluents along the shore of the bay entrance is high because of the relatively drastic changes in depth for each tidal cycle. The swift moving water in the deeper channels leading from the North Bay and South Bay causes rapid dilution. The ebb tides carry most of the discharged water out to sea and bring in water from the sea on the following tide.

The finished grade elevation for the plant was established at +12.0 feet to be above the U.S. Coast and Geodetic Survey estimate of the highest high tide of +9.5 feet.

1.4.4 Seismology

There have been numerous geology and seismology studies conducted for the site with respect to the effects of potential seismic events in the area. These studies are analyzed in Appendix 10.3 to the Environmental Report.

1.4.5 Climatology and Meteorology

The climate at HBPP is mesic oceanic, characteristic of the northwestern coast of the continental United States. The area has two distinct seasons differentiated by precipitation rather than temperature. The wet season extends roughly from November through March and yields approximately 75 percent of the average annual precipitation. The dry season, extending from May through September, contributes only 10 percent of the average annual precipitation. The transitional months, April and October, contribute the balance. The mean annual precipitation is 39 inches.

The range of air temperatures is minimal, averaging 52°F annually, 46°F in winter and 56°F in summer.

The prevailing wind direction is from the north. The wind distribution is 24.3 percent offshore, 57 percent onshore, and 18.7 percent light and variable. Average wind speeds are strongest for the north winds (16 mph) and the southeast winds (12.5 mph) during the wet seasons. These are lower during the dry season. During the rainy seasons, the wind from the south-southwest dominates slightly.

Prevailing winds can be expected to carry airborne effluents from the plant south and inland 55 percent of the time. Approximately 20 percent of the effluents would be distributed across the bay entrance to the ocean.

Approximately 25 percent of the effluents would be discharged into calm air and distributed randomly.

1.5 SAFSTOR ACTIVITES

1.5.1 OBJECTIVES

The objectives of SAFSTOR activities are:

- To secure non-operating plant systems to prevent deterioration and minimize potential for release of contained radioactivity.
- To process and dispose of radioactive wastes generated during SAFSTOR.
- To decontaminate plant facilities to the maximum extent practical, to minimize the potential for spread of contamination outside of the facility, and to minimize the requirements for periodic surveys.
- To reduce general area radiation levels in the vicinity of equipment operated or maintained during the SAFSTOR period to as low as reasonably achievable (ALARA).
- To maintain plant facilities to support long-term storage of spent fuel, to minimize generation of radioactive wastes, and to minimize necessary maintenance and surveillance during SAFSTOR.
- To establish baseline conditions and a monitoring and surveillance program for the SAFSTOR period.

1.5.2 ORGANIZATION

Organization, Administration and Control are discussed in detail in Section 4 of the DSAR.

1.5.3 DECOMMISSIONING ACTIVITIES

The following sections describe the major activities that have been performed as part of the project to place Unit 3 into the custodial SAFSTOR mode. For those activities for which specific tasks can be identified, a general description of those tasks is included. Detailed descriptions of specific tasks were defined at the procedural level at the time the task was performed.

1.5.3.1 Preparations for SAFSTOR

Systems and equipment not required by the Unit 3 Operating License for the Cold Shutdown Mode and not required to support SAFSTOR activities were secured in preparation for SAFSTOR. Preparations included unloading the reactor core; draining, flushing, and securing systems; deenergizing instruments and controls which are no longer required; and isolating non-operational systems from systems still in operation.

Securing and lay-up of these systems and components were performed in accordance with procedures reviewed by the Plant Staff Review Committee (PSRC) and approved by the Plant Manager. The reviews included a determination that the system and equipment lay-ups did not involve an unreviewed safety question as defined in 10 CFR 50.59(c) or a change to the Unit 3 Technical Specifications.

As a result of ALARA considerations during the performance of system lay-ups and decontamination work, certain piping sections or components were removed. For systems that are to remain secured for the SAFSTOR period, the piping and equipment was not replaced. Open pipes were sealed to prevent the spread of contamination.

Also during the preparations for SAFSTOR decommissioning, some radioactive wastes on-site were processed and shipped to licensed disposal facilities. These wastes were primarily radioactive wastes that had been generated during the years that Unit 3 had operated and had been stored on-site awaiting final disposal. Additionally, liquid wastes generated as a result of draining and flushing plant systems were processed by the radioactive waste treatment system.

1.5.3.2 System Layup

In preparation for SAFSTOR, systems no longer required by the Unit 3 Technical Specifications were secured and isolated. In addition, systems that were required to support entering SAFSTOR, but not required during SAFSTOR, were secured when they were no longer needed. The objectives of the system lay-up are as follows:

- Systems containing fluids were drained to the maximum extent practical.
- Significant sources of radiation in areas that will be routinely accessible during SAFSTOR were either removed or shielded.

- Connections between secured systems and operating systems were sealed by either using blank flanges or by cutting and capping the lines. This prevents leakage from an operating system from refilling a system that has been drained.

- Motors, valves, instrumentation, and other electrical components associated with secured systems were deenergized.

1.5.3.3 Operational Systems

A number of systems and components will remain operational over the SAFSTOR period. They are required (1) to support storage of spent reactor fuel, (2) to maintain environmental conditions for personnel protection, (3) to provide remote monitoring and alarms, and (4) to collect and process liquid or solid radioactive wastes. System descriptions and operating procedures were revised, where necessary, to reflect the SAFSTOR status and operational requirements.

The physical SAFSTOR status of the systems and components of Unit 3 are detailed in Sections 2.3 and Appendix G.

1.5.4 CONTINUED CARE PLAN, SAFSTOR TO DECON

During the SAFSTOR period and until initiation of the final DECON decommissioning program, necessary plant systems will be operated and maintenance will be performed on structures, systems, and components as required to ensure continued safe conditions within the plant.

1.5.4.1 Operation of Plant Systems

Other sections of this DSAR identify those systems required to operate during the SAFSTOR period. The following system classifications are included:

- Service Systems, including the Refueling Building Ventilation System, the Spent Fuel Pool Service System, the Fire Protection System, and Electrical Systems

- Waste Disposal System, including the Liquid Radioactive Waste System and the Solid Radioactive Waste System

- Monitoring System, including the Stack Gas Radiation Monitoring System, the Process Water Monitor, Area Monitors and Portable Monitoring Equipment, Offsite Environmental Monitoring Stations, and Spent Fuel Storage Pool Water Level Monitors.

These systems will be operated as required during the SAFSTOR period. Their operation will be in accordance with approved procedures, and the operational schedule will be of sufficient regularity to ensure adherence to the Unit 3 Technical Specifications and other

licensing basis documents such as the ODCM. This operational schedule may vary over the SAFSTOR period as conditions warrant.

1.5.4.2 Maintenance of Structures, Systems, and Components

The maintenance program established for SAFSTOR is a modified continuation of the previous maintenance program at the plant and includes aspects for both preventive and corrective maintenance.

The preventive maintenance aspect provides for a regularly scheduled series of inspections, tests, and services for structures, systems, and components. The frequency of preventive maintenance was established on the basis of prior experience, ongoing operational use, plant conditions, and where applicable, requirements in the Technical Specifications and other licensing basis documents such as the DSAR and the ODCM . The objective of the preventive maintenance program is to ensure continued reliable function of necessary structures, systems, and components equivalent to or better than the reliability existing at the beginning of the SAFSTOR period.

The corrective maintenance aspect of the program provides for analysis and appropriate action to be taken for all conditions where function of structures, systems, and components is determined to be degraded. Adequate stores of spare parts and servicing equipment are maintained to ensure that corrective action can be taken in a timely manner and where required within the timeframe required by the Technical Specifications and other licensing basis documents such as the DSAR and the ODCM. Plant maintenance will implement the requirements of 10 CFR 50.65 as appropriate (Maintenance Rule).

1.5.4.3 Dismantlement of Structures, Systems, and Components (SSCs) During SAFSTOR

During the period of SAFSTOR, it may be necessary to remove SSCs for a variety of reasons including removal of interference, accessibility, and ALARA considerations. In addition, minor decommissioning or salvaging of equipment for used outside of Unit 3 may be authorized. The evaluation as to future need, reclassification of the status of, and subsequent removal of the SSC occurs in two phases:

Phase I: Evaluation (Administrative Phase)

- Prior to authorizing the permanent removal of SSCs from Unit 3, a determination shall be made that the SSC will not be required during SAFSTOR of during the decommissioning (DECON) period.
- The determination described above should be included in a DSAR change request as justification for the reclassification.
- Submission of a request to change the DSAR, including the Licensing Basis Impact Evaluation (LBIE), to reclassify the SSC as "Abandoned".

- Upon approval of the DSAR change request, update the DSAR to reflect the "Abandoned" status.

Phase II: Removal (Physical Phase)

- When the DSAR condition for SAFSTOR (as described in Table 2-1) is "Abandoned", the SSC may be removed as described below:

  - Prepare and approve the appropriate design documents, procedures, plant problem reports, etc., for the removal of the SSC.

  - Remove the SSC.

  - Update the appropriate plant drawings and design documents.

  - Submit a DSAR Change Request to update the condition for SAFSTOR in DSAR Table 2-1 to "Removed".

2.0 FACILITY DESCRIPTION

HBPP is comprised of two fossil-fueled units (Unit 1 - 52 MWe and Unit 2 - 53 MWe), a single nuclear unit (Unit 3 - 63 MWe), and two gas turbine-powered mobile emergency power plants (MEPP No. 2 and 3 - 15 MWe each). Necessary support structures, equipment, and tanks are also located on the plant site. A site plan is shown in Appendix H, Figure 2-1.

The principal activities of the Plant are those related to the generation and transmission of electric power and the associated service activities. Activities associated with Unit 3 consist of storage and surveillance of spent fuel, monitoring and surveillance of the decommissioned facility, and operations and maintenance to support the above-mentioned activities in accordance with Regulatory Guide 1.86, Section C.5. The following description is specific to Unit 3 unless otherwise stated.

2.1 GENERAL PLANT DESCRIPTION

Unit 3 consists of a General Electric natural circulation, single cycle boiling water reactor, the associated turbine-generator, and necessary support and auxiliary systems. The reactor vessel is a 10 feet diameter, 40.5 feet long pressure vessel that is suspended in a drywell containment vessel.

The reactor primary containment is located entirely below grade and is comprised of the drywell vessel, which houses the reactor, and a suppression chamber located concentrically around the drywell. The drywell and suppression chamber are located inside a reinforced concrete caisson which, in the vicinity of the reactor, is approximately 60 feet in diameter and extends to an inside depth of 78 feet below grade. A caisson access shaft extends from the top of the caisson to the space beneath the drywell. The access shaft contains the reactor auxiliary systems.

The refueling building encloses the space above the caisson. In addition to the reactor caisson, the refueling building contains the spent fuel storage pool and the new fuel storage vault.

Adjacent to the refueling building is the power building and turbine pedestal. The power building contains the condenser, feedwater and condensate systems, and the steam cycle auxiliary systems. The control room is also contained in this building. The turbine-generator is located on the turbine pedestal. The turbine is contained in a shielded enclosure while the generator and associated exciter are located outside.

A hot machine shop/calibration facility is located southeast of the power building. The machine shop is used for maintenance of radioactively contaminated equipment. The calibration facility contains sources used to calibrate radiation survey instruments.

Liquid wastes are processed in the radwaste treatment building, located in an excavated portion of an earthen embankment north of the refueling building. This building contains two concentrated liquid radioactive waste storage tanks and the resin disposal tank and is

surrounded on three sides by earth. A steel building encloses the entire liquid radwaste treatment area.

North of the radwaste building are three high-level solid radioactive waste storage vaults, a low-level waste storage building, and a low-level waste handling building.

2.2 PLANT STRUCTURES

The plant structures are shown in Appendix H, Figures 2-2 through 2-18, which provide details of plant layout and equipment locations. The figures in the appendix are provided for general information purposes only. Plant drawings reflecting current plant conditions are maintained by the Engineering Department.

2.2.1 FOUNDATIONS

The power building and turbine pedestal are supported on a 3 feet 6 inches thick continuous concrete mat foundation resting on a grid of 30-ton timber pilings penetrating to the sand strata at elevation -24 feet.

The number of pilings along the north edge of the mat was increased to carry the concentrated edge load of the power building.

The refueling building is supported on the reactor caisson by six 100-ton H-piles driven into the same strata that supports the caisson. The reactor caisson also serves as the former ventilation stack foundation by providing the projecting stack support bracket. The few equipment foundations and column foundations not resting on the power building structure or mat foundation are supported on spread footings below grade. The radwaste treatment building, hot machine shop/calibration facility, condensate tank, and radwaste storage and handling buildings are supported on grade slabs or footings.

2.2.2 POWER BUILDING

The power building is a monolithic, reinforced concrete, two-story structure which houses and provides radiation shielding as required for the main condenser and its auxiliary equipment, the makeup demineralizer system, the condensate demineralizer system, reactor feed pumps, access control, monitoring, and laundry facilities.

The control room is located on the +27 foot level and is enclosed by the refueling building, the turbine shield wall, and a structural steel framing and roof system. An extension north of the control room housing the instrument shop is of concrete block construction with a concrete roof.

2.2.3 TURBINE PEDESTAL AND SHIELDING

The turbine-generator is supported at the +27 foot level by a massive concrete pedestal

composed of longitudinal and transverse rigid frame bents and shielding walls bearing on the foundation mat. The pedestal is separated from the power building by a joint for vibration isolation.

The turbine is enclosed by massive precast concrete shield walls and roof elements, which are sized for removal by the powerhouse crane to permit turbine overhauls.

2.2.4 REFUELING BUILDING

The refueling building is a reinforced concrete structure located immediately above the reactor caisson. The refueling building is 43 feet x 103 feet x 35 feet high, constructed of reinforced concrete walls with a composite roof of precast, prestressed concrete double tee sections and concrete topping. The 12-inch minimum thickness walls and roof were designed to function as a weather enclosure, contamination control barrier, and radiation shield.

The building encloses the refueling area consisting of the spent fuel storage pool, new fuel storage vault, drywell vessel opening, and shipping cask railroad spur. The deactivated spur enters the building through pneumatically operated 10 ton concrete shielding doors, which, when open, provide a 13 feet 11 inches wide x 13 feet 6 inches high clear opening. The building wall pilasters support the structural steel girders for the 75-ton bridge crane required for handling the reactor shield plug and the spent fuel transfer cask, and other heavy loads in the refueling building. The main floor of the refueling building is at elevation +12 feet. Personnel access to the ground floor is through airlocks entering from the +27 and +12 foot levels. Access to the reactor caisson is through the refueling building. Refueling building penetrations for personnel and equipment entry and for ventilation systems are described in Table 2-3.

2.2.5 REACTOR CAISSON

The reactor caisson consists of a reinforced concrete structure, 59 feet 6 inches in diameter, 78 feet 0 inches inside depth, which houses the reactor vessel and auxiliary equipment, the drywell vessel, suppression chamber, and the rectangular portion described below containing the spent fuel storage pool and new fuel storage vault.

The drywell vessel is centrally located in the caisson and serves as the primary containment vessel. Surrounding the drywell vessel is the 300-degree annular suppression chamber (12 feet 6 inches wide by 49 feet 0 inches high with a 4 feet 0 inches exterior wall and a 3 feet 10 inches interior wall).

The suppression chamber is constructed of reinforced concrete and lined with carbon steel plate. Six 40-inches-diameter vent pipes connect the drywell to a common 40-inches equalizing ring header at the top of the suppression chamber. From the header bottom, 46 evenly spaced 14 inches pipes extend to a point 6 feet below the normal operating water level. There were baffle plates between the vent pipes and deflector plates in front of the six 40 inches entrance vents. The baffle plates were removed during SAFSTOR. An

access was created at the -66 foot elevation into the West chamber and between the East and West chambers to facilitate the removal of the baffle plates.

Collectively, the drywell, the suppression chamber, and the vent piping comprised a pressure suppression containment system that surrounded the reactor during plant operations.

A 60-degree portion of the circular caisson below elevation -14 feet serves as the access shaft and houses principally the control rod hydraulic system and instrument vaults. Access to the bottom of the drywell and the various levels in the access shaft is provided by a vertical manlift, 2-ton jib hoist, and emergency ladder. In addition, a sealed vertical escape shaft with four access doors in the access shaft provides emergency escape to outside the refueling building.

The caisson above elevation -14 feet and up to grade at elevation +12 feet is rectangular (49 feet 0 inches wide x 75 feet 0 inches long) and serves as the structural foundation for the refueling building and former stack projection. The drywell vessel and biological shield continue up to grade and serve as a central support pier for the floors at elevations -14 feet, -2 feet, and + 12 feet.

A spent fuel storage pool, integral to the reactor caisson, is provided for (1) storage of spent fuel assemblies; (2) removal and inspection of fuel assembly channels; and (3) underwater loading of the spent fuel shipping cask.

A new fuel storage vault, formerly used for storage of new fuel, is located adjacent to the spent fuel storage pool. The vault measures 16 feet long, 12 feet 6 inches wide, and 12 feet deep. Contained in the vault are the fuel storage pool coolers and access hatches to the turbine building drain tank vault below. The top of the new fuel storage vault is at the refueling building floor level, and the vault is normally covered by ¼-inch thick checkered steel plates that are hinged for access to the vault.

The base of the caisson was sealed with tremie concrete at elevation -66 feet underneath the drywell and suppression chamber, and at elevation -14 feet and -24 feet under the spent fuel pool. A 6-inch pervious gravel blanket on the tremie and below the 6-inch floor slab was designed to collect seepage and to prevent any pressure buildup.

2.2.6 VENTILATION STACK

The 50-foot high, 48-inch diameter ventilation stack is made of carbon steel and is the single point discharge of plant effluents. The 50-foot ventilation stack replaced the preexisting 250-foot ventilation stack that was removed down to the 40-foot elevation. The base of the preexisting 250-foot stack still serves as the enclosure for three floors of gas treatment equipment.

2.2.7 RADWASTE TREATMENT FACILITIES

The radwaste treatment building is recessed into the hill north of the refueling building. It consists of a 37 x 96 foot slab at grade with a rear retaining wall, wing walls, tank and equipment vaults, and an enclosed control room. All walls and roof slabs are of monolithic reinforced concrete. The slab at grade provides support for eight liquid waste tanks; five are not vaulted but within the LRW enclosure, and the other three are housed in shielded vaults. The solid waste vault is an underground reinforced concrete vault with a capacity of 1,200 cubic feet. The vault is located on top of an earth bank directly north of the radwaste treatment building. The top of the vault is at ground level. The interior dimensions are 20 x 8.5 x 8 feet deep. Two interior walls are provided that divide the vault into three equal compartments. Three reinforced concrete roof slabs are designed to overlap and interlock with the walls to prevent entry of rainwater.

North of the solid waste storage vaults is the low-level waste storage building. The building is of concrete block construction and is divided into two sections, one for storage of low-level solid radioactive waste awaiting disposal and the other for storage of contaminated reusable tools and equipment.

North of the low-level waste storage building is the low-level solid waste handling building. The handling building is a prefabricated metal building that consists of a 30 x 40 foot waste handling area and a 30 x 50 foot covered truck loading area. The building provides weather-protected storage for empty radioactive waste packages (drums and boxes) and packages awaiting shipment.

2.2.8 YARD STRUCTURES

The yard structures consist of the hot machine shop/calibration facility and the off-gas treatment vault.

The hot machine shop/calibration facility is constructed of reinforced concrete block masonry bearing walls, structural steel framing, and corrugated roof decking. The ground floor is a reinforced concrete slab on 6 inches of rock. The roof decking is welded to the structural steel framing to provide diaphragm action for resisting lateral loads.

The condenser off-gas treatment vault was constructed in 1976 to contain a new condenser off-gas treatment system. The vault is a monolithic, reinforced concrete structure. After construction, neither the building nor the contained equipment was used since the plant was not returned to power operation. The condenser off-gas treatment system was removed from the vault in 2002.

2.2.9 INTAKE AND DISCHARGE STRUCTURES

The intake structure is a compartmented, reinforced concrete box with overall dimensions 53 feet long x 22 feet wide x 24 feet deep, similar to adjacent Units 1 and 2 structures. The top slab of the structure is at elevation 12 feet 0 inches.

The intake structure is separated by two cross walls, one between the bar racks and traveling screens and the other between the screens and the pumps. These cross walls have gated openings. The structure is designed to be stable against uplift at high tide with the gates closed and the compartments dewatered. The walls and floor design also provide for resistance to the pressures existing under these extreme conditions.

A precast concrete deck is provided between Units 2 and 3 intake structures for access with a 25-ton capacity truck crane for maintenance of the bar racks and traveling screens. The Unit 3 screen wash troughs extend across this deck and discharge into the Unit 2 sump pit.

The discharge structure is a gated reinforced concrete box located at the upstream end of the discharge canal and forms the end anchor of the cooling water discharge line. The structure is open at the top, providing access to the pipe when the gate is closed.

Both the intake structure and discharge structure were provided with embedded sheetpile sections to allow future expansion immediately adjacent to the structures.

2.2.10 SEISMIC UPGRADING

Structures at Unit 3 were upgraded to provide additional protection against earthquake damage. These modifications primarily consisted of the addition of structural steel and piping supports.

2.2.11 ONSITE COMBUSTIBLE FUEL STORAGE

The description of combustible fuel storage facilities at the HBPP is given in Table 2-2.

2.3 PLANT SYSTEMS DESCRIPTION

For this DSAR, the plant systems are addressed in eight major system groupings that are functionally oriented. These major groupings are:

- Spent Fuel Pool and Associated Systems
- Waste Disposal Systems
- Service Systems
- Electrical Systems
- Radiation Monitoring Systems
- Instrumentation and Control (I & C) Systems
- Nuclear Steam Supply System
- Turbine Plant Systems

The operational systems and major components comprising each of these major system

groupings are described in the sections below. The systems and components that have been abandoned in place, or removed altogether, are described in detail in Appendix G. The systems, components and their physical SAFSTOR status are summarized in Table 2-1.

2.3.1 SPENT FUEL STORAGE POOL AND ASSOCIATED SYSTEMS

A spent fuel storage pool, integral to the reactor caisson, is provided for (1) storage of spent fuel assemblies; (2) removal and inspection of fuel assembly channels; and (3) underwater loading of the spent fuel shipping cask. The water in the pool provides shielding and contamination control. The spent fuel storage pool is approximately 22 feet wide by 28 feet long. The pool depth is 26 feet deep except for the cask loading pit in the southeast corner, which is 36 feet deep. The pool is constructed of reinforced concrete and has a stainless steel liner. The stainless steel liner shall completely cover the inside surfaces of the spent fuel storage pool with a nominal gap of ¼ inch between the liner and the walls and the floor. A cover has been installed over the spent fuel storage pool to prevent objects from falling into the pool and to serve as a contamination control barrier.

A motor-operated movable service platform with a 500-pound hoist is provided for handling fuel assemblies in the pool.

Storage Racks. Storage racks are provided to contain spent fuel. A total of 89 rack assemblies are provided, and 44 of these racks can each contain eight fuel assemblies or two control rod assemblies. The remaining 45 racks can each contain three fuel assemblies. One location cannot be used because of a bolt protruding into the bottom of the location. This gives a total capacity of 486 fuel assemblies. Presently, there are 390 fuel assemblies stored in these racks.

Channel Stripping Machine. The channel stripping machine for removing irradiated channels from irradiated fuel assemblies is mounted in the pool. An irradiated channel inspection stand is installed on the stripping machine to allow underwater examination of an irradiated channel.

Spent Fuel Storage Pool Cover. A cover was installed over the spent fuel storage pool to minimize the spread of airborne contamination originating from the spent fuel storage pool and to provide protection against dropping of objects into the spent fuel storage pool. The cover includes the following characteristics:

- a structure designed to prevent dropping objects into the pool,
- a contamination control barrier, and
- the airspace between the pool level and the contamination control barrier has been tied into the ventilation system.

The cover is designed and constructed to be installed and removed in a manner, which prevents dropping the cover or its component parts into the pool. The cover design and the attachment design to the building structure are compatible with the current structure design of the pool.

Spent Fuel Storage Pool Service System. The spent fuel storage pool service system consists of the equipment necessary to maintain water level and quality in the pool as well as equipment needed for handling and movement of spent fuel. Major components include the pool liner gap pump, two fuel pool circulating water pumps, two fuel pool coolers, channel handling tools, miscellaneous fuel handling tools, a jib crane, a transfer cask and winch, an extension tank, and the service platform.

Makeup water for the spent fuel storage pool shall be provided from the demineralized water system. The capacity of the demineralized water tank shall be 5,000 gallons. Water to the demineralized water tank shall normally be supplied from the Units 1 and 2 condensate storage tanks. Emergency makeup water to the spent fuel storage pool shall be available from the plant fire system.

Spent fuel storage pool water quality shall be maintained by circulation of pool water through a demineralizer by either of two spent fuel pool circulating water pumps.

Water level in the gap between the spent fuel pool liner and the concrete wall shall be maintained by using a liner gap pump, which discharges to the liquid radioactive waste collection system.

Spent Fuel Storage Pool Water Level Monitors. Two water level indicating devices shall be installed in the spent fuel storage pool. The outputs of these monitors shall be indicated in the control room. Annunciation (visual and audible) of low water level shall be provided in the control room.

Spent Fuel Storage Pool Liner Gap Pump. This pump is located in a sump in the cask area at the bottom of the spent fuel storage pool. It takes suction on the gap between the fuel pool liner and the wall to maintain the water level below the groundwater level outside the building. Discharge is to the TBDT. The net effect is to maintain a head difference between groundwater outside the building and water in the liner, providing for preferential inflow leakage into the liner gap from outside. This minimizes leakage of radioactive contaminants to the outside of the building. The pump will be required throughout SAFSTOR.

Fuel Pool Circulating Water Pumps. Two pumps are located on the ground floor (elevation +12 feet) in the refueling building adjacent to the hatch into the new fuel storage vault. These pumps circulate water from the spent fuel storage pool through the spent fuel pool demineralizer and strainer. The fuel pool circulating water pumps will be required during the SAFSTOR period.

Fuel and Channel Handling Tools. The fuel and channel handling tools are stored until they are needed for final removal of the spent fuel from the spent fuel storage pool or for other fuel handling operations and training as required.

Spent Fuel Pool Jib Crane. This crane is used for movement of spent fuel within the spent fuel storage pool. It is mounted on the refueling platform and will be required for final transfer of spent fuel to the shipping cask.

Extension Tank. While not directly associated with the spent fuel storage pool, the extension tank is associated with refueling and is included in this section for accountability

purposes. The extension tank was installed on the reactor vessel flange after closure head removal. It was then filled with water for shielding purposes during spent fuel removal into the transfer cask. It will no longer be required and will remain in its stored position in the refueling building for the SAFSTOR period. Its use may be required during final DECON decommissioning of the plant at the end of the SAFSTOR period.

Transfer Cask and Winch. This cask and winch were used to transfer spent fuel from the reactor vessel to the spent fuel storage pool. It will be stored during SAFSTOR in the event that a shielded cask of this type is needed during final shipments of the spent fuel or during final DECON decommissioning of the plant.

Spent Fuel Pool Service Platform. The spent fuel pool service platform moves north and south over the spent fuel storage pool and is used along with the spent fuel pool jib crane to move spent fuel within the pool. It will be required for final transfer of spent fuel to the shipping cask. The spent fuel pool service platform and hoist will receive preventive maintenance and steps will be taken to minimize susceptibility to deterioration.

Any components related to the Spent Fuel Storage Pool not described in this section can be found in Appendix G.

2.3.1.1 Spent Fuel Storage

A total of 390 spent fuel assemblies will remain in the spent fuel storage pool until such time that they can be shipped offsite for final disposition. PG&E is not presently contemplating spent fuel pin consolidation prior to shipment to the Federal high-level waste repository. This spent fuel is a potential source of high radiation within the plant environs.

Fuel cladding failure might result from mechanical damage due to accidents or from corrosive deterioration. While occurrence of the above-noted adverse conditions is unlikely, PG&E has taken the following actions during SAFSTOR to further enhance the margin of safety for the spent fuel:

- A redundant level indicating system is provided to alarm low water level in the spent fuel storage pool.
- Water chemistry is maintained to minimize corrosive deterioration.
- Radioactivity levels in the pool water are monitored to provide an indication of fuel integrity.
- A cover has been installed over the spent fuel storage pool. The cover was designed to prevent dropping of objects into the pool and provides control within the refueling building of airborne contamination from the pool area. This cover is exempt as a heavy load traveling over the spent fuel storage pool, as it is an integral part of the pool and was analyzed as part of the original Safety Analysis Report and subsequent Safety Evaluation Report. The cover may be removed for extended periods of time to allow work to be performed in the pool. The cover may remain off while pool work is not actively taking place, such as during backshifts, weekends, and during a pause in a pool work project. Installation and removal of the cover is controlled administratively. Prior to

removing the SFP cover beams and panels, the item will be secured to the crane before it is disconnected from the pool structure. When installing the SFP cover beams and panels, the item shall be connected to the pool structure before it is removed from the crane.

- Spent fuel is stored in a configuration to provide maximum separation of assemblies during the SAFSTOR period.
- Spent fuel is only handled as necessary for loading and transfer of spent fuel shipping casks and for special inspections or tests that have been reviewed by the PSRC and approved by the Plant Manager.
- The handling of heavy loads over the spent fuel storage racks is prohibited and is controlled administratively.
- Two sources of makeup water for the pool are to be maintained.
- Maintenance is performed on structures, systems, and components as required to ensure that the spent fuel will remain in a safe storage condition.
- The refueling building bridge crane will remain stored away from the spent fuel storage pool.

The above-noted actions, in conjunction with routine surveillance and maintenance activities in effect for the plant, will ensure adequate management of spent fuel for the SAFSTOR period.

2.3.1.2 Inspection of Spent Nuclear Fuel During SAFSTOR

Spent fuel (except for assembly UD-6N) will remain stored in neutron absorbing containers, except for occasional inspections, during the SAFSTOR period. A fuel assembly may be removed from its neutron absorbing container, channels (if necessary), and storage rack to allow visual inspection. Movement of fuel, whether remaining in its neutron absorbing container, or removed for inspection, is allowed providing:

- Spent fuel handling operations are conducted in accordance with procedures reviewed by the PSRC, recommended for approval, and approved by the Plant Manager,
- Spent fuel handling operations are under the direct supervision of a management Certified Fuel Handler,
- At least one Certified Fuel Handler shall be present at the location of each fuel movement, and
- An individual qualified in radiation protection procedures shall be onsite during fuel handling operations.

The approved procedures used to conduct an inspection of fuel assemblies removed from their neutron absorbing containers will assure that:

- No more than one fuel assembly is removed from its neutron absorbing container at any time. (Removed means that the fuel assembly is not in the neutron absorbing container with 2 locking tabs in place).
- The purpose of removal is limited to inspection of the assemblies or their neutron absorbing containers.
- The inspection process for each fuel assembly shall last no longer than necessary.
- The procedure shall specify the maximum expected duration an assembly will be removed from its neutron absorbing container. If this duration is exceeded, no more than 7 additional days will be allowed for returning the assembly to the stored condition (inside the neutron absorbing container with 2 tabs in place). Beyond this period will be considered outside the limits specified in Technical Specification 4.2.1.

2.3.2 WASTE DISPOSAL SYSTEMS

The waste disposal systems in Unit 3 include the gas treatment system, the liquid waste collection system, the liquid waste treatment and disposal system, and the solid waste facilities. Collectively these systems control and dispose of all plant wastes that are normally or potentially contaminated with radioactive materials.

2.3.2.1 Gas Treatment System.

The gas treatment system (GTS) consists of two exhaust fans, a gas scrubber column, gas scrubber recirculation tank, two recirculation pumps, and associated system piping, valves, filters, instruments, and controls. The system was intended to remove halogen gases from the refueling building in the event of fuel damage and a resulting release of fission product gases. The system components are located on three levels in the base of the former main ventilation exhaust stack.

Due to the age of the spent fuel stored on site, the inventory of radioactive halogens is sufficiently low to preclude the need for the gas treatment system. The sodium hydroxide solution from the gas scrubber recirculation tank has been drained and the system secured.

The gas treatment system (GTS) can be used to mitigate accidents involving the release of airborne particulate radioactive material into the atmosphere of the refueling building. This system consists of the original gas scrubber column (with the original packing and scrubber solution removed), demister, high efficiency particulate air (HEPA) filter, and fan.

In the event of an accident that results in high airborne particulate radioactive material in the refueling building, the refueling building ventilation system can be isolated and the refueling building air is then exhausted through the gas treatment system HEPA filter.

Condenser Off-Gas System. This system has been removed from service. A detailed description of this system can be found in Appendix G. The sump pump in the vault where this system was stored will be maintained in an operational status.

2.3.2.2 Liquid Waste Collection System.

The liquid waste collection system consists of the turbine building drain tank, reactor equipment drain tank, reactor caisson sump, the turbine building floor drain pump, two turbine building drain tank pumps, two reactor equipment drain tank pumps, the reactor caisson sump pumps, the laundry waste tank, the laundry hold tank, the laundry waste filter, two laundry waste pumps, a vent separator, and a yard drain system.

The turbine building drain tank (TBDT), turbine building floor drain pump, and TBDT pumps are located at elevation -14 feet in the reactor caisson in a shielded vault beneath the new fuel storage vault. The vault is accessible via a ladder through a hatch in the new fuel storage vault.

The tank is pumped using the turbine building drain tank pump or can be valved to drain directly to the reactor equipment drain tank via the caisson floor drain system. The turbine building drain tank will continue to be used during the SAFSTOR period along with the associated valves, pumps, and instrumentation and controls.

The reactor equipment drain tank (REDT) and associated REDT pumps are located at the -66 foot level of the reactor caisson access shaft. The contents of this 500 gallon capacity tank are pumped automatically to the radwaste treatment system using either of the two REDT pumps. The reactor equipment drain tank and its associated pumps will continue to be used throughout the SAFSTOR period. They will be maintained along with associated valves, instrumentation and controls in an operable condition.

The reactor caisson sump and its associated reactor caisson sump pumps are located at the -66 foot level of the access shaft. The sump, which collects groundwater in-leakage, has a capacity of 50 gallons. The pumps normally transfer the sump's contents automatically to the discharge canal, but may be valved to the radwaste treatment system if groundwater contamination is suspected or detected through routine samples. The reactor caisson sump and its pumps (2) are required throughout the SAFSTOR period. The tank, pumps, valves, instrumentation, and controls will be maintained in an operable condition.

The laundry waste tank is a 250-gallon tank located in the power building underneath the laundry. It is suspended from the underside of the operating floor slab (elevation +20 feet), and collects potentially contaminated drains from the decontamination area. The laundry waste tank discharges to the TBDT.

The laundry waste tank, laundry hold tank, and other equipment associated with the laundry remained in operation until the completion of the SAFSTOR decommissioning activities. The laundry system has been secured. The laundry waste tank remains in service in order to collect drains from the decontamination shower and sink and other miscellaneous drains requiring processing by the radwaste processing system. It is presently planned that during

the SAFSTOR period, anti-contamination clothing and materials used will either be disposable or will be shipped off-site for cleaning.

The yard drain system is a storm water collection system located in the yard. All yard drainage from Units 1, 2 and 3 goes to the yard drain sump. Normally the water entering this sump flows out of the sump overflow to the inlet canal. Should any hazardous material enter the drainage system, a pump and necessary piping are provided to transfer the contents of the sump to either the Unit 2 oily water sump or the TBDT in order to prevent its discharge to the canal. The system continues to be used in its current configuration.

2.3.2.3 Liquid Waste Treatment System.

This system will remain operational throughout the SAFSTOR period. The liquid waste treatment system processes, stores, and provides for disposal of radioactively contaminated liquid wastes and other liquid wastes that are potentially radioactively contaminated. These wastes are first collected by the radwaste collection system and are then pumped to the radwaste building on the north side of the refueling building. The system consists of the following major equipment:

- Radwaste Building Sump Tank
- Radwaste Building Sump Pump
- Radwaste Receiver Tanks (3)
- Radwaste Pump
- Radwaste Demineralizer
- Resin Disposal Tank
- Concentrated Waste Tanks (2)
- Waste Hold Tanks (2)
- Treated Waste Pump
- Radwaste Filters (2)

In the radwaste building, wastes are handled on a batch basis with each batch being analyzed and handled appropriately in accordance with the analysis. Final disposition consists of storage awaiting offsite disposition, or disposal to the discharge canal, which flows into Humboldt Bay. There is no disposal to the ground.

The radwaste treatment facility was modified with the construction of a metal building to enclose the existing liquid radioactive waste treatment building and radioactive waste tankage area. The purpose of this modification is to minimize the potential for the spread of contamination outside of the building and to minimize the generation of potentially contaminated waste requiring processing by eliminating the need to collect rainwater from the building. The building ventilation is connected to the plant ventilation system.

Radwaste Building Sump Tank and Pump. This 250-gallon tank is located beneath the radwaste building floor and receives liquids from drains in the vicinity of the radwaste building. The sump pump is located on the operating floor of the radwaste building

(elevation +12 feet) over the sump tank. This pump automatically maintains the level of the tank and discharges to one of the waste receiver tanks.

Radwaste Receiver Tanks and Hold Tanks. Three 7,500-gallon carbon steel radwaste receiver tanks are for wastes coming from the radwaste collection system. Two 7,500 gallon carbon steel waste hold tanks are for storing treated wastes for retreatment or disposal. These tanks are located in an external section of the radwaste building, but are within the prefabricated steel radwaste enclosure.

Radwaste Pump. The radwaste pump is located in the radwaste building and takes suction from any of the five receiver or hold tanks for the purposes of processing the wastes through various equipment.

Radwaste Concentrator and Condenser. See Appendix G.

Radwaste Demineralizer. The radwaste demineralizer is a single, mixed bed unit with a flow capacity of 50 gpm. The demineralizer tank is 24 inches in diameter and was designed for 75 psig in accordance with the ASME Code. There are no provisions for regeneration; spent resins are sluiced to the resin disposal tank. The demineralizer is located in a shielded cubicle in the radwaste building.

Resin Disposal Tank. This 10,000-gallon tank is located in an individual shielded vault within the radwaste building. It is accessed through a hatch in the top of the vault. All spent resins from the various demineralizers on site are routed to this tank.

Treated Waste Pump. This pump is also located in the radwaste building and takes suction on the waste hold tanks. After sampling indicates that the contents of these tanks are within specifications, this pump is used to discharge the contents to the discharge canal. Alternate routings from this pump include (1) recirculation to either hold tank, (2) discharge to the condensate storage tank, or (3) recycle to waste receiver tanks for retreatment.

Radwaste Filters. Two radwaste filters are available in the radwaste building. These are cartridge-type filters, 50 gpm capacity, which can remove particles down to 25 microns in diameter.

2.3.2.4 Solid Radwaste System

The only equipment available for processing solid radwaste is a compactor that provides higher density loading of 55-gallon drums than can be achieved by hand loading. It is presently located in the liquid radwaste enclosure. The compactor will remain operational for processing solid waste during the SAFSTOR period.

Any waste disposal systems or components not listed in this section have either been removed from service or abandoned in place. The physical SAFSTOR statuses of all components are summarized in Table 2-1, and complete descriptions of components no longer in service can be found in Appendix G.

2.3.3 SERVICE SYSTEMS

Domestic Water System. Domestic water to Unit 3 is supplied by the Units 1 and 2 domestic water system. This system will continue to operate during the SAFSTOR period.

Fire Protection System. The fire protection system consists of three fire pumps with associated piping, valves, instrumentation, and controls. Carbon dioxide lines to Unit 3 from the CARDOX system located in Unit 2 have been isolated. Removal of oil from the lube oil system eliminated the need for this system in Unit 3. The system will remain in service.

Heating and Ventilation System. The plant heating and ventilation system consists of a single exhaust fan, a High Efficiency Particulate Air (HEPA) filter that exhausts from the refueling building to the ventilation exhaust stack, the multizone air handling unit, which supplies filtered air to the refueling building and selected areas of the power building, the drywell purge fan which ventilates the reactor-caisson access shaft, and several small air handling units that ventilate selected areas of the plant.

The heating and ventilation system will remain operational to supply filtered air to the refueling building and to exhaust air from the refueling building, hot lab, hot machine shop, and radwaste treatment building (enclosure). The system has been adjusted wherever possible to maintain flow from areas of low contamination to areas of higher contamination. Ventilation exhaust is through the ventilation exhaust stack, which is provided with the stack monitoring system to monitor any release. This effluent receives no routine treatment.

The approximate distance from the stack to the site boundary in each of the 16 compass direction sectors is listed in Table 2-4.

Refueling Building Ventilation System. The refueling building ventilation system shall provide normal ventilation to the refueling building. The system shall exhaust to the main ventilation exhaust stack. Isolation valves shall be provided to permit isolation of the refueling building. NOTE - Isolation is the ability to separate the refueling building ventilation system from the rest of Unit 3 and to limit the leakage to a rate of 134 CFM when the refueling building is under a negative pressure of at least ¼ inch of water. The separation of the refueling building ventilation shall be done through the use of isolation valves. These valves are CV 4352 and CV 4353, which are used to isolate the dry well purge penetration, CV 4355 and CV 4356, which are used to isolate the main plant exhaust system from the refueling building, and CV 4354, which is the 16-inch ventilation inlet valve to the refueling building. These isolation valves shall be capable of a leak rate that does not limit the capability to control the refueling building leakage to a maximum of 134 CFM, when the building is under a negative pressure of at least ¼ inch of water.

Other than the areas discussed above, there are no other locations for the release of airborne radioactive material from buildings.

The structure that previously held the (never operational) condenser offgas treatment system was and still is an uncontaminated area, and the future use of this structure does not involve radioactive materials. This structure has a ventilation system (separate from the Unit 3 system) with a discharge that is neither monitored, nor required to be monitored.

No controlled ventilation is provided (or needed) for the waste storage vaults, the low-level waste storage building, or the low-level waste handling building. Wastes in these locations will be packaged prior to storage to preclude a potential for release of airborne radioactivity. As decommissioning and SAFSTOR activities progress, the ventilation system may be modified to reduce airflow or to secure airflow to unoccupied areas of Unit 3. In addition, ventilation from the radwaste treatment facility has been tied into the ventilation system.

Manlift. An electric operated manlift provides access between elevations -14 feet and -66 feet within the access shaft of the caisson. The manlift will be maintained in an operational condition.

75-Ton Bridge Crane (or Refueling Building Crane). This crane is supported at elevation 35 feet 9 inches in the refueling building. The crane is used to handle the reactor vessel head, fuel transfer cask, shipping casks, the service platform, and other heavy components within the refueling building. The crane bridge, trolley, and trucks are constructed of built-up steel members with welded, riveted, and bolted connections. The bridge consists of two box girder sections spanning 41 feet between rails, which are supported on built-up steel girders spanning 20 feet between refueling building columns. A 10-ton capacity auxiliary hook provides additional range, speed, and simplicity for handling smaller loads.

While the crane is not anticipated to be routinely required during the SAFSTOR period, it will ultimately be required to support spent fuel shipping operations and final plant dismantlement. The crane will also be available for use during the SAFSTOR period. To prevent deterioration of the crane, PG&E will maintain the crane in an operable, standby status during SAFSTOR. The reactor vessel refueling platform will be stored and steps will be taken to minimize its susceptibility to deterioration.

Hydrogen and Seal Oil System. Hydrogen gas was utilized to cool the main generator during operation and also to provide a passive environment for electrical insulation. Seal oil was provided to the air-hydrogen seals of the generator to aid in maintaining air-hydrogen separation.

The seal oil system has been drained and hydrogen bottles have been removed and system openings sealed. No further action is considered necessary. The system is available, but not actively maintained.

Sanitary System. The existing sanitary system for Unit 3 will be maintained and remain functional during the SAFSTOR period. This system is common with the site sanitary system.

2.3.4 ELECTRICAL SYSTEMS

Switchgear shall be provided so that Unit 3 electrical equipment can receive power from any one of three independent power sources: Unit 1, Unit 2, or the Plant 60 kV bus.

The Electrical Systems at HBPP Unit 3 consist of (1) the main generator and its 13.8 kV system, (2) the 115 kV system (which originates from step-up of 13.8 kV), (3) the auxiliary power system which includes a 2400V system, a 480V system (including an emergency

480V system) and a 208/120V system, (4) the 125V DC system, (5) the preferred 120V AC system, (6) the annunciator system, and (7) the communications system.

The auxiliary power system (including the 2.4 kV, 480V, emergency 480V, and the 208/120V distribution systems) will remain in service during the SAFSTOR period. As system lay-ups are completed, individual loads no longer required are disconnected from these systems. Some loads may be relocated to different load centers to consolidate and equalize loadings, but a loss of redundancy in those systems that require redundant power supplies will not be permitted.

Since the main generator and reactor will not be operational at any time during the SAFSTOR period, the 13.8 kV and 115 kV systems will not be operational, and the preferred 120V AC system will not be required to supply regulated power for nuclear reactor safety control and instrumentation.

These portions of the overall plant electrical system will remain deenergized and electrically isolated and will not be further discussed in this report. The following sections describe the remainder of the electrical systems.

2400V System. The 2400V system, as the higher voltage component of the auxiliary power system, provides power through step down transformers for the remainder of the auxiliary power system.

House Transformer No. 2 is the normal source of power for Unit 3. House Transformer No. 3 is out of service and electrically isolated. House Transformer No. 2 can be supplied by either operating Unit via the plant 60 kV bus, from the MEPPs or from either of two 115 kV transmission lines from the Cottonwood Substation via the Humboldt Substation and the plant 60 kV bus.

The 2400V system consists of a single distribution bus, with power from the bus supplied to two 2400/480V transformers. One of the transformers supplies power to Load Center (LC) No. 10, and the other supplies power to LC No. 11. Plant low voltage distribution originates at these two load centers with feeders to other motor and valve control centers, lighting and heating transformers, and power panels.

Centralization of distribution has occurred subsequent to reactor defueling, and additional centralization may occur during the SAFSTOR period. This reduces the number of centers, transformers, and panels that must be maintained, but does not affect the reliability of the power source for equipment important to safety. No changes have been or will be made that would result in greater loads than have been experienced during normal plant operation or which would result in less backup power availability for this equipment than has been available during normal shutdown periods for the plant in the past.

480V System. The 480V system consists of LC Nos. 10, 11, and 13, and Valve Control Center No. 1. LC Nos. 10 and 11 (located in the control room) are supplied power from 2400/480V Transformers No. 5 and 6, respectively. LC No. 10 has a physically and electrically separate emergency section that provides power to Power Panel No. 1. LC No. 10 has a physically and electrically separate emergency section that provides power to various emergency loads on Power Panel No. 1. LC No. 10 also provides power to LC No.

13 in the radwaste treatment building. The emergency section of LC No. 10 switches automatically to LC No. 5 of Unit 1 upon loss of power from LC No. 10. Simultaneously with this transfer, the propane-fueled emergency engine generator starts, and after the generator attains rated voltage and frequency, the emergency bus supply is transferred to this source. Return to normal configuration is accomplished manually.

208/120V System. The 208/120V system is a 3-phase, 4-wire distribution system, which supplies single-phase, 120V power for station lighting, convenience outlets, and fractional horsepower motors. Power to this system is supplied through 208/120V transformers from LC Nos. 10 and 11.

125V DC System. The 125V DC System consists of a 58-cell storage battery (125V DC Battery No. 3), two chargers, and the 125V, DC Distribution Panel No. 1. The chargers operate in parallel to carry the continuous load and float charge the battery. The battery leads terminate first in a disconnect link cabinet. From this point they are run and connect directly to 125V DC Distribution Panel No. 1. The DC distribution is made from this panel to the various boards, switchgear, and motor starters requiring DC service for control and power. The DC section of Valve Control Center No. 1 is powered by the 125V DC Distribution Panel No. 1. The 125V DC Battery No. 4 (non-vital), the 125V DC Distribution Panel No. 2, and two associated chargers (No. 2 and 4) have been removed from service.

Due to significantly reduced DC requirements during SAFSTOR, Battery No. 3 has been retained in service at reduced load. During the SAFSTOR period, Unit 3 DC load may be transferred to the Unit 2 battery and Battery No. 3 removed for salvage.

Annunciator System. The annunciator system permits efficient operation of the station with a minimum number of operators by announcing (visually and audibly) any change in operating conditions requiring attention. An operator can silence the audible horn or bell but visual annunciation will persist until the trouble has cleared and the initiating alarm device has been reset. Annunciators for systems, which have been removed from service, have been deenergized and the alarm boxes removed from the annunciator panel. During SAFSTOR it is not planned to continuously man the Unit 3 control room. A remote annunciator has been installed in Unit 2 to indicate an alarming condition in Unit 3.

Communication System. The existing communication system will remain functional during the SAFSTOR period. This system consists of the following:

- Telephones connected to and operated with the plant's Private Automatic Exchange (PAX)
- The code call system for locating personnel
- The emergency signal system for sounding of emergency alert and code signals (including siren alert)
- The voice communication system to provide quick communication between selected areas of the plant
- Intercom system

2.3.5 RADIATION MONITORING SYSTEMS

2.3.5.1 Area Monitors

Area monitors in the following locations will remain in service:

- Control Room
- Refueling Building (+27 feet El.)
- Refueling Building (+ 12 feet El.)
- Caisson Access Shaft (- 66 feet El.)
- Radwaste Treatment Facility
- Radwaste Handling Building

All other area monitors will be secured. See Section 3.5.3.7 of the DSAR for additional information about area monitors.

2.3.5.2 Stack Radiation Monitoring System

The stack gas monitoring system consists of an isokinetic sampling device (located at the top of the 50-foot plant stack), a quick disconnect particulate filter holder, two shielded gas sample chambers (connected in series), beta scintillation detectors (situated one in each gas sample chamber), a flow regulator, and carbon vane pump. A 2-cfm sample of stack gas is continuously pulled from the isokinetic sampler through the particulate filter, then through each sample chamber, the flow regulator, and into the pump. The pump discharges into plant ventilation ductwork leading back to the stack.

The particulate filter is replaced weekly and the old filter is analyzed in the plant laboratory to determine particulate activity in stack effluent. Particulate activities down to the range of 10 μCi released over a calendar quarter may be detected using this approach.

The beta scintillation detectors consist of a thin Mylar window and phosphor crystal, photomultiplier tube, and preamplifier (mounted in a lightproof, watertight probe). Setpoints are based on gaseous effluent discharge limits using the Offsite Dose Calculation Manual (ODCM).

The stack monitors are designed to be sensitive to ^{85}Kr in the stack gas for a range from approximately 5×10^{-7} μCi/cc to 2×10^{-2} μCi/cc. This range is intended to detect a small fraction of the 40 CFR 190 limits for routine operation, and estimated maximum anticipated releases following an accident that results in damage to spent fuel assemblies.

2.3.5.3 Process Radiation Monitoring System

The radwaste discharge line monitor uses a gamma-sensitive scintillation detector consisting of a sodium iodide crystal (thallium-activated), and a photomultiplier tube, mounted in a light proof, watertight probe. The detector is mounted in a sample chamber bolted into the liquid radioactive waste discharge line. The detector monitors the activity of the water flowing through the liquid radioactive waste discharge line and is connected to a preamplifier. The preamplifier is then connected to a rate meter located in the Unit 3 control room. The rate meter displays the incoming count rate in logarithmic form and has a range of 10 to 10^6 cpm. An alarm is provided to alert personnel if elevated levels of radioactivity are being released into the discharge canal. Radwaste discharge pumps can be turned off from within the control room. Process alarm levels are set to assure that the limitations on the instantaneous (averaged over a one hour period) concentrations of radioactive material being released to Humboldt Bay conform to ten times the effluent concentration limits of 10 CFR 20, Appendix B, Table 2, column 2; provided that at least one circulating water pump is in operation as described in the ODCM.

The discharge canal sample station is designed to collect a composite, representative sample of the discharge canal water being released into Humboldt Bay.

The sample station consists of a small electric motor-driven sample pump, a small motor-driven metering pump, piping for sample collection and system back flush piping from the plant fire water system. The sample pump continuously draws from the discharge canal with water flowing into a sample scupper and back into the canal. The metering pump continuously draws from the scupper into a 5-gallon sample bottle. The sample is periodically collected and analyzed for radioactivity. This system is intended to provide a final check to assure liquid radioactive effluent limits are not being exceeded.

No other effluent and process monitoring or sampling systems are planned for SAFSTOR. Grab samples are utilized as required to determine activity levels in other process streams.

2.3.6 INSTRUMENTATION AND CONTROL (I&C) SYSTEMS

Unit 3 was provided with numerous I&C systems to optimize plant performance, protect equipment from damage, protect plant operating personnel, and protect the public and the environment from harm due to accidents of a radiological or nonradiological nature. Table 2-1 identifies the major I&C systems and their status during the SAFSTOR period. For those systems or parts of systems that were removed from service in preparation for SAFSTOR, or those that may be removed as convenient during the SAFSTOR period; the following typical actions may have been or may be accomplished subsequent to such removal:

- Physical removal of instruments or controls from the installed position. If such removal exposes contaminated system internals, openings will be appropriately resealed to prevent the spread of contamination.

- Contaminated instruments or controls will be (1) decontaminated for salvage or

disposal, (2) disposed of as radwaste, or (3) stored for potential future use in a manner to minimize deterioration and prevent the spread of contamination.

- Noncontaminated instruments or controls will be salvaged or stored for future use in a manner to minimize deterioration.

The general approach for preparation of instrumentation and controls for the SAFSTOR period was to remove from service all such equipment not required to support "continued care" operations and to inspect, perform maintenance, and test as necessary all such equipment which must remain operational to provide reasonable assurance of its continued, reliable performance. For those instrumentation and controls, which are to remain in a standby condition or in continuous or intermittent operation, a maintenance and calibration program was instituted to ensure reliable performance and availability through the "continued care" period of SAFSTOR.

2.3.7 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system (NSSS) consists of the reactor vessel and internals, control rod system, liquid poison system, reactor cleanup system, reactor shutdown cooling system, emergency condenser system, and the suppression tank cooling and core spray system.

The components of the nuclear steam supply system are either abandoned in place, or have been removed altogether, with the exception of a liquid level detection system that has been added to the suppression chamber. Table 2-1 lists the individual components and their SAFSTOR status. The components are described in detail in Appendix G.

2.3.8 TURBINE PLANT SYSTEMS

The turbine plant systems consist of the turbine system, the lube oil system, the condensate system, the gland seal system, the feedwater system, the closed cooling water system, the main circulating water system (saltwater), the compressed air system, and the demineralized water system. Most of the systems and their components are abandoned in place, or have been removed completely, with the exception of the items described below. The physical SAFSTOR status of the systems and components in this section are summarized in Table 2-1. Complete descriptions of the systems and components not included in this section can be found in Appendix G.

<u>Condensate Storage Tank</u>. The condensate storage tank is located outside the north side of the refueling building. It is a 34,000 gallon net capacity, 15 feet in diameter by approximately 29 feet high, cone-roof, vertical cylindrical storage tank of aluminum construction. An internal head separates the tank into an upper 5,000 gallon section (designated the demineralized water storage tank) for fresh demineralized water and a lower 29,000 gallon section for condensate. The condensate storage tank section has been drained. The demineralized water storage tank remains operational.

Compressed Air System. The compressed air system was designed to provide service air and instrument air throughout the plant. The system consists of the instrument air compressor, the aftercooler, instrument air receiver, service air receiver, instrument air dryer, and filters. Headers are provided for both service air and instrument air, and these headers are interconnected with equivalent headers in Units 1 and 2. Normally all compressed air for Unit 3 is supplied through this interconnection.

The Unit 3 air compressor (No. 5 air compressor) will remain available as a standby unit during SAFSTOR. Due to the lay-up of the Unit 3 closed cooling water system, the No. 5 air compressor and the aftercooler have been modified to receive cooling from the Unit 2 bearing cooling water system. The system is located in the reactor feed pump room. The service air and instrument air systems are presently in service and will remain in service during the SAFSTOR period.

Demineralized Water System. Demineralized water for Unit 3 is supplied from the Units 1 and 2 demineralized water systems. The demineralized water storage tank (top section of the condensate storage tank), the demineralized water pump, and distribution system piping and valves will remain in service during the SAFSTOR period.

Lube Oil System. The turbine lube oil system for Unit 3 is a complete and self-contained system consisting of a reservoir, pumps, coolers, bowser filter, and necessary piping.

The main and auxiliary pumps delivered high pressure oil to the turbine hydraulic control system and cooled, low pressure oil to the bearing header which supplies the unit bearings and other parts to be lubricated. Backup protection to the bearing oil system was provided by the turning gear oil pump and the DC emergency oil pump. The bearing oil was drained to a common header and was returned back to the reservoir. A related system, the hydrogen and seal oil system, received its oil supply from this system. The lube oil system has been drained and isolated from other systems. The Lube Oil System is considered available however; no further use of this system is planned.

2.4 UNITS I AND 2 SYSTEMS COMMON TO UNIT 3

The following Units 1 and 2 systems supply services to Unit 3, or they are systems shared jointly by these units, which will be required during the continued care period of SAFSTOR. The following descriptions describe the extent of the connections between the units and what services will continue to be provided to Unit 3 during SAFSTOR.

Fire Protection System. A fire water loop around Unit 3 ties to the loop around Units 1 and 2 north and south of the station building. The system will remain in service. The emergency low-pressure core flooding line, the emergency makeup line to the dry-well air coolers, and the reactor shield coolers have been disconnected from the fire main and blank flanged. This reduces the possibility of a leak in an abandoned system affecting the fire system.

Compressed Air System. Both the service air and instrument air headers of all three units are tied together. It is planned to maintain the Unit 3 air compressor and its aftercooler in standby with compressed air supplied from Unit 2. The remaining ancillary equipment

(receivers, dryers, and filters) will be maintained operational.

Demineralized Water. Demineralized water services to Unit 3 are supplied by makeup water from the Units 1 and 2 evaporators. Makeup water is normally transferred from one of the four Units 1 and 2 demineralized water storage tanks to the Unit 3 demineralized water storage tank for use in Unit 3.

Auxiliary Steam. A steam line supplies steam to the heating coils of the Unit 3 control room air handling unit. The air handling unit will continue to receive auxiliary steam.

Domestic Water. A line from the Units 1 and 2 domestic water system is piped to Unit 3. This system provides laundry, sanitary, and drinking water in the operating floor area. The system will continue to be used in its present configuration.

Oily Water Drains. The floor drains serving the Unit 3 turbine lube oil reservoir and the air compressor are routed to the Unit 2 oily water separator. The system will continue to be used in its present configuration.

Yard Drains. Outside yard drains serving Units 1, 2, and 3 are interconnected and normally flow to the inlet canal, with interception in a yard drain sump in case of spills. The yard drain sump can be valved to the oily water separator or the turbine building drain tank through the turbine building drain system. The system continues to be used in its present configuration.

Unit 2 Bearing Cooling Water System. Bearing cooling water is routed from Unit 2 to the Radwaste Building for use as a sink for the heat rejected by the radwaste concentrator vapor condenser. It is also routed to the reactor feed pump room to provide cooling water for the Number 5 air compressor.

Radwaste Discharge to Units 1 and 2 Discharge Tubes. The radwaste system effluent discharge line to the Units 1 and 2 discharge tubes mixes with the cooling water before entering the outfall canal; this line will remain operational during the SAFSTOR period.

Minimum dilution flow can be provided by one of the circulating water pumps supplying either Unit 1 or Unit 2. Each unit has two circulating water pumps, each with a capacity of 12,500 gpm (nominal). The radioactive waste discharge line can be connected to the circulating water discharge line from either unit.

Electrical. The 2400V bus in Unit 3 is supplied from the 2400V buses in either Unit 1 or Unit 2. House transformer No. 2 has been and will continue to be utilized to be the normal supply of Unit 3, 2400V power through the SAFSTOR period.

TABLE 2-1
Humboldt Bay Power Plant, Unit 3
Physical Characterization-Summary

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
1. Spent Fuel Storage Pool and Associated Systems		2.3.1	
• Liner Gap Pump	Refueling Building (+'12')	2.3.1	Available – Actively maintained
• Fuel Pool Circulating Water Pump (2)	Refueling Building ('24')	2.3.1	Available – Actively maintained
• Fuel Pool Coolers (2)	Refueling Building (+'12')	Appendix G	Abandoned – In place
• Fuel Pool Skimmer	Refueling Building (+'12')	None (Note 2)	Available – Not actively maintained
• Channel Handling Tools	Refueling Building (+'12')	2.3.1	Available – Not actively maintained
• Manual Fuel Handling Tools	Refueling Building (+'12')	2.3.1	Available – Not actively maintained
• Spent Fuel Pool Jib Crane (500 lb)	Refueling Building (+'12')	2.3.1	Available – Actively maintained
• Extension Tank and Refueling Platform	Refueling Building (+'12')	2.3.1	Available – Not actively maintained
• Transfer Cask and Winch	Refueling Building (+'12')	2.3.1	Available – Not actively maintained
• SFP Demineralizer	SFP Demin Room (+ 12)	2.3.1	Available – Actively maintained
• SFP Demineralizer Strainer	SFP Demin Room (+ 12)	None (Note 2)	Available – Actively maintained
• SFP Water Level Monitors (2)	Refueling Building (+'12'), Control Room	2.3.1	Available – Actively maintained
2. Waste Disposal Systems		2.3.2	
2.(a) Gas Treatment System		2.3.2.1	
• Exhaust Fans (2)	Former Stack (+27')	2.3.2.1	Available – Actively maintained
• Gas Scrubber Column	Former Stack (0')	2.3.2.1	Available – Not actively maintained
• Gas Scrubber Recirc. Tank	Former Stack (0')	2.3.2.1 (Note 1)	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
2.(a) Gas Treatment System (cont.)		2.3.2.1	
• Gas Scrubber Recirc. Pumps	Former Stack (0')	2.3.2.1 (Note 1)	Abandoned – In place
• Absolute Filter (Scrubber Column Exhaust)	Former Stack (+ 12')	2.3.2.1	Available – Actively maintained
• Holdup Piping	Yard, below grade	2.3.2.1	Available – Not actively maintained
2.(b) Condenser Off-gas Treatment System (New)		Appendix G	
• Jet Compressors	Condenser Off-gas Treatment Vault	None	Removed
• Preheater	Cond. Off-gas Treatment Vault	None	Removed
• Recombiners	Cond. Off-gas Treatment Vault	None	Removed
• Condenser/Moisture Separator	Cond. Off-gas Treatment Vault	None	Removed
• Off-gas Prefilter (HEPA)	Cond. Off-gas Treatment Vault	None	Removed
• Refrigerant Air Dryer	Cond. Off-gas Treatment Vault	None	Removed
• Desiccant Towers/Dryer Subsystems (2)	Cond. Off-gas Treatment Vault	None	Removed
• Carbon Guard Bed	Cond. Off-gas Treatment Vault	None	Removed
• Carbon Absorber Columns	Cond. Off-gas Treatment Vault	None	Removed
2.(c) Liquid Waste Collection System		2.3.2.2	
• Turbine Building Drain Tank (TBDT)	Below New Fuel Storage Vault (-14')	2.3.2.2	Available – Actively maintained
• Turbine Building Floor Drain Pump	Below New Fuel Storage Vault (-14')	2.3.2.2 (Note 1)	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
2.(c) Liquid Waste Collection System (continued)		2.3.2.2	
• TBDT Pumps (2)	Below New Fuel Storage Vault (-14')	2.3.2.2	Available – Actively maintained
• Reactor Equipment Drain Tank (REDT)	Access Shaft (-66')	2.3.2.2	Available – Actively maintained
• REDT Pumps (2)	Access Shaft (-66')	2.3.2.2	Available – Actively maintained
• Reactor Caisson Sump	Access Shaft (-66')	2.3.2.2	Available – Actively maintained
• Reactor Caisson Sump Pumps (2)	Access Shaft (-66')	2.3.2.2	Available – Actively maintained
• Yard Drain System	Yard	2.3.2.2	Available – Actively maintained
• Laundry Waste Tank	Pipe Tunnel (+20')	2.3.2.2	Available – Actively maintained (Modified to be used as the drain for the decon sink and shower.)
• Laundry Waste Filter	Condensate Demin. Room (+ 12')	None (Note 2)	Available – Not actively maintained
• Laundry Hold Tank (Formerly Regenerated Resin Storage Tank)	Condensate Demin. Room (+ 12')	Appendix G	Abandoned – In place
• Laundry Waste Pumps	Condensate Demin. Room (+ 12')	None	Abandoned – In place
• Vent Separator	Valve Gallery (+0')	None	Abandoned – In place
2.(d) Liquid Waste Treatment		2.3.2.3	
• Radwaste Bldg. Sump Tank	Radwaste Building	2.3.2.3	Available – Actively maintained
• Radwaste Bldg. Sump Pump	Radwaste Building	2.3.2.3	Available – Actively maintained

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
2.(d) Liquid Waste Treatment (cont.)		2.3.2.3	
• Radwaste Receiver Tanks (3)	Radwaste Building	2.3.2.3	Available – Actively maintained
• Radwaste Pump	Radwaste Building	2.3.2.3	Available – Actively maintained
• Concentrator Feed Pump	Radwaste Building	Appendix G	Abandoned – In place
• Radwaste Concentrator	Radwaste Building	Appendix G	Abandoned – In place
• Radwaste Concentrator Condenser	Radwaste Building	Appendix G	Abandoned – In place
• Radwaste Demineralizer	Radwaste Building	2.3.2.3	Available – Actively maintained
• Resin Disposal Tank	Radwaste Building	2.3.2.3	Available – Actively maintained
• Concentrated Waste Tanks (2)	Radwaste Building	Appendix G	Abandoned – In place
• Waste Hold Tanks (2)	Radwaste Building	2.3.2.3	Available – Actively maintained
• Treated Waste Pump	Radwaste Building	2.3.2.3	Available – Actively maintained
• Radwaste Filters(2)	Radwaste Building	2.3.2.3	Available – Actively maintained
• Spent Fuel Pool Filter	Radwaste Building	Appendix G	Removed
• Resin Addition Tank	Radwaste Building	None	Removed
• Concentrator Flash Pot	Radwaste Building	None	Abandoned – In place
• Concentrator Drip Receiver Tank	Radwaste Building	Appendix G	Abandoned – In place
• Absolute Filter (Liquid Waste Tankage Vents)	Radwaste Building Roof	None	Removed

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
2. Waste Disposal Systems (cont.)			
2.(e) Solid Radwaste System		2.3.2.4	
• Compactor	Liquid Rad-waste encl.	2.3.2.4	Available – Actively maintained
3. Service Systems		2.3.3	
3.(a) Fire Protection System		2.3.3	
• Fire Pumps (3)	Fire Pump House	2.3.3	Available – Actively maintained
3.(b) Heating and Ventilation System		2.3.3	
• Multizone Air Handling Unit	Over Laundry (+'37')	2.3.3	Available – Actively maintained
• Air Handling Unit #1	Control Room Roof	None (Note 2)	Available – Actively maintained
• Air Handling Unit #2	Yard	None	Removed
• Air Handling Unit #3	Turbine Encl. Roof	None	Removed
• Reactor Feedpump Rm. Supply Fan	Reactor Feed Pump Rm (+12')	None	Abandoned – In place
• Reactor Feedpump Rm. Exhaust Fan	Reactor Feed Pump Room (+12')	None	Abandoned – In place
• Turbine Bldg. Exhaust Plenum (No. 1)	Pipe Tunnel	None (Note 2)	Available – Actively maintained
• Refueling Bldg. Exhaust Plenum	Yard	None (Note 2)	Available – Actively maintained
• Plant Exhaust Fan	Yard	2.3.3	Available – Actively maintained
• Drywell Cooling Unit	Lower Drywell	None	Abandoned – In place
• Drywell Purge Fan	Former Stack ('27')	None (Note 2)	Available – Actively maintained
• Lab Hood Exhaust Fan	Hot Lab Roof	None (Note 2)	Available – Actively maintained

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
3. Service Systems (continued)			
3.(b) Heating and Ventilation System (continued)		2.3.3	
• Absolute and Roughing Filter (Hot Lab)	Hot Lab Roof	None (Note 2)	Available – Actively maintained
• Absolute Filter	Laundry	None	Removed
• Exciter Cooling Fan	Exciter House	None (Note 2)	Available – Not actively maintained
• Heater and Fan Unit	Hot Machine Shop	None (Note 2)	Available – Actively maintained
• Exhaust Fan	Hot Machine Shop	None (Note 2)	Available – Actively maintained
• Absolute and Roughing Filter (Hot Machine Shop)	Hot Machine Shop	None (Note 2)	Available – Actively maintained
• Heater and Fan Unit	Inst. Repair Room	None (Note 2)	Available – Actively maintained
• Isokinetic Sampler	Stack (Top)	2.3.5	Available – Actively maintained
3.(c) Hydrogen and Seal Oil System		2.3.3	
• Hydrogen Coolers	Generator Housing	None (Note 2)	Available – Not actively maintained
• Hydrogen Dryer	Seal Oil Room (+6')	None (Note 2)	Available – Not actively maintained
• Seal Oil Storage Tank	Seal Oil Room (+6')	None (Note 2)	Available – Not actively maintained
• Main Seal Oil Pump	Seal Oil Room (+6')	None (Note 2)	Available – Not actively maintained
• Emergency Seal Oil Pump	Seal Oil Room (+6')	None (Note 2)	Available – Not actively maintained
• Seal Oil Filters (2)	Seal Oil Room (+6')	None (Note 2)	Available – Not actively maintained
3.(d) Reactor Shield Cooling System		Appendix G	
• Cooling Coils	Drywell (Embedded)	Appendix G	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
3.(e) Cranes		2.3.3	
• 75-Ton Bridge Crane	Refueling Building	2.3.3	Available – Not Actively maintained
• The 10-Ton Aux. Hook and Assoc.Components (Part of the 75-Ton Bridge Crane)	Refueling Building	2.3.3	Avaliable – Actively maintained
• 2-Ton Jib Crane	Refueling Building	None (Note 2)	Available – Actively maintained
• 5-Ton Hot Machine Shop Crane	Hot Machine Shop	None (Note 2)	Available – Actively maintained
3.(f) Security System	Plant wide	None	Available - Actively maintained
4. Electrical Systems		2.3.4	
4.(a) Protective Relay System	Control Room, Plant wide	None (Note 2)	Available – Actively maintained
4.(b) Annunciator System	Control Room, Plant wide	2.3.4	Available – Actively maintained
4.(c) Communications Systems	Plant wide	2.3.4	Available – Actively maintained
4.(d) Emergency AC Systems	Control Room, Yard	2.3.4	Available – Actively maintained
4.(e) Preferred AC System (Motor-Generator AC System)	Reactor Feed Pump Room (+12')	2.3.4 (Note 1)	Abandoned – In place
4.(f) 125 Volt DC System		2.3.4	
• Vital	Adjacent to Counting Room	2.3.4	Available – Actively maintained
• Non-vital	Control Room, Yard	2.3.4 (Note 1)	Removed
4.(g) Auxiliary Power Systems		2.3.4	
• 480 Volt System	Control Room, Yard	2.3.4	Available – Actively maintained
• 2400 Volt System	Control Room, Yard	2.3.4	Available – Actively maintained

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
4.(g) Auxiliary Power Systems (continued)		2.3.4	
• 208/120 Volt Distribution System	Plant wide	2.3.4	Available – Actively maintained
4.(h) 13.8 KV System	Yard	2.3.4	Available – Not actively maintained
5. Radiation Monitoring Systems		2.3.5	
5.(a) In-core Flux Monitoring System	Control Room, Drywell	None	Abandoned – In place
5.(b) Area Radiation Monitoring System	Control Room, Plantwide	2.3.5	Available – Actively maintained
5.(c) Refueling Bldg. Isolation Monitoring System	Control Room, Ref. Bldg., Access Shaft	None	Abandoned – In place
5.(d) Stack Gas Radiation Monitoring	Control Room, Former Stack Base	2.3.5	Available – Actively maintained
5.(e) Off-gas Monitoring System	Control Room, Air Ejector Rm.	None	Abandoned – In place
5.(f) Process Radiation Monitoring System	Control Room, Radwaste Bldg	2.3.5	Available – Actively maintained
• Condensate Demineralizer LIne	Cond. Demin. Hallway	None	Removed
• Closed Cooling Water return Line to Storage Tank	Condenser Bay	None	Removed
• Liquid Waste System Vent Monitor	Radwaste Bldg	None	Removed
• Radwaste Discharge Line to Outfall Canal	Radwaste Bldg	2.3.5	Available – Actively maintained
• Reactor Water to Cleanup Demineralizer	Control Room, Access Shaft	None	Removed
5. (g) Discharge Canal Sample Station	Outfall Canal	2.3.5	Available – Actively maintained

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
6. Control and Instrumentation		2.3.6	
• Reactor Protection System	Control Rm/Drywell	None	Abandoned – In place
• Reactor Nuclear Instrumentation (including Out-of Vessel and In-Core Neutron Monitoring Systems)	Control Rm/Drywell	None	Abandoned – In place
• Reactor Vessel Instrumentation	Control Room, Drywell, Access Shaft	None	Abandoned – In place
• Containment Leak Rate Monitoring	Access Shaft, Reactor Feed Pump Room	None	Abandoned – In place
• Control Rod Position Indication	Control Room, Drywell, Access Shaft	None	Abandoned – In place
• Control Rod Drive Instrumentation	Control Room, Acc. Shaft ('44')	None	Abandoned – In place
• Feed water Control System	Control Room, React. Feed Pump Room	None	Abandoned – In place
• Meteorological Facility	North of Oil Storage Tanks	None	Removed
7. Nuclear Steam Supply System			
7.(a) Reactor Vessel and Internals		Appendix G	
• Reactor Vessel	Drywell	Appendix G	Abandoned – In place
• Control Rods (32)	Reactor Vessel	Appendix G	Abandoned – In place
• Control Rod Guide Tubes (32)	Reactor Vessel	Appendix G	Abandoned – In place
• Core Support Assembly	Reactor Vessel	Appendix G	Abandoned – In place
• Lower Core Shroud	Reactor Vessel	Appendix G	Abandoned – In place
• Upper Core Shroud	Reactor Vessel	Appendix G	Abandoned – In place
• Upper Guide	Reactor Vessel	Appendix G	Abandoned – In place
• Chimney	Reactor Vessel	Appendix G	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
7.(a) Reactor Vessel and Internals (continued)			
• Feedwater Sparger	Reactor Vessel	Appendix G	Abandoned – In place
• Steam Dryer	Reactor Vessel	Appendix G	Abandoned – In place
7.(b) Control Rod Hydraulic System		Appendix G	
• System Piping	Drywell, Access Shaft & Refuel.Building	Appendix G	Abandoned – In place
• Control Rod Drives	Drywell	Appendix G	Abandoned – In place
• Accumulators	Access Shaft (-54')	Appendix G	Abandoned – In place
• Supply Pumps	Refueling Building (+ 12')	Appendix G	Abandoned – In place
• Scram Dump Tank	Access Shaft (-66')	Appendix G	Removed
7.(c) Liquid Poison System		Appendix G	
• System Piping	Refueling Building (+ 12')	Appendix G	Abandoned – In place
• Liquid Poison Tank	Refueling Building (+ 12')	Appendix G	Abandoned – In place
7.(d) Reactor Cleanup System		Appendix G	
• System Piping	Access Shaft (-66' to -2')	Appendix G	Abandoned – In place
• Cleanup Pump	Access Shaft (-66')	Appendix G	Removed
• Regenerative Heat Exchangers	Cleanup Heat Exch. Room (-2')	Appendix G	Abandoned – In place
• Non-regenerative Heat Exchangers	Cleanup Heat Exch. Room (-2')	Appendix G	Abandoned – In place
• Demineralizer	Cleanup Heat Exch. Room (-2')	Appendix G	Abandoned – In place
• Resin Storage Tank	Ref. Building (+12')	Appendix G	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
7.(e) Reactor Shutdown Cooling System		Appendix G	
• System Piping	Access Shaft (-14')	Appendix G	Abandoned – In place
• Reactor Shutdown Pumps	Shutdown Room (-14')	Appendix G	Abandoned – In place
• Shutdown Heat Exchangers	Shutdown Room (-14')	Appendix G	Abandoned – In place
7.(f) Emergency Condenser System		Appendix G	
• Emergency Makeup Pump	Yard	Appendix G	Abandoned – In place
• Emergency Condenser	Refueling Building (+28')	Appendix G	Abandoned – In place
7.(g) Suppression Tank Cooling and Core Spray System		Appendix G	
• Core Spray Pumps (2)	Access Shaft (-66')	Appendix G	Removed
• Suppression Chamber Cooler	Access Shaft (-2')	Appendix G	Abandoned – In place
• Low Pressure Core Flooding Supply Line	Access Shaft (-14') Ref. Building (+ 12')	None (Note 2)	Available – Not actively maintained
• Suppression Chamber	Reactor Caisson	Appendix G	Abandoned In Place
• Suppression Chamber Liquid Level Monitor	Reactor Caisson	2.3.7	Available – Actively maintained
8. Turbine Plant Systems			
8. (a) Turbine Systems		Appendix G	
• Turbine	Turbine Encl. (+27')	Appendix G	Abandoned – In place
• Main Steam Stop Valve	Power Building Pipe Tunnel (+6')	None	Abandoned – In place
• Hydrogen Cooler	Seal Oil Room (+6')	None	Available – Not actively maintained
• Hydrogen Oil Seal Cooler	Main Generator	None	Available – Not actively maintained

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
8. (a) Turbine Systems (cont.)			
• Turbine Bypass Valves	Power Building Condenser Rm. (+ 12')	None	Abandoned – In place
8. (b) Lube Oil System		2.3.8	
• Bowser Return Pump	Reactor Feed Pump Rm.(+ 12')	2.3.8	Available – Not actively maintained
• Lube Oil Coolers	Reactor Feed Pump Rm.(+ 12')	2.3.8	Available – Not actively maintained
• Bowser Tank Lube Oil Filters	Reactor Feed Pump Rm.(+ 12')	2.3.8	Available – Not actively maintained
• Vapor Extractor Pump	Reactor Feed Pump Rm.(+ 12')	None (Note 2)	Available – Not actively maintained
• Oil Driven Booster Pump	Reactor Feed Pump Rm.(+ 12')	None (Note 2)	Available – Not actively maintained
• Lube Oil Reservoir	Reactor Feed Pump Rm.(+ 12')	2.3.8	Available – Not actively maintained
• Turning Gear Oil Pump	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In place
• Auxiliary Oil Pump	Reactor Feed Pump Rm.(+ 12')	2.3.8	Available – Not actively maintained
• Main Oil Pump	Turbine Enc. Turbine Shaft	2.3.8 (Note 1)	Abandoned – In place
• Reactor Feed Pump Reservoir	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In place
• Reactor Feed Pump Aux. Lube Pump	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In place
• Reactor Feed Pump Lube Pump	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In place
• Reactor Feed Pump Lube Cooler	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In Place
• Reactor Feed Pump Oil Filter	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In place
• Reactor Feed Pump Emergency D.C. Oil Pump	Reactor Feed Pump Rm.(+ 12')	None	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
8. Turbine Plant Systems (cont.)			
8. (c) Condensate System			
• Main Condenser	Condenser Rm. (+6')	Appendix G	Abandoned – In place
• Condensate Storage Tank	Yard	2.3.8 (Note 1)	Abandoned – In place
• Condensate Pumps (2)	Condensate Pump Room (+ 12')	Appendix G	Abandoned – In place
• Condensate Demineralizer (2)	Condensate Demin. Room (+ 12')	Appendix G	Abandoned – In place (The third condensate demineralizer was moved into the sfp demineralizer system and is used as the sfp demineralizer.)
• Contaminated Drain Tank	Condenser Rm. (+6')	None	Abandoned – In place
• Contaminated Drain Tank Pump	Condenser Rm. (+6')	None	Abandoned – In place
• Condenser Vacuum Pump	Condensate Pump Rm.(+ 12')	Appendix G	Abandoned – In place
• Air Ejector	Air Ejector Room (+ 12')	Appendix G	Abandoned – In place
• Inter-after Condenser (Air Ejector Condenser)	Air Ejector Room (+ 12')	Appendix G	Abandoned – In place
• Resin Mixing Tank for SFP Demineralizer (Formerly Cation Regeneration Tank)	Condensate Demin.Rm(+ 12')	Appendix G	Abandoned – In place
• Anion Regeneration Tank	Condensate Demin.Rm(+ 12')	Appendix G	Abandoned – In place
• Regenerated Resin Storage Tank (Laundry Hold Tank)	Condensate Demin.Rm(+ 12')	Appendix G	Abandoned – In place

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
8. Turbine Plant Systems (cont.)			
8.(d) Gland Seal System		Appendix G	
• Gland Seal Condenser	Air Ejector Room (+ 12')	Appendix G	Abandoned – In place
• Gland Seal Exhausters	Air Ejector Room (+ 12')	Appendix G	Abandoned – In place
8.(e) Feedwater System		Appendix G	
• Feedwater Pumps	Reactor Feed Pump Rm (+ 12')	Appendix G	Abandoned – In place
• LP Feedwater Heater	Pipe Tunnel (+0')	None	Abandoned – In place
• IP Feedwater Heater	Pipe Tunnel (+6')	None	Abandoned – In place
8.(f) Closed Cooling Water System		Appendix G	
• Return Tank	Yard, below grade	Appendix G	Abandoned – In place
• Pumps	Yard	Appendix G	Abandoned – In place
• Heat Exchangers	Yard	Appendix G	Abandoned – In place
8.(g) Main Circulating Water System		Appendix G	
• Intake Structure	Intake Canal	Appendix G	Abandoned – In place
• Traveling Screens	Intake Structure	Appendix G	Removed
• Screen Wash Pumps	Intake Structure	Appendix G	Removed
• Circulating Water Pumps	Intake Structure	Appendix G	Removed
• Sluice Gates	Intake Structure	Appendix G	Removed
• Intake Piping	Yard, below grade	Appendix G	Abandoned – In place
• Discharge Piping	Yard, below grade	Appendix G	Abandoned – In place
• Sluice Gate Operators	Intake Structure	Appendix G	Removed

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
8. Turbine Plant Systems (cont.)			
8.(h) Compressed Air System		2.3.8	
• Air Compressor	Reactor Feed Pump Rm(+ 12')	2.3.8	Available – Actively maintained
• Aftercooler	Reactor Feed Pump Rm (+ 12')	2.3.8	Available – Actively maintained
• Service Air Receiver	Reactor Feed Pump Rm (+ 12')	2.3.8	Available – Actively maintained
• Instrument Air Receiver	Reactor Feed Pump Rm (+ 12')	2.3.8	Available – Actively maintained
• Instrument Air Dryer	Reactor Feed Pump Rm (+ 12')	2.3.8	Available – Actively maintained
• Instrument Air Filters	Reactor Feed Pump Rm (+ 12')	2.3.8	Available – Actively maintained
8.(i) Demineralized Water System		2.3.8	
• Demineralized Makeup Water Heater	Makeup Demin. Area (+27')	None	Removed
• Demineralized Water Storage Tank	Yard	2.3.8	Available – Actively maintained
• Demineralized Water Pump	Yard	2.3.8	Available – Actively maintained
• Makeup Demineralizer Filter	Makeup Demin. Area (+27')	None	Removed
• Makeup Demineralizer	Makeup Demin. Area (+27')	None	Removed
• Concentrated Caustic Storage Tank	Yard	None	Removed
• Concentrated Acid Storage Tank	Yard	None	Removed
• Acid Pump, Makeup Demin.	Makeup Demin. Area (+27')	None	Removed

TABLE 2-1
(Continued)

SYSTEMS/COMPONENTS	LOCATION	DSAR SECTION	STATUS
8. Turbine Plant Systems (cont.)			
8.(i) Demineralized Water System (continued)		2.3.8	
• Caustic Pump, Makeup Demin.	Makeup Demin. Area (+27')	None	Removed
• Caustic Dilution Water Heater	Makeup Demin. Area (+27')	None	Removed
• Filter Aid Tank	Makeup Demin. Area (+27')	None	Removed
• Filter Aid Pump	Makeup Demin. Area (+27')	None	Removed

Note 1: Some components are listed as "Abandoned in Place," but are also identified as being described in the body of the DSAR. These components are listed in the DSAR as part of a larger system and are generally not described in detail.

Note 2: Several components are labeled "None" under the DSAR SECTION column. Most of these components have been abandoned or removed completely. Some of the components are listed as being available, but are minor components of a larger system or subsystem and do not merit a description in the DSAR.

Table 2-2

COMBUSTIBLE FUEL STORAGE FACILITIES

	FUEL	MAXIMUM CAPACITY (gals.)	STORAGE METHOD	LOCATION* (ft.)
1.	Residual fuel oil (Number 6 fuel oil or Bunker C)	5,760,678	Tanks	559
2.	Diesel storage tank (Number 2 diesel oil)	84,940	Tank	473
3.	Diesel day tanks	19,800	Tanks	401
4.	Propane	2,098	Tank	229
5.	Gasoline	120	Portable tank	321

EPA restrictions limit HBPP to less than one million gallons of petroleum products on site. All of the fuels are delivered to the plant site by tank trucks.

* Locations reflect the distance from the center of the reactor to the center of the closest tank.

TABLE 2-3

REFUELING BUILDING PENETRATIONS FOR PERSONNEL AND EQUIPMENT ENTRY AND FOR VENTILATION SYSTEMS

Penetration[a]	No.	Size	Type of Closure
Railroad door entry	1	13 ft 11 in wide x 13 ft 6 in high (clear opening)	Normally locked entry
Personnel air locks	2	25-1/2 in. x 66 in.	Two mechanically interlocked doors in series
Caisson personnel emergency exit shaft	1	3 ft dia	Hatch and doors in series[b]
Main ventilation system supply duct	1	16 in. dia	One 16-in. air-operated isolation valve
Main ventilation system exhaust ducts	2	16 in. dia	Two 16-in. air-operated isolation valves[c]
Dry well purge system exhaust duct	1	14 in. dia	Two 8-in. air-operated isolation valves

a In addition to the ventilation system penetrations listed above, two ducts associated with the gas treatment system are considered building penetrations. Valves in the ducts shall normally be closed, but shall be opened when the refueling building ventilation exhaust is through the gas treatment system.

b Hatch is locked and cannot be opened from outside the exit shaft except with key. The lock will not prevent emergency exit from the caisson.

c Two 16-inch exhaust ducts from the refueling building join together to form a single duct. The two isolation valves are in series in this duct.

TABLE 2-4

APPROXIMATE DISTANCE TO SITE BOUNDARY FROM VENTILATION EXHAUST STACK (16 COMPASS DIRECTIONS)

DIRECTION	DEGREES	DISTANCE
North	0°	405'
North - Northeast	22 ½°	560'
North - East	45°	765'
North - West	315°	380'
North - Northwest	337 ½°	360'
East - Northeast	67 ½°	1350'
East	90°	925'
East - Southeast	112 ½°	780'
South - East	135°	785'
South - Southeast	157 ½°	930'
South	180°	1390'
South - Southwest	202 ½°	1415'
South - West	225°	1105'
West - Southwest	247 ½°	1080'
West	270°	770'
West - Northwest	292 ½°	485'

3.0 RADIATION PROTECTION

At the outset of the SAFSTOR period, conditions were established and activities continued to protect the health and safety of the plant workers and the public. This section describes the establishment of the following:

1. A benchmark (baseline) radiation survey to document quantitative initial radiological conditions at the outset of the SAFSTOR period.

2. A radiological characterization of the facility.

3. The surveillance and monitoring program by which assurance can be provided that conditions are not deteriorating.

4. Radioactive waste processing and disposal.

5. Health Physics including the ALARA program and the Radiation Protection Program

3.1 BASELINE RADIATION SURVEY

The Baseline Radiation Survey establishes the activity levels and nuclide concentrations in the plant and its environs at the beginning of SAFSTOR. The survey includes:

- Area and contact dose rates; beta-gamma and alpha by elevation and room
- Proportion of loose versus fixed contamination by elevation and room
- Surveys of system components
- Inventory of nuclides associated with each plant system
- Radionuclide concentrations (if any) in systems which have cross-connections to Units 1 and 2
- Onsite waste inventory
- Activity levels in the Unit 3 area
- Activity levels in the Unit 3 area surrounding the Unit 3 restricted area

- Radionuclide concentrations in:

    - Vegetation
    - Soil (surface and cores around the Refueling Building)
    - Canal sediments and slough sediments
    - Bay sediments
    - Bay mussels
    - Bay algae

Baseline conditions for soil, biota, and sediments were established prior to SAFSTOR and will only need to be reestablished prior to DECON if a significant release occurs during SAFSTOR.

Baseline conditions in the plant will be compared with surveillance values obtained from routine quarterly monitoring. Waste inventory will be updated and documented for 10 CFR 61 disposal. The Baseline Radiation Study is included as Appendix D.

3.2 RADIOLOGICAL CHARACTERIZATION

3.2.1 RADIONUCLIDE INVENTORY

The largest percentage of the onsite radionuclide inventory is contained in the spent fuel, with the reactor vessel and internals containing the next largest percentage. Radionuclides are also present in corrosion films within various in-plant systems.

These radionuclide sources are not readily dispersible in their present condition and will continue to decay during the SAFSTOR period.

Additional contributions to the radionuclide inventory are those sources external to the closed systems addressed above. These sources include fixed and removable surface contamination, the radionuclides contained in the spent fuel pool water, and associated systems.

As of July 1984, the largest inventory of radionuclides is contained in the spent fuel rods with an estimated activity of 1.2E+6 Ci. Within 30 years, this activity will be less than 5.0E+5 Ci due to the decay of ^{137}Cs and ^{90}Sr with half-lives of approximately 30 years.

The largest radionuclide inventory outside of the spent fuel pool consists of activation products in the reactor vessel and surrounding structures. The total activity is estimated at 12,000 Ci. The primary radionuclide is ^{60}Co with an activity of 7,100 Ci (61 percent of the total.) During the SAFSTOR period the ^{60}Co activity will decrease to 110 Ci, and ^{63}Ni with an activity of 2,700 Ci will be the most abundant radionuclide in the reactor vessel system. Since the primary source of exposure is currently from ^{60}Co, the exposure rate will continue to decrease during the SAFSTOR period.

Other than several sealed sources and surface contamination, the remaining inventory is estimated at 100 Ci contained in corrosion films in various piping and components. The primary radionuclides are ^{55}Fe (81 percent) and ^{60}Co (14 percent). After 30 years, the primary radionuclides will be ^{63}Ni and ^{137}Cs, which will comprise 90 percent of the estimated remaining 2.5 Ci. The transuranic radionuclide inventory is estimated at 32 mCi (0.04 percent). The most abundant transuranic nuclide is ^{241}Am, which comprises 38 percent of the corrosion film transuranic inventory.

Refer to Appendix E for the Radiological Characterization and associated tables.

3.3 MONITORING AND SURVEILLANCE

Section 6.2.1 of the Environmental Report estimates annual gaseous emissions and liquid discharges of radioactivity based on average emissions and discharges for the period 1980-1983. A breakdown by radionuclide is contained in the Annual Effluent Reports submitted to the NRC. This breakdown is summarized in the Humboldt Bay Power Plant (HBPP) Environmental Report, Tables 10.4.3 and 10.4.4.

During the SAFSTOR period, the primary activity will be associated with the operation and maintenance of the spent fuel storage pool and the processing of wastes resulting from the spent fuel pool. This activity is expected to account for the majority of the releases during the SAFSTOR period.

3.3.1 IN-PLANT MONITORING

Routine surveys will be conducted at pre-established locations using portable beta-gamma and alpha dose rate meters. Surveys will include area readings and contact readings at several locations such as step-off pads, storage areas, and maintenance workstations.

Samples of the following will be periodically taken and analysis will include total gamma, beta, and alpha activity, and concentrations of indicator nuclides:

- Spent Fuel Storage Pool Water
- Radwaste Liquids, including all discharges from the Radwaste Treatment System
- Solid waste in compliance with 10 CFR 61.

Filters and resins will be replaced as required to minimize concentrations of nuclides in spent fuel storage pool water and radwaste discharge.

3.3.2 ONSITE ENVIRONMENTAL MONITORING

The following monitors will be maintained through the SAFSTOR period:

- Stack radiation monitor
- Continuous sampler in discharge canal
- Fenceline dosimetry stations
- Groundwater monitoring wells

Annual reports will be submitted in accordance with the Offsite Dose Calculation Manual (ODCM) requirements.

In the area of radwaste treatment buildings, routine surveys will be conducted to identify contamination and record dose rates. These surveys will be conducted quarterly or as needed to support waste management operations.

3.3.3 OFFSITE ENVIRONMENTAL MONITORING

Four monitors will be maintained offsite, Stations 1, 2, 14 and 25 (ODCM Figure 2-3). These represent a gradient downwind of the prevailing wind direction. These stations will be equipped with dosimeters and annual reports will include both average and maximum recorded values.

No additional offsite monitoring will be required during the SAFSTOR period unless significant releases occur as a result of an accident.

3.4 RADIOACTIVE WASTE PROCESSING AND DISPOSAL

3.4.1 SOURCES OF RADIOACTIVE WASTES

Wastes remaining on site at the start of the SAFSTOR period activity included liquids and sludges stored in several collection tanks located in Unit 3 and in waste receiving and storage tanks located in the radwaste building. Solid wastes consisted of drums and metal boxes located in various waste storage areas of the plant. Other sources of solid waste on site included contaminated tools and equipment, lumber, and soil.

Waste generated during initial SAFSTOR activities included liquids and sludges resulting from decontamination activities and sludge from final cleanout of tanks and surface decontamination cleaning solutions. Processing these liquids as well as those remaining on-site generated a variety of ion exchange resins and concentrates that were solidified and transported to a shallow land burial facility. The generation of solid wastes such as contaminated protective clothing and tools, along with other typical dry radioactive wastes,

also occurred during initial activities. During the SAFSTOR period, wastes will continue to be generated. Spent fuel storage pool water, rain and groundwater in-leakage will be collected and processed on a routine basis. Specific operational improvements or maintenance projects in conjunction with routine maintenance requirements will result in the generation of liquid and solid waste.

3.4.2 WASTE PROCESSING AND DISPOSAL

During SAFSTOR activities, radioactive wastes generated will be processed on or off site and shipped to a licensed burial site for disposal. Off-site secondary processors may be used as appropriate to sort, survey, decontaminate, free-release, and consolidate wastes. The radioactive waste treatment facility will remain operational throughout the SAFSTOR period.

Liquid radioactive wastes released from the site may be processed by filtration, and/or demineralization, and/or other appropriate methods when treatment is required. Samples of liquid wastes are analyzed before release to ensure that they are within the discharge limits specified in 10 CFR Part 20.

The only release point for liquid radioactive waste is the liquid radioactive waste discharge line that discharges into either the Unit 1 or Unit 2 circulating water discharge line prior to reaching the plant discharge canal.

The expected sources of liquid radwaste from Unit 3 include: spent fuel pool liner leakage; spent fuel pool recirculation pump packing leakage; resin sluice water; wastewater from ongoing decontamination efforts; hot lab waste; caisson inleakage; and rainwater runoff from contaminated areas. Prior to release into the plant discharge canal, the liquid radwaste will be diluted with the liquid effluent from the Unit 1 and/or Unit 2 circulating water pumps. The discharge from each of these four pumps is expected to be no less than 12,500 gpm. Since Units 1 and 2 are expected to be in service for the duration of the SAFSTOR period, and normal operations requires at least one unit operating, the typical flow rate (two circulators) will usually be at least 25,000 gpm.

Liquid radioactive wastes that must be treated before discharge may be treated by vendor (contractor) systems on site if filtration or demineralization is not adequate. Concentrated liquid radioactive waste will be stored in the concentrated waste tanks. Processing of liquid radioactive wastes and wet solid (sludge) wastes will be in accordance with the plant or vendor procedures and in accordance with current regulations. Liquid radioactive wastes and wet solid wastes may be shipped to secondary processors for final treatment before disposal.

Chemical and liquid decontamination wastes generated during SAFSTOR will be treated with other liquid radioactive waste.

Spent resins from the radwaste demineralizer and the spent fuel storage pool demineralizer are also accumulated on site in the resin storage tank. When a sufficient quantity of resins has accumulated, it will either be dewatered and shipped or solidified and shipped to a licensed burial site in accordance with applicable regulations. An off-site secondary processor may be used for volume reduction or further processing prior to disposal.

Spent cartridge-type filters (and filter crud) will be packaged, processed and shipped in accordance with applicable regulations.

Activated components will be handled via current technology in accordance with applicable regulations.

Solid radioactive wastes include: contaminated protective clothing, plastic, rags, piping, surplus equipment, contaminated soil, rubble, etc. Solid wastes are accumulated and stored in appropriate containers. Radioactive waste containers are inventoried, marked, and stored on site until a sufficient quantity has been accumulated to ship the wastes off site for processing and disposal. Waste processing (e.g. sorting, decontamination, compaction) may occur on-site. Off-site secondary processors may be used for processing and volume reduction before disposal.

Low level liquid and solid wastes will be sampled and analyzed as required by regulation when they are packaged or prior to processing for shipment to a secondary processor or disposal site. Analysis will be conducted using a combination of onsite gamma spectrometry, offsite laboratory analysis, and the development of standard plant mixtures for similar wastes that can then be ratioed based on a significant isotope or dose rate. The results of the analysis will determine waste classification in accordance with 10 CFR Part 61, the disposal site license, and any other regulatory requirements in effect at the time. Regulatory guidelines, such as NRC Branch Technical Position, will also be used to characterize wastes.

Records of samples and analysis will be retained to demonstrate the basis for waste classification and stability requirements.

Disposal of processed and packaged radioactive wastes will be accomplished by shipping the wastes to an authorized secondary processor or shallow land burial facility. Shipments will normally be made by truck in accordance with Department of Transportation regulations contained in 49 CFR Parts 171-179. Low-level wastes shipped for land burial disposal will be characterized in accordance with and meet the waste form requirements in 10 CFR Part 61.

3.4.3 DECONTAMINATION

3.4.3.1 Purpose

During the SAFSTOR activities an ongoing program of facility decontamination will continue. The purposes of this program are:

- To minimize contamination levels and radiation dose rates in areas of Unit 3 that will be accessible during the SAFSTOR period for routine maintenance or for periodic surveys.
- To minimize the requirements for surveys to detect the spread of contamination
- To minimize the requirements for protective clothing and the potential for contamination of tools and equipment. This will reduce maintenance costs during SAFSTOR by reducing the amount of radioactive waste generated.

- To reduce the potential for contamination to spread outside of controlled areas

Decontamination efforts will be concentrated in areas to which access will be available during SAFSTOR. Areas to which access will not be routinely permitted without special authorization are sealed to minimize spread of contamination from the areas and secured with locked gates or physical barriers to prevent access.

3.4.3.2 Decontamination Methods

The decontamination program during the SAFSTOR period will be a continuation of the decontamination work that is routinely performed at Unit 3. The decontamination method utilized is dependent on the level of contamination encountered, the type of surface, type of radioactivity involved, and whether or not the contamination is fixed or loose.

The methods listed below are considered to be representative. Although others may exist, they are not included because of their similarity to one of the listed methods or because of some characteristic (such as extreme toxicity) that may render them unsuitable.

Hand Wiping. Rags moistened with water or a solvent such as acetone can be an effective decontamination process. This method may not work well if the item is rusty or pitted. It requires access to all surfaces to be cleaned, is a relatively slow procedure and its hands-on nature can lead to high personnel exposure. On the positive side, wiping can provide a high decontamination factor (DF), generates easily handled decontamination wastes (contaminated rags), requires no special equipment, and can be used selectively on portions of the component. For these reasons, hand wiping will normally not be used for decontamination to reduce levels for shipping where high dose rates and inaccessible internal contamination usually exist. For purposes of decontamination to allow the item to be its own shipping container, the generally large component size will restrict hand wiping to small areas that are discovered to be above shipping limits. Wiping can be used extensively and effectively on smaller items with low-to-medium external contamination levels and easily accessible internal contamination.

Steam Cleaning. This may be performed either remotely in a spray booth or directly by decontamination personnel using some type of hand-held wand arrangement. In the former case, only minimal internal decontamination is possible; however, reasonable external cleaning can be accomplished quickly with low exposure expenditures. Containment of the generated wastes and protection of personnel from radioactive contamination become more difficult.

Abrasive Blasting. This is a highly effective procedure that can effect total decontamination of even rusty or pitted surfaces. As with hand-held steam cleaning, this method suffers from internal accessibility problems. It also generates large amounts of solid wastes and, being a dry process, produces significant quantities of airborne radioactivity. Abrasive blasting may be used if its high effectiveness can be justified after taking the exposure, waste, and accessibility limitations into account.

Hydrolasing. The use of high-pressure water jets for decontamination falls somewhere between steam cleaning and abrasive blasting in effectiveness. Less effective than

abrasive blasting, it has the advantage of producing liquid wastes (that can be processed) rather than solid wastes. As an external cleaning technique, it offers reduced airborne generation potential although this is offset by the need to control splashing. The utility of hydrolasing is generally limited to operations where internal accessibility is not required.

Ultrasonic Cleaning. Since this is basically an immersion process and, as such, is limited to smaller items, it is generally unsuitable for large-scale decontamination. In addition, although ultrasonic cleaning can be especially effective in removing contamination from crevices, it is doubtful that releasable levels can be reached consistently enough with this technique to make it a viable option. Therefore, this method, if employed, would be used mostly for decontamination of smaller, highly contaminated components that have crevices or poor internal accessibility.

Electropolishing. This is an electrochemical process where the object to be decontaminated serves as the anode in an electrolytic cell and radioactive contamination on the item is removed by anodic dissolution of the surface material. Although it is a relatively new process and has not yet been used for a full-scale decontamination operation, it nevertheless requires consideration as a technique on the basis of its superior effectiveness in cleaning almost any metallic surface to a completely contamination-free state. On the other hand, this process has several drawbacks including the cost of electrolyte and special equipment, the consumption of considerable power, and the production of highly radioactive solutions.

Impactors. Two surface removal methods used more extensively than the rest are jack hammers and impactors. Jack hammers, powered by compressed air, are readily available and are easily operated by one man. They are used to chip off the surface material deep enough to remove the contamination. Because they are difficult to position on walls and ceilings, jack hammers are used primarily on floors. Impactors (or hoe rams), similar in operation to jack hammers but much larger, have been used successfully in several decontamination projects. An impactor uses a pick chisel point that is driven into the concrete surface with high-energy impacts several times per second. Impactors are powered either pneumatically or hydraulically, and are held and positioned with linkages typical of those on tractor-mounted backhoes. A medium-size air filtration system is necessary to control the dust produced by both of these surface removal methods.

Chemical Decontamination. This technique uses concentrated or dilute solvents in contact with the contaminated item to dissolve either the contamination film covering the base metal or the base metal itself. Dissolution of the film is intended to be nondestructive to the base metal, and is generally used for operating facilities. Dissolution of the base metal should be considered only in a decommissioning program where reuse of the item will never occur.

Chemical flushing is recommended for remote decontamination of intact piping systems. Chemical decontamination has also proven to be effective as an alternative to partial or complete removal in reducing the radioactivity of large surface areas such as floors and walls.

3.4.3.3 System Internal Flushing

Secured plant systems that contained liquids have been flushed (where practical) to remove loose contaminants and the systems drained. Systems/components, which contribute radiation exposures significantly above the general area dose rates, have been reviewed to determine action warranted as a result of ALARA considerations. Systems/components to which routine access will not be required during the SAFSTOR period are isolated using locked barriers. In these cases system internal decontamination was not warranted. Systems/components to which or near which routine access will be required during the SAFSTOR period have been evaluated on a case-by-case basis. In some cases, a portion of the system piping or component has been removed. Where such action was not practical, installation of shielding or internal decontamination of the system or component was performed to reduce radiation levels to the minimum practical.

3.5 HEALTH PHYSICS

During the SAFSTOR period, radiation protection and health physics programs will be provided to ensure the health and safety of Unit 3 workers. The programs also provide the necessary monitoring and control of radiological conditions to protect the health and safety of the general public and to ensure compliance with Unit 3 license requirements. In addition, programs will be provided to maintain radiation exposures as low as reasonably achievable (ALARA).

3.5.1 ORGANIZATION AND RESPONSIBILITIES

The organization described below is the organization, as it exists during the SAFSTOR period. The organization may be changed during the SAFSTOR period as staffing levels or work requirements dictate. Responsibilities assigned to a position, which is deleted, will be assigned to another position in order to maintain continuity.

The HBPP Plant Manager has the overall responsibility for all onsite activities, including assurance that corporate ALARA policies are carried out at the plant. The Plant Manager is the Chairman of the HBPP Plant Staff Review Committee (PSRC), which also serves as the ALARA Committee.

The Radiation Protection Manager is designated as the on-site manager responsible for implementing the radiation protection and ALARA programs. The Radiation Protection Manager serves as a member of the PSRC (refer to Unit 3 Quality Assurance Plan). He has the authority and responsibility to halt operations he deems to be unsafe and to report the matter to the Plant Manager; and communicate his concerns directly to any level of Nuclear Power Generation Department management, including the Senior Vice President, Generation and Chief Nuclear Officer, if he deems it to be appropriate.

Radiation and Process Monitors are the employees who perform chemical and radiological sampling analyses and radiation and contamination surveys. In addition, they implement

the personnel radiation monitoring program, maintain radiation protection records, and provide monitoring for work in radiologically controlled areas. Three such positions will normally be filled during SAFSTOR.

Plant staff qualifications are discussed in section 4.1.3 of the DSAR.

3.5.2 ALARA PROGRAM

It is the policy of PG&E to design, operate, and maintain its nuclear power plants in such a manner as to maintain personnel radiation Total Effective Dose Equivalent (TEDE) ALARA. The TEDE ALARA concept is implemented by assuring that every effort be made by all HBPP personnel involved in the planning or performance of radiation work to maintain individual exposures to radiation sources or materials as far below the occupational dose limits as is reasonably achievable, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest. The Company's commitment to maintaining TEDE ALARA involves:

- Design - planning, reviews, system, subsystem, and component selection and location; operator usage considerations and maintainability
- Construction - procedures, planning, methods, testing, and scheduling
- Operation - procedures, license compliance, techniques, equipment usage, maintenance, and operating experience feedback from company and industry experience
- Personnel - training, management support, motivation, and supervision
- Administration - policy, guidance, controls, licensing position, and documentation
- Management - involvement, commitment, supervision and oversight

The HBPP PSRC also functions as the plant ALARA Committee.

The committee meets quarterly or as called for by the chairman or the Radiation Protection Manager and has the following functions and responsibilities:

- Review radiation exposures associated with routine operations and maintenance and recommend future exposure reduction goals
- Review planned jobs where potential exposures might exceed 500 person-mRem for the job and establish exposure limits and person-rem goals for that job
- Review completed jobs for achievement of goals and future improvements
- Review plant radiation and contamination levels annually and recommend future exposure reduction goals
- Review plant design changes and plant procedures for ALARA considerations (when applicable)

Before the ALARA committee review of a proposed job, the individuals planning the job make estimates of the expected radiation exposures. Estimates are based on radiation surveys conducted in the area where the job will be performed and estimates of the time required to perform the job based on prior experience. These estimates are reviewed by the Radiation Protection Department. If the established review threshold of 500 person-mRem for the total job is expected to be exceeded, an ALARA review checklist is completed for review by the ALARA Committee. The purpose of the checklist is to document the consideration of specific actions that may be taken to reduce radiation exposures.

All radiation workers at HBPP receive as part of their radiation protection training, an indoctrination in the principles of ALARA radiation exposure control. In this training, the responsibility of the individual worker to follow procedures and safety rules and to maintain his/her own exposure ALARA, are emphasized. The principles of minimizing the duration of exposure (time), maintaining distance from the source (distance) and reducing the source term (shielding) are included in the training.

3.5.3 RADIATION PROTECTION PROGRAM

All employees who routinely work in the restricted areas of the plant, and transient workers whose work may involve significant radiation exposure, will participate in the radiation protection program. Radiation protection training will be commensurate with an individual's work requirements and the areas to which they are permitted access. Nonradiation workers (i.e., persons whose work rarely, if ever, requires they enter a restricted area and/or who receive minimal radiation exposure) shall be provided with fundamental training on radiation, health effects, and risks as appropriate.

Members of the Radiation Protection Department and Certified Fuel Handlers are responsible for implementing the requirements of the Radiation Protection Program. These individuals, as part of their initial qualification, will receive additional training in radiological work practices and the use of specialized survey and analysis equipment to the extent necessary to perform their duties.

The radiation protection program that has been implemented for the SAFSTOR period is an extension of the program that was in effect during operation of Unit 3. The radiation protection program shall be organized to meet the requirements of 10 CFR 20. Radiation protection procedures shall be prepared, approved, adhered to, and made available to all plant personnel. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. Detailed procedures implement the program at the plant level. The following sections describe the radiation protection program.

3.5.3.1 Personnel Monitoring

Personnel entering the Unit 3 Restricted Area (area to which access requires written authorization for the purposes of radiation protection) wear personnel monitoring thermo-luminescent dosimeters (TLDs), if they are required to be monitored for occupational radiation exposure in accordance with 10 CFR 20. The TLDs provide a record of radiation exposure received and are normally worn on the upper portion of the body unless the nature of the work or the principle source of radiation is such that another portion of the whole body is likely to receive a larger dose, in which case the TLD will be worn on that

portion of the body. Additional TLDs and/or finger rings are used to monitor extremity doses for jobs in which significant extremity doses are expected.

Direct reading pocket dosimeters or similar devices are used to provide estimated exposure. TLDs provide official exposure. The TLDs are evaluated quarterly or whenever exposure estimates indicate exposures approaching an exposure limit.

Other personnel monitoring instruments consist of personnel contamination monitor(s) and count rate meters. These instruments are used to conduct personnel contamination surveys.

Internal radiation exposure monitoring is performed through a program of evaluating lapel and routine air sample results. Whole body counting is used to:

1. Verify the results of the air sampling.
2. Screen incoming workers and,
3. In the event of suspected internal contamination.

Personnel who are regularly assigned to work in the Unit 3 restricted area and who are likely to receive 500 mRem of exposure per year are routinely given one whole body count per year. Persons terminating work assignments are given whole body counts if there is reason to suspect significant intakes of radioactive material. Other types of bioassay testing (urine, fecal, swabs, etc.) may be used if circumstances warrant.

The Humboldt Bay Power Plant Radiation Control Procedures for performing bioassays conform to the requirements of 10 CFR 20.

3.5.3.2 Airborne Radioactivity

The potential for internal radiation exposure is minimized through the use of engineering controls and through the use of a respiratory protection program. Air samples are taken to monitor airborne radioactivity.

The spread of airborne radioactivity within the facility is minimized by maintaining airflows from areas of low potential airborne radioactivity to areas of higher potential whenever possible. Other engineering controls such as temporary containments, encapsulating contamination, controlled ventilation techniques, and/or HEPA air filters are utilized when practical.

Areas with a potential for existing airborne radioactivity are evaluated prior to the start of work. Air samples are taken during activities that may produce airborne radioactivity. In addition, the Refueling Building +12 feet elevation is continuously monitored to detect airborne radioactivity. Samples are evaluated to identify types and sources of radioactivity. Airborne Radioactivity Areas are posted to prevent unauthorized entry. Notices to inform personnel of unusual radiation, contamination, or airborne conditions will be posted at Access Control(s). The use of respiratory protection equipment is specified by the radiation protection department.

3.5.3.3 Respiratory Protection Program

A respiratory protection program will be maintained during the SAFSTOR period. Use of respiratory protection equipment will be based on a TEDE ALARA review. Whenever practical, engineering controls will be used to maintain airborne concentrations not only below specified limits but as low as reasonably achievable.

Personnel assigned to use respiratory protection equipment receive a physical examination to qualify them to wear respirators. In addition, respirator users receive training in the proper use of respirators and are fitted to ensure that they can achieve a proper respirator seal. Only NIOSH-approved respiratory protective equipment is used.

Air-purifying respirators utilize NIOSH-approved filters capable of removing particulate radioactivity. Air-supplied respirators or self-contained respirators may also be used when appropriate. Plant procedures are provided for the fitting, issue, and maintenance of respiratory protection equipment.

The respiratory protection program is designed to comply with the provisions of NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

3.5.3.4 Protective Clothing

Personnel working in contaminated portions of the Unit 3 restricted area are provided with protective clothing to minimize the potential for personnel contamination. Protective clothing requirements for specific jobs are specified by the Radiation Protection Department as part of the work authorization for that job. Protective clothing such as coveralls, lab coats, surgeon's caps, hoods, shoe covers, gloves, etc., is readily available for routine use in the controlled area.

Additional items of protective clothing such as waterproof clothing, face shields, etc., are issued when required for a specific job. Contamination control points ("step-off pads") are established at certain locations in the plant to permit changing of protective clothing to prevent the spread of contamination. Clothing potentially contaminated to levels in excess of that permitted in certain areas are either removed or exchanged for cleaner items at the step-off pads.

During previous activities in preparation for SAFSTOR, the laundry facility was operational to ensure an adequate supply of clean protective clothing. Subsequent to the entry into SAFSTOR, the laundry has been secured and protective clothing used will be either disposable or will be shipped off site for cleaning.

3.5.3.5 Control of Access

To limit radiation exposures, personnel access is controlled in areas where such exposure is possible. This control consists of a system of physical barriers, warning signs and signals, and administrative procedures which govern authorized entries. Written authorization for all entries into the restricted area is required. This written authorization is either in the form of a Routine Work Permit (RWP) or a Special Work Permit (SWP).

RWPs are established to authorize work of a routine nature in the restricted area under relatively stable conditions. RWPs are normally effective for an extended period of time (up to a maximum of 2 years) but are subject to revision at any time.

SWPs are short-term authorizations for personnel to perform work of a non-routine nature in specific areas. All entries into the restricted area not covered by a valid RWP must be covered by an SWP.

Both RWPs and SWPs contain a description of the radiological conditions existing in the area covered by the permit, general and special instructions to be followed by persons working under the permit, and a list of the protective equipment requirements.

3.5.3.6 Facilities Monitoring

A program for routine surveys and monitoring will be continued during the SAFSTOR period. Radiological surveys will be used to maintain a record of radiological conditions in Unit 3 and to evaluate radiological trends during decommissioning activities.

Radiological surveys include measuring general area dose rates (radiation surveys) and the collection and analysis of representative samples of airborne particulates, water, and removable surface contaminants.

Air Samples. Airborne radioactivity surveillance is conducted to detect airborne radioactivity to which personnel may be exposed, to detect equipment degradation or failure, and to limit airborne releases from the plant to permissible amounts. Routine air particulate samples are obtained from a sampler located in the Refueling Building. These samples are counted for alpha and beta-gamma activities. Particulate samples are obtained from the ventilation exhaust stack weekly to determine release rates, if any.

Non-routine air samples to establish protection requirements for maintenance activities or to verify airborne radioactivity conditions during work activities are obtained and analyzed when routine samples are not sufficient for monitoring plant conditions.

Radiation Surveys. Radiation surveys are conducted for the following purposes:

- Measure and document radiation and contamination levels in areas of interest.
- Identify trends in radiation and contamination levels, particularly during work in progress.
- Determine appropriate protective measures for personnel working in restricted areas.
- Provide information so that workers can maintain their doses TEDE ALARA.
- Identify locations and situations where special dosimetry may be required.

On average, a "B" survey (routine survey) is conducted weekly by Radiation Protection Department personnel. A "B" survey includes an extensive dose rate survey of an area. Contamination levels are determined using smears and appropriate instrumentation. When

alpha contamination is suspected, an alpha scintillation meter is used. Results are recorded on survey maps or SWP/RWP job history sheets and maintained for reference. Schedules have been established for surveys to ensure that all plant areas are surveyed periodically. The frequency for "B" surveys of a specific area is determined by the level of work activity in that area and the radiation levels existing in the area. Surveys of high radiation areas are performed prior to entry rather than on a routine schedule.

Special radiation surveys of particular items or areas are performed on an "as needed" basis. Examples of special radiation surveys are the removal of equipment or materials from a restricted area, leak testing of sealed radioactive sources, or the shipment or receipt of radioactive material packages.

Water Samples. Samples of water containing radioactivity are collected and analyzed on a routine basis. Spent fuel pool water is analyzed to detect indications of degradation of the fuel stored in the pool. Samples of liquid radioactive wastes and processed wastes are analyzed to ensure levels of radioactivity are below the levels permitted for release. Samples are analyzed by Radiation Protection Department personnel and/or offsite Laboratories, (as applicable) in accordance with established procedures.

3.5.3.7 Radiation Protection Equipment and Instrumentation

A variety of equipment and instruments are used as part of the radiation protection program. Equipment and instrumentation are selected to perform a particular function. Sensitivity, ease of operation and maintenance, and reliability are factors that are considered in the selection of a particular instrument. As the technology of radiation detection instrumentation improves, new instruments are obtained to more accurately measure radioactivity and ensure an effective radiation protection program.

A suitable number of appropriate radiation detection instruments will be available to perform required radiation and contamination surveys to meet the requirements of the applicable federal regulations. In addition to the area monitoring system, suitable portable and fixed, alpha and beta-gamma detection instruments and beta-gamma dose rate instruments shall be provided for use of personnel entering the Unit 3 radiation areas and radioactive materials areas and for analyzing samples.

Portable Instruments. Portable instruments are for radiation surveys to measure dose rates for beta and gamma radiation and to perform contamination surveys.

Portable dose rate survey instruments are source checked prior to use, or daily, and are calibrated at least semi-annually by a vendor or PG&E at an offsite location. Detailed procedures are available describing the operation of the instruments. Plant personnel are trained in their operation as part of the radiation protection training program.

Area Radiation Monitors. The area radiation monitoring system utilizes fixed gamma monitors, with a range of 0.01 to 100 mR/hr, installed at various locations throughout the unit. The outputs of these monitors shall be recorded in the control room. Each channel shall have an adjustable high radiation alarm, which is annunciated. These instruments are calibrated at least quarterly. The instruments are source checked to test their response at least monthly. HBPP has chosen to comply with 10 CFR 50.68(b) in lieu of 10 CFR 70.24,

in part by the use of two area radiation monitors located at the +12 ft. (south wall access door) and +27 ft. (northwest access door) elevations in the refueling building.

A high radiation signal from either of these channels shall provide a "Building Above Normal Radiation" signal, which is annunciated in the control room. These two area monitoring channels shall also provide gamma monitoring of the fuel storage areas. A high radiation level signal from either of these channels shall sound the evacuation horns in the refueling building. See section 2.3.5 for additional information.

Laboratory Instrumentation. Laboratory instrumentation used to analyze radioactive samples includes equipment for the analysis of alpha, beta, or gamma emitting isotopes. A quality control program for laboratory counting equipment is in effect to perform calibrations and calibration checks of the equipment when it is in use. Sealed sources traceable to NIST standards are used for calibration references for this equipment. In addition, the HBPP Radiation Protection Department participates in an inter-laboratory sample splitting program with PG&E's Diablo Canyon Power Plant and an offsite laboratory. The plant has also participated in NRC directed sample splitting programs.

Maintenance of Radiation Protection Instruments. Routine maintenance and calibration of radiation protection instruments is performed by technicians in the HBPP Instrument Maintenance Department. In the case of instruments, which cannot be calibrated by plant personnel, the instruments are sent to an offsite calibration facility. Certification of NIST traceable calibration is required for all instruments included in the calibration program.

3.5.3.8 Radiation Protection Records

The following records related to the radiation protection program are maintained:

- Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant where radiation exposure monitoring is required in accordance with 10 CFR 20.
- Records of plant radiation and contamination surveys.
- Records of radioactivity in liquid and gaseous effluents released to the environment.
- Records of training and qualification of personnel associated with the radiation protection program.
- Records of periodic checks, tests, and/or calibrations of radiation protection instruments and equipment.

4.0 ORGANIZATION AND ADMINISTRATIVE CONTROLS

4.1 PLANT STAFF ORGANIZATION AND RESPONSIBILITIES

4.1.1 SAFSTOR ORGANIZATION

Functions and responsibilities for the SAFSTOR organization are contained in the Quality Assurance Plan.

4.1.2 STAFFING DURING SAFSTOR

Key positions in the plant organization during the SAFSTOR period are described in the HBPP Quality Assurance Plan. During this period, sufficient expertise will be maintained to perform the required maintenance, operations, and surveillance activities for the plant. Contractor assistance will continue to be utilized to perform services beyond the capabilities of the plant staff.

The permanent plant staff for operation and maintenance of the fossil-fueled units and gas turbines, and for SAFSTOR maintenance of HBPP Unit 3 during the SAFSTOR period is estimated to be approximately 100 PG&E employees. Minimum staffing requirements are addressed in various HBPP documents (e.g., Emergency Plan, Security Plan, DSAR Section 5.1.4 and Technical Specification 5.2.2).

Most plant employees currently reside in Humboldt County. The majority reside in Eureka, Arcata, McKinleyville, Fortuna, and Ferndale. The remaining employees live in other smaller communities or unincorporated areas. No traffic congestion or problems of interference with local traffic are expected to occur due to SAFSTOR activities. The staffing levels at the plant will not be significantly increased due to SAFSTOR activities. Waste shipment traffic is expected to be infrequent with periodic increased activity to accommodate certain decommissioning projects.

4.1.3 PLANT STAFF QUALIFICATIONS

The minimum qualifications for members of the plant staff shall be the following. An individual may be assigned to any of these positions without meeting the requirements if a sufficient number of other persons who meet those requirements are assigned to the plant full time to assist the individual until the minimum qualifications are met. Note that for the following positions, nuclear power plant experience includes experience at Humboldt Bay Power Plant (HBPP) during operations or during SAFSTOR:

Plant Manager

- Ten years of responsible power plant experience, of which 1 year shall be nuclear power plant experience. A maximum of 4 years of the remaining 9 years of experience may be fulfilled on a one-for-one time basis by academic training in an engineering or scientific field generally associated with power production.

Supervisor of Operations

- Eight years of responsible power plant experience, of which 1 year shall be nuclear power plant experience. A maximum of 4 years of experience may be fulfilled on a one-for-one time basis by academic training in an engineering or scientific field generally associated with power production.
- Certified Fuel Handler in accordance with the HBPP Fuel Handler Certification Training Program.

Radiation Protection Manager

- Bachelor's degree in engineering or a scientific field including some formal training in radiation protection.
- At least 5 years of professional experience in applied radiation protection. (A master's degree may be considered equivalent to 1 year of professional experience, and a doctorate may be considered equivalent to 2 years of professional experience where course work related to radiation protection is involved.) At least 3 years of this experience should be in applied radiation protection in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations.
- Specialized knowledge of health physics, thorough knowledge of radiation and criticality requirements and practices, and knowledge of related regulatory requirements and practices.

Supervisor of Maintenance

- Seven years of responsible power plant experience or applicable industrial experience, of which a minimum of 1 year shall be nuclear power plant experience. A maximum of 2 years of the remaining 6 years of power plant or industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis.
- Thorough knowledge of safety requirements specifically related to maintenance under radioactive contamination conditions.

Nuclear Quality Services (NQS) Supervisor HBPP

- Five years of responsible power plant experience or applicable industrial experience, of which a minimum of 1 year shall be experience in the nuclear field. A maximum of 2 years of the remaining 4 years of power plant or applicable industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis.
- Thorough knowledge of nuclear materials handling, safeguards, and quality assurance methods and procedures.

Engineering Manager (Technical Manager)

- Bachelor's degree or the equivalent in an engineering or scientific field.
- Four years in responsible positions related to power generation, of which 1 year shall be nuclear industry experience.
- Thorough knowledge of radiation and criticality safety requirements and practices, including safety requirements specifically related to maintenance and operations under radioactive contamination conditions.

Certified Fuel Handler

- High school diploma or equivalent.
- At least 1 year of experience at HBPP.
- Satisfactory completion of HBPP Fuel Handler Certification Training Program.

4.1.4 OFFSITE SUPPORT

Diablo Canyon & General Office Support. Support for HBPP Unit 3 SAFSTOR activities are available from several departments located at the Diablo Canyon Power Plant and PG&E's General Office. Such technical support covers all functions performed at Humboldt.

Three Diablo organizations actually have matrixed staff present on the Humboldt site. These include Quality Assurance, Licensing, and Budget & Performance Management.

The Nuclear Quality Services (NQS) Department continues to be responsible for the HBPP Unit 3 Quality Assurance Plan and for performing audits to verify that SAFSTOR activities are performed in accordance with the DSAR and the HBPP Unit 3 License and Technical Specifications.

The Regulatory Services (RS) Department provides assistance in preparing license amendment submittals and coordinating responses to NRC correspondence.
The Budget & Performance Management Department provides assistance in the preparation of job estimates and budgets during the SAFSTOR period. In addition, the Procurement Services Section of the Site Services Department provides contract

administration services for contracts used to obtain assistance in performing activities during the SAFSTOR period.

4.2 ADMINISTRATION AND CONTROL

4.2.1 COST ESTIMATES AND FINANCING

The project to place HBPP Unit 3 into the custodial SAFSTOR mode cost approximately $14 million (direct costs plus indirect costs and overheads).

During the SAFSTOR period, funds are required to maintain the unit in accordance with the possession-only license and associated technical specifications. Maintenance and surveillance activities during the SAFSTOR period are similar to activities conducted during operation, but reduced in scope. Funds for SAFSTOR costs are allocated in the Humboldt Bay Power Plant annual operations and maintenance budget.

Pacific Gas and Electric Company Application Number 83-09-049, "Authority to Increase its Electric Rates to Reflect Retirement and Decommissioning of Humboldt Bay Power Plant Unit 3", was filed with the Public Utilities Commission of the State of California on September 19, 1983.

In Application No. 83-09-049, PG&E requested rate recovery for the costs associated with the retirement of the plant, including the cost of decommissioning HBPP Unit 3. PG&E requested that the unrecovered capital expenditures for HBPP Unit 3 be included in the electric rate base. PG&E also requested permission to recover through rates the costs associated with placing HBPP Unit 3 into the SAFSTOR condition.

The California Public Utilities Commission (CPUC) has authorized that the HBPP Unit 3 operation and maintenance expenses including certain expenditures for SAFSTOR activities be recovered in rates. The CPUC has granted the Company's request to collect the remaining SAFSTOR costs from PG&E customers. This authorization is expected to continue during the SAFSTOR period.

All electric generating facilities are eventually decommissioned. The estimated costs for the removal of the plants are included as part of depreciation and recovered through rates. Accordingly, the costs for SAFSTOR are part of the costs of removal and will be amortized together with PG&E's unrecovered capital. PG&E requested that the CPUC authorize recovery of the estimated nonrecurring cost of $10 million that PG&E used to place HBPP Unit 3 into SAFSTOR. PG&E proposed, however, that only the actual SAFSTOR costs be recovered. The CPUC reviews actual SAFSTOR costs in PG&E's general rate case and adjusts the amortization rate to incorporate the recorded SAFSTOR costs. An additional $4 million of initial SAFSTOR work was recovered as part of operating and maintenance costs.

The annual, continuing costs of maintaining the unit in SAFSTOR will be included in base rates. These annual costs will be reviewed as part of PG&E's periodic general rate case.

During the SAFSTOR period, plant workers responsible for the maintenance and operation of fossil-fueled Units 1 and 2 will provide the maintenance and surveillance for Unit 3.

The dollar estimate of the taxes attributable to HBPP Unit 3 during the SAFSTOR time period is zero dollars per year. The State Board of Equalization valued HBPP Unit 3 at essentially zero in 1984. The local taxing jurisdiction did not levy a tax on the unit beginning in 1984, and is not expected to levy a property tax on the unit during the SAFSTOR time period.

PG&E proposed that the CPUC establish an external fund, which would qualify under the Tax Reform Act of 1984, for the purpose of accruing the required funds for decommissioning. The cost to decommission was accrued over 5 years, starting in 1987, and the trust fund was determined to be fully funded by 1991. The 5 annual payments to cover the decommissioning costs were invested in an external sinking fund (under the Tax Reform Act of 1984) and managed by an outside party. For the eventual complete decommissioning of Unit 3, PG&E contracted TLG Services, Inc., to prepare a site-specific decommissioning cost estimate in 1997. Information pertaining to the cost estimates for complete decommissioning of Unit 3 is contained in the PSDAR.

4.2.2 PROCUREMENT

For procurement of quality-related services, either: 1) an approved supplier, vendor, or contractor QA Program, or 2) the PG&E QA Program may be used as determined by PG&E. Contracts and contract revisions are reviewed by NQS to verify that the contractor's quality assurance program is adequate.

PG&E standard practices and administrative procedures govern the selection of contractors and the administration of contracts.

4.2.3 TRAINING PROGRAM

4.2.3.1 Training Program Description

PG&E has established general employee training (GET) requirements for PG&E and contractor employees who work in Unit 3. In addition to GET, programs have been designed to assist personnel with technical aspects of their work. Such topics include Hazardous Material (Waste) Program Training and Radiation Protection Technician Training. Additional topics may include such topics as Radioactive Waste Volume Minimization, Contaminated Asbestos Materials, and Decontamination Workers Training.

Personnel who enter Unit 3 for the purpose of conducting work need to have basic knowledge of HBPP and its procedures. Initial training is given prior to any assignment of

work in Unit 3. Training may be accomplished through the use of formalized classroom lecture(s), video/cassette tapes, Computer Based Training, and/or handouts.

The level of training provided to employees is based upon a review of the information employees will require in order to perform their job duties safely and efficiently. Consideration is also given to the employee's past experience and training. The program provides the flexibility for making the decision on a case-by-case basis.

In addition, special training will be provided as needed when it is deemed necessary or prudent to assist employees involved with unusual or infrequent procedures associated with decommissioning activities. Special training relating to decommissioning activities may include such topics as radioactive waste volume minimization, handling of contaminated materials, and decontamination workers training. Employees actively involved with such activities will receive special training appropriate to their job duties and responsibilities as necessary and on a timely basis.

4.2.3.2 General Employee Training

The GET Program provides the flexibility to adapt training depth and scope to personnel needs commensurate with job duties and responsibilities, the areas of HBPP to be entered, and prior experience. Presentations and lessons are supplemented with videotapes, written handouts, and other techniques designed to improve training efficiency.

Description of Plant and Facilities. Three levels of training are available.

Level 1; "Plant Orientation" training is given to all new employees and visitors depending on the scope of work to be performed. The training provides orientation of site restrictions and rules, and includes an overview of the structures, allowable entry areas, escorted access, security areas, inadvertent entry and alarms, and radiation hazards as appropriate.

Level 2; "Power Plant Overviews" training is given to all new permanent plant staff in addition to the Plant Orientation, Level 1 training described above. This training provides more detailed information regarding plant equipment, operation, and organization. Additional activities included in this training are: QA/QC; radiation protection; emergency plan training; security plan training; and fire brigade introduction. If appropriate, material safety and hazardous materials management are also included as part of the Power Plant Overview, Level 2 training.

Level 3; "Detailed Description of Plant Systems," training is detailed technical training reserved for workers engaged in operation, maintenance, or testing of systems. This training also includes any annual retraining and requalification requirements.

General Site Rules. This session discusses plant policies on health and safety, vehicle operation, firearms, liquor and drugs, cameras and other personal items, as well as any disciplinary actions, which may be accorded to policy violations.

Radiation Protection Program. The sophistication of information presented depends upon the responsibilities of the individual.

Radiation Protection Engineers; Radiation Protection Foremen; and Radiation and Process Monitors positions are classified as radiation protection "professionals" for decommissioning activities.

Individuals in these positions are responsible for the implementation and day-to-day work associated with the Radiation Protection Program. Persons in these positions have prior training/experience in the radiation protection field and/or receive extensive training in the radiation protection procedures and work practices at HBPP.

Radiation protection "professionals" are given comprehensive training in radiation protection, covering theory and practical experience with equipment at a technical level beyond GET.

Radiation workers complete a training course that provides classroom instruction and practical demonstrations to permit them to perform their work in an efficient, safe manner. Topics include: radiation types, exposure, and biological effects; health protection problems, ALARA philosophy and program, and NRC rules and regulations; exposure reports, protective devices, routine radiation work permits (RWPs), and special work permits (SWPs); and high hazard areas, contamination, and decontamination. Training is based upon 32 key questions identified in Regulatory Guide 8.29, "Risks From Radiation Exposure." An examination is administered with a minimum passing grade of 70 percent. Non-radiation workers are provided fundamental information on radiation, health effects, and risks, as necessary.

Respiratory Protection. Individuals required to wear respirators are medically qualified, fit tested, and trained in the proper care and use of respirators. When negative pressure respirators are used, training includes qualitative fit testing. Individuals are acquainted with OSHA, Cal OSHA, NRC, and ANSI requirements for respirators, their use, and when they should be used.

Site Emergency Plans. Basic instruction helps individuals to recognize and respond correctly to emergency or warning signals and how to report fires or injuries. Annual emergency drills and exercises are conducted to demonstrate proficiency in various aspects of site emergency plans.

Industrial Safety, First Aid, and Fire Protection. PG&E's Accident Prevention Program provides basic instruction in these topics. Persons having unescorted access to Unit 3 must know the significance of barrier tapes and equipment status tags and must be familiar with fire prevention and fire fighting techniques. Individuals learn to use water and CO_2 system extinguishers and fire brigade equipment. A fire brigade training program is provided to fire brigade members for more extensive practical training.

Security Program. Individuals granted unescorted access to Unit 3 are required to understand the security program including badging, procedures for entering secure areas, measures to avoid causing security alarms, recognition and reporting of unauthorized personnel, and searching policy.

Quality Assurance/Control Program. Individuals who perform quality-related work receive training in the use of procedures, the documentation of work, discrepancy reporting, and the role of Quality Control inspectors.

4.2.3.3 Technical Training

In addition to GET as discussed above, some workers are provided with more extensive technical training appropriate for their work assignments. As an example, hazardous material and waste handling training is provided when necessary to supply information on waste management, environmental protection, safety standards, and personnel protection.

Operator Training and Certification Program. Under the Unit 3 possession-only license, all fuel movements are required to be performed under the supervision of a Certified Fuel Handler and a Certified Fuel Handler is to be present at the location of each fuel movement.

During the SAFSTOR period it is not expected that movements of spent reactor fuel will be made, except for training, special tests, or inspections to monitor the fuel in storage. At some time during the SAFSTOR period, fuel handling may be performed to transfer the spent fuel assemblies to the DOE for disposal.

A training and certification program has been implemented to maintain a staff properly trained and qualified to maintain the spent fuel, to perform any fuel movements that may be required, and to maintain Unit 3 in accordance with the possession-only license. This program provides the training, proficiency testing, and certification of fuel handling personnel. A detailed description of the program is provided in Appendix F.

The Operator Training and Certification Program ensures that people trained and qualified to operate Unit 3 will be available during the SAFSTOR period. This program is similar to that required by 10 CFR Part 72, Subpart I for Independent Spent Fuel Storage Facility personnel. Licensee certification of personnel makes it unnecessary for the NRC to periodically conduct license examinations for persons involved in infrequent activities and prevents delays due to obtaining NRC Fuel Handler Licenses for any evolutions that may require fuel movements.

Radiation Protection Department Training Program. A comprehensive program is presented to HBPP Radiation and Process Monitors (RPMs). The initial training consists of academic classroom training, on-the-job training, and retraining to implement changes and improve skills. The course requirements include:

- Nuclear Technology - basic nuclear and radiation protection theory
- Plant Design and Operation - plant layout, system functions, and equipment (as related to SAFSTOR conditions)
- Chemistry - analyses, calibration, and instrumentation
- Radiochemistry - sample preparation, counting, and data reduction. Use and maintenance of instruments
- Emergency Plan and Procedures - emergency responsibilities, surveys, analysis, radiation protection during accident conditions, and environmental monitoring
- Radiation Protection - in-depth training is presented on atomic theory, radioactivity, properties of types, units and dose, biological effects, standards, detection, dosimetry,

instruments, personnel monitoring, air sampling, instrument operation, counting statistics, internal dose calculations, shielding, exposure control, surveys, respiratory protection, and decontamination methods

- Review of 10 CFR 19, 20, 61, and 71

A retraining program covering the above topics shall be administered on a 2-year cycle.

Use of Monitoring Equipment. RPMs are provided with on-the-job training to enhance skills for operating various types of radiation detection equipment. Information discussed includes the proper use of probes, read-out evaluation, surveys and data to be collected, instrument response time, efficiency, and modes of operation.

Individuals required to use monitoring equipment are required to demonstrate proficiency with the equipment. The length of training is dependent on the individual's ability to perform his duties satisfactorily. As part of the RPM's periodic retraining, proficiency regarding monitoring equipment is verified every 2 years unless new equipment or procedural changes dictate a more frequent proficiency check.

RPMs are required to be requalified at least annually by written examination. In addition, every 2 years, RPMs are required to demonstrate satisfactory performance of the practical skills covered in the initial qualification program.

The Radiation Protection Engineers and RP Foremen are qualified by training and experience, and are not required to be requalified.

Nuclear Quality Services Training

The Nuclear Quality Services Department provides review and input into the plant General Employee QA/QC Training content, and coordinates or performs other QA/QC training as required.

4.2.3.4 Other Training

It is anticipated that other technical topics will be presented to personnel on an as-needed basis. Current administrative guidelines will be followed to establish new procedures and to ensure the training is completed. Suggested content is presented for two such courses:

Radioactive Waste Volume Minimization.

The course should identify the methods and techniques of achieving the following goals:

- Liquid radioactive waste control
- Control of material entering radioactive materials area
- Contamination control
- Waste segregation program

- Radiological work planning
- Decontamination process
- Waste packaging and transport (RPMs only)

Additional special topics may be added to this list if the need arises to assure safe and timely execution of new work tasks. Training on any such topics will be presented to those persons who will actively participate in those new work tasks.

<u>Decontamination Workers</u>

Decontamination workers will be qualified for performing routine decontamination tasks by either experience or training, or a combination of both. Training for non-routine decontamination processes, e.g., chemical decontamination, mechanical decontamination, ultrasonic decontamination, electropolishing, etc., and training for using specialized decontaminating equipment, e.g., hydrolasers, scabblers, ultrasonics, sandblasters, solvents, etc. should be provided on an as-needed basis.

Decontamination workers may be trained on additional special topics, as deemed appropriate by Radiation Protection Supervision, which shall include the Radiation Protection Manager and the Radiation Protection Foreman, and may include the RPMs when appropriate.

4.2.3.5 Training Program Administration and Records

The HBPP Plant Manager is responsible for ensuring that the training requirements and programs are satisfactorily completed for site personnel. The HBPP Training Coordinator is responsible for the organization and coordination of training programs, ensuring that records are maintained and kept up to date, and assisting in training material preparation and classroom instruction.

Training that is required to satisfy a regulatory or procedural requirement is documented on a record of training sheet and accompanied by an attendance sheet. The training topic is identified on the record sheet and any applicable training materials is included or referenced as part of the package.

Records of training and qualification are retained for the duration of an individual's assignment to HBPP or for 5 years, whichever is longer. An exception to this is radiation protection training records, which are retained for the duration of the facility license.

Pursuant to 10 CFR 50.120, "each nuclear power plant licensee, [by November 22, 1993], shall establish, implement, and maintain a training program derived from a systems approach to training as defined in 10 CFR 55.4." The intent of 10 CFR 50.120 is to ensure that civilian nuclear power plant operating personnel are trained and qualified to safely operate and maintain the facility commensurate with the safety status of the facility.

On December 9, 1993, the U. S. Nuclear Regulatory Commission granted an exemption from the requirements in 10 CFR 50.120 to establish, implement, and maintain training programs, using the systems approach to training, for the categories of personnel listed in 10 CFR 50.120.

Exemption from the training rule, 10 CFR 50.120, does not relieve HBPP of any other training requirements or commitments, which have been established, with the NRC.

4.2.4 QUALITY ASSURANCE PROGRAM

Decommissioning and SAFSTOR activities will be performed in accordance with the Humboldt Bay Power Plant Unit 3 SAFSTOR Quality Assurance Program (QA Program). The QA Program is designed to ensure that decommissioning activities and activities during the SAFSTOR period are performed in accordance with the license, applicable codes, standards, and regulatory requirements, and that these activities will provide adequate protection for the health and safety of the public. Items and activities subject to the QA program include, but are not necessarily limited to:

- Radioactive material licensed shipping containers, and activities which could affect the required function thereof, as required by 10 CFR 71. This applies to shipment of licensed material in excess of type A quantities.
- Effluent and environmental monitoring equipment, and the activities that could affect the validity and accuracy of such measurements, as required by USNRC Regulatory Guide 4.15.
- Activities required by the Technical Specifications.

The QA Program is implemented by quality assurance procedures and HBPP procedures and instructions.

4.3 INDUSTRIAL HEALTH AND SAFETY PROGRAM

The Humboldt Bay Power Plant participates in an industrial safety program under the direction of the PG&E Safety, Health, and Claims Department. This program includes accident prevention, hazardous materials control, and hazardous waste management programs. The PG&E Safety, Health and Claims Department has overall responsibility for industrial safety programs within PG&E.

5.0 DSAR OPERATING AND SURVEILLANCE REQUIREMENTS

Testing of system components, monitors, and other equipment to which this section applies shall be performed within the specified time interval with:

- A maximum allowable extension not to exceed 25% of the test interval
- A total interval time for any three consecutive test intervals not to exceed 3.25 times the specified test interval

Appropriate tests shall also be performed following maintenance on these systems that could impair their operation.

5.1 FIRE PROTECTION PROGRAM

The fire protection system consists of the fire water system, fire hose stations, and penetration fire barriers:

a. The fire water system shall consist of three plant fire pumps each rated for 500 gpm at a discharge pressure of 120 psig. These pumps shall take suction from the 300,000-gallon raw water storage tank. The raw water storage tank shall have a low level alarm set at 200,000 gallons. Two of these pumps shall be powered from separate Unit 1 and 2 480-volt ac systems. The third pump shall be diesel powered. The pumps shall start automatically to maintain system pressure.

b. Fire hose stations in the Unit shall consist of those stations and equipment listed in Table 5-1.

c. Penetration fire barriers shall be provided for penetrations between the control room and the feed pump room, between the feed pump room and the refueling building, and between the control room and the refueling building. Penetration fire barriers shall be passive devices that ensure that a fire will be confined or adequately retarded from spreading to other portions of the facility.

5.1.1 FIRE WATER SYSTEM

a. The fire protection system shall be operable[1] with:

1. Two plant fire pumps, each with a rated capacity of 500 gpm, with their discharge aligned to the fire suppression header.

[1] Operable: A system, subsystem, train, component, or device may be considered operable or have operability when it is capable of performing its specific function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication, or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

2. A minimum of 200,000 gallons in the raw water storage tank.
3. An operable flow path capable of taking suction from the water supply tank and transferring the water through the distribution piping (with operable sectionalizing control or isolation valves) to the yard hydrant curb valves and the front valve in each hose standpipe listed in Table 5-1.

b. If the fire water system is inoperable, the system shall be restored to operable status within 7 days or a Special Report shall be prepared and submitted to the NRC Regional Administrator for Region IV within 30 days outlining the compensatory measures taken and a schedule of corrective action.

c. The following inspections and tests shall be performed:

1. At least once per 7 days, the raw water supply tank volume shall be verified to be at least 200,000 gallons.
2. At least once per 31 days, on a staggered test basis[2] each operable pump shall be started and operated for at least 15 minutes on recirculation flow.
3. At least once per 31 days, it shall be verified that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.
4. At least once per 12 months, each testable valve in the flow path shall be cycled through at least one complete cycle of full travel.
5. At least once per 18 months:
    a) A system functional test shall be performed which includes simulated automatic actuation of the system throughout its operating sequence.
    b) Each pump shall be verified to develop at least 450 gpm flow at a total head of 240 feet.
    c) Each fire pump shall be verified to start (sequentially) at a sustained fire suppression water system pressure no less than 45 psig.
6. At least every 2 years, the raw water storage tank low level alarm shall be calibrated to 200,000 gallons ±5 percent
7. At least once per 3 years, flow tests of the system shall be performed in accordance with Chapter 5, Section 11, "Fire Protection Handbook," 14th edition, published by the National Fire Protection Association.
8. The fire pump diesel engine shall be demonstrated operable as follows:
    a) At least once per 31 days:
        1) The fuel storage tank shall be verified to contain at least 150 gallons of fuel.

[2] Staggered test basis: (a) A test schedule for n (where n is equal to a number) systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and (b) the testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

2) The diesel shall be verified to start from ambient conditions and operate for at least 20 minutes.

b) At least once per 92 days, a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, shall be verified to be within the acceptable limits specified in Table 1 of ASTM D-975-74 with respect to viscosity, water content, and sediment.

c) At least once per 18 months, the diesel shall be verified to start from ambient conditions on the autostart signal and operate for at least 15 minutes while loaded with the fire pump.

d) The diesel shall be subjected to manufacturer's recommended servicing.

9. The fire pump diesel starting 12-volt batteries and chargers shall be demonstrated operable as follows:

a) At least once per 31 days:

1) The electrolyte level of each battery shall be verified to be above the plates.

2) Each battery's overall voltage shall be verified to be no less than 12 volts.

b) At least once per 92 days, specific gravity shall be verified to be 1.19 or greater.

c) At least once per 18 months:

1) The batteries, cell plates, and battery racks shall be verified to show no visual indication of physical damage or abnormal deterioration.

2) The battery-to-battery and terminal connections shall be verified to be clean, tight, free of corrosion, and coated with anticorrosion material.

5.1.2 FIRE HOSE STATIONS

a. The fire hose stations listed on Table 5-1 shall be operable whenever equipment in the area is required to be operable.

b. When a hose station required to be operable is inoperable, an additional equivalent capacity hose shall be routed to the unprotected area from an operable hose station within 1 hour.

c. Each fire hose station shall be verified operable as follows:

1. At least once per 31 days by visual inspection of the station to ensure that all equipment is available.
2. At least once per 18 months by removing the hose for inspection and reracking and by replacing all gaskets in the couplings that are degraded.

3. At least once per 3 years by partially opening each hose station valve to verify valve operability and no blockage.
4. At least once per 3 years by conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

5.1.3 PENETRATION FIRE BARRIERS

a. Penetration fire barriers shall be functional at all times.
b. When a penetration fire barrier is inoperable, a continuous fire detection capability shall be established on at least one side of the affected penetration within 1 hour and shall be maintained for as long as equipment on either side of the barrier is required to be operable.
c. Penetration Fire Barriers shall be verified to be functional by a visual inspection:
    1. At least once per 18 months
    2. Prior to declaring a penetration fire barrier functional following repairs or maintenance

5.1.4 FIRE PROTECTION PROGRAM PLANT STAFF

The Plant Manager is responsible for the fire protection program. A member of the plant staff, trained and experienced in the principles of industrial fire prevention and control, serves as Fire Marshal with responsibilities for periodic evaluation of equipment provided for fire fighting, brigade training, and maintaining a current and effective fire protection program.

At least three members of the shift operating organization shall be trained members of the fire brigade. These individuals shall be available to respond in the event of a fire emergency in the unit. The fire brigade shall not include any personnel required for other essential functions during a fire emergency.

5.1.5 FIRE PROTECTION TRAINING

A training program for the fire brigade shall be maintained under the direction of the Fire Marshall and shall meet or exceed the requirements of NFPA 600, "Standard on Industrial Fire Brigades."

5.2 STRUCTURES

5.2.1 REFUELING BUILDING

A thorough visual inspection of the refueling building shall be conducted at least quarterly. Evidence of deterioration shall be evaluated with regard to the function of the building as a weather enclosure, contamination control barrier, and radiation shield.

5.2.2 SPENT FUEL STORAGE POOL

Water quality in the spent fuel storage pool shall be maintained within the limits specified in Table 5-2. Spent fuel storage pool water shall be sampled and analyzed at least once per month. If water quality limits are exceeded, action shall be taken to restore the water quality to within the limits and an evaluation shall be conducted to determine the cause.

If water quality cannot be restored within the limits specified before the next required sampling, a report shall be submitted to the Regional Administrator, NRC Region IV, within the following 30 days.

5.3 SERVICE SYSTEMS

5.3.1 REFUELING BUILDING VENTILATION SYSTEM

The refueling building ventilation system shall be operated to maintain a negative pressure of at least ¼ inch of water whenever spent fuel is being moved or whenever work that might potentially damage spent fuel assemblies is in progress.

The capability of the refueling building ventilation system to maintain a negative pressure of ¼ inch of water in the refueling building shall be tested at the following times:

- Before removal of the spent fuel pool cover
- After any opening of the railroad door
- After maintenance that may have affected any of the refueling building penetration closure seals.

If not required for these reasons, the test shall be performed at least once each quarter.

5.3.2 SPENT FUEL STORAGE POOL SERVICE SYSTEMS

A minimum of 2,000 gallons shall be maintained in the demineralized water tank.

At least once per 31 days, the operability of the spent fuel storage pool liner gap pump shall be verified.

5.3.3 ELECTRICAL SYSTEMS

The emergency section of the 480 volt ac system normally shall be supplied from one of the Unit's two 480 volt ac buses. Provision shall be made for transferring the emergency section to a 480 volt ac source from Unit 1 or 2 and subsequently to an emergency generator rated at 60 kW. These transfers shall be initiated automatically by undervoltage relays on the emergency section. The emergency section shall supply the following loads:

- Emergency lighting
- Main annunciator system
- The following radiation monitoring systems: stack gas, process monitor, and area monitors

The transfer of the emergency 480 volt ac shall be tested for proper operation at least quarterly. This transfer shall be functionally tested annually with all loads connected to simulate emergency operation.

5.4 MONITORING SYSTEMS

5.4.1 AREA MONITORS AND PORTABLE MONITORING EQUIPMENT

The area monitors shall normally be in service at all times. They may be taken out of service for maintenance purposes but shall be returned to service as soon as practicable.

The two area monitors in the refueling building shall alarm at 15 mR/hr. At least one of these channels shall be available to monitor the fuel storage area and sound the evacuation horns in the refueling building. If this condition cannot be met, monitoring shall be accomplished with portable instruments whenever personnel are in the refueling building.

The remainder of the area monitors shall be set to alarm either at 1 mR/hr or at a radiation level within a factor of 2 of the normal maximum indicated radiation level.

The calibration of area monitors shall be checked at least once each quarter. The monitors shall be source-checked at least once each month and shall be calibrated annually.

Portable radiation detection instruments shall be calibrated at least annually.

Fixed and portable equipment will be used to support the following survey and sampling program: A gross beta-gamma radiation survey and a contamination survey of the Plant shall be conducted at least quarterly to verify that no radioactive material is escaping or being transported through containment barriers. Contamination samples shall be taken along the most probable path by which radioactive material (such as that stored in the inner containment regions) could be transported to the outer regions of the Plant and ultimately to the environs.

5.4.2 SPENT FUEL STORAGE POOL WATER LEVEL MONITORING

At least one water level monitor shall normally be operable at all times. One water level monitor at a time may be taken out of service for maintenance purposes but shall be returned to service as soon as practicable. At any time when both spent fuel pool water level monitors are inoperable, the water level shall be visually checked at least once each day.

The monitors shall be set to annunciate a low level condition whenever the water level in the spent fuel storage pool drops below elevation 10 feet, 8 inches. Level indication of the spent fuel storage pool water level monitors shall be verified monthly. The level monitors shall be calibrated and the alarm setpoints verified annually.

5.4.3 SEALED SOURCE LEAK TESTING

Each sealed source containing radioactive material in excess of 100 μCi of beta-and/or gamma-emitting material or 10 μCi of alpha-emitting material shall be tested for leakage and contamination.

If the test reveals the presence of contamination in excess of 0.005 μCi of removable contamination, the source shall be immediately removed from service and decontaminated, repaired, or disposed of in accordance with Commission regulations. A report shall be prepared and submitted to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, within 30 days of the date the leak test result greater than 0.005 μCi is discovered. The report shall specify the source involved, the test results, and corrective action taken. Records of leak test results shall be kept in units of microcuries.

Each sealed source shall be tested for leakage and contamination by the licensee or other persons specifically authorized by the U.S. Nuclear Regulatory Commission or an Agreement State. The test method shall have a detection sensitivity of at least 0.005 μCi.

Each sealed source described above (excluding startup sources and fission detectors previously subject to core flux) shall be tested for leakage and contamination as follows:

- Sources in Use - At least once per 6 months for all sources containing radioactive material, other than hydrogen 3, with a half-life greater than 30 days and in any form other than gas.
- Stored Sources Not in Use – Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

TABLE 5-1

FIRE HOSE STATIONS

Fire Hose Station Location	Size	Type *
West wall of the reactor feed pump room opposite the motor-generator sets	1.5 in. x 50 ft.	Hose reel spray nozzle
Reactor feed pump room at foot of stairs	1.5 in. x 50 ft.	Hose reel spray nozzle
Refueling building north wall near air lock	1.5 in. x 50 ft.	Hose reel spray nozzle
Refueling building top of stairs air lock	1.5 in. x 50 ft.	Hose reel spray nozzle
North wall of makeup demineralizer room	1.5 in. x 50 ft.	Hose reel spray nozzle
East wall of control room	1.5 in. x 100 ft.	Hose reel fog nozzle
North wall of turbine wash-down	1.5 in. x 100 ft.	Hose reel fog nozzle
Roof of refueling building	1.5 in. x 100 ft.	Hose reel fog nozzle

* In the National Fire Protection Association Fire Protection Handbook (Sixteenth Edition), Section 15, "Public Fire Protection," Chapter 6, "Fire Department Apparatus, Equipment and Facilities," Nozzles are grouped into two major categories: Spray and straight stream. In the description of spray nozzles, the terms spray and fog are used interchangeably; therefore, it is appropriate to use the terms "spray nozzle" and "fog nozzle" in the table, interchangeably.

TABLE 5-2

LIMITS FOR SPENT FUEL STORAGE POOL WATER CHEMISTRY AND ACTIVITY DURING SAFSTOR

	Parameter	Limits[a]
1.	pH	5.3 to 6.5
2.	Chlorides[b]	0.5 ppm (maximum)
3.	Conductivity	10.0 μmho/cm (maximum)
4.	^{137}Cs activity[c]	1.0×10^{-4} μCi/ml (maximum)

a Verification shall be accomplished by analysis of samples taken at least once each month .

b Chloride analysis is required only if conductivity exceeds 2.0 μmho/cm (Reference NRC Regulatory Guide 1.56, Figure 2).

c By gamma spectrometry.

APPENDIX A

Implications of Accidents During SAFSTOR

The fuel in the Spent Fuel Pool contains the greatest percentage of the facility inventory of radionuclides. The Spent Fuel Pool will remain in service during the SAFSTOR period to provide a stable, benign environment for storing the fuel. In comparison to the reactor core during operations, water temperatures and flows are a small fraction of those experienced previously by the fuel. The large volume of water in the pool provides a heat sink and containment for small amounts of radioactivity transferred from the fuel. Releases of radioactive materials will be minimized by protection of the cladding integrity, containment of spent fuel pool water, and removal of radioactive and other contaminants from the water.

The purity of the water will be maintained during SAFSTOR to prevent corrosion and to control radioactive materials transferred from the fuel. Water pH and levels of contaminants will be maintained in ranges where chemical attack is minimized to protect the primary (fuel cladding) and secondary (pool liner) barriers against release of radioactivity.

By maintaining radioactive concentrations in the pool water ALARA, radiation levels in the vicinity of the pool will be reduced and increases in normal rates of radionuclide transfer will be readily detectable. Also, if failure of the lining should occur, releases will be minimized.

Early in the operation of Unit 3, leakage of the Spent Fuel Pool was detected and a stainless steel liner was installed to alleviate this problem. Approximately 50 liters (12 gallon) of water is pumped from the liner every 5 to 7 days with leakage from the pool accounting for about 5 percent of this volume. Sampling of the french drain (under the Spent Fuel Pool), is conducted on a periodic basis. ^{137}Cs and ^{134}Cs radionuclide concentrations in the blotter samples are approximately 1 percent of the concentrations found in the liner. The radionuclide concentrations are below the limits specified in 10 CFR 20.

The fuel itself does not greatly contribute to the personnel exposure associated with the spent fuel storage. The water depth (approximately 18 feet over the top of the fuel) provides adequate shielding of the spent fuel. A study conducted by Pacific Northwest Laboratory at the Morris Spent Fuel Storage Facility (PNL-3065) found that the direct gamma radiation cannot be distinguished from the contribution made by radioactive water contaminants (approximately 3 mR/h at fuel depths of 8 feet).

Radiation sources in the vicinity of the spent fuel storage pool are due to deposits of radioactive material on the pool wall, particularly at the surface (tub ring effect), pool water, cleaning lines, pumps, filters, etc.

Accidents during the SAFSTOR period have a low probability of occurrence and are of minor consequence, especially when compared with accidents associated with reactor operations. Accidents possible during SAFSTOR operations are analyzed in the assessment presented below.

1.1 IDENTIFICATION AND PROBABILITY OF ACCIDENTS DURING SAFSTOR

The following five accidents were identified as credible and/or worthy of assessment for the SAFSTOR period:

- Spent fuel handling accident
- Spent Fuel Storage Pool rupture
- Heavy load drop into the Spent Fuel Storage Pool
- Uncontrolled release of radioactive liquid radwaste to the environment
- Explosions, delayed ignition of flammable vapor clouds, release of toxic chemicals, or fire

1.1.1 Spent Fuel Handling Accident

It is anticipated that the spent fuel assemblies will continue to be stored in racks throughout much of the SAFSTOR period. A protective cover is in place over the pool. The only anticipated fuel handling is for inspection, training, or testing or for removal of the fuel for shipment to a permanent repository. The potential for dropping a spent fuel assembly during SAFSTOR has been calculated based on Unit 3 experience and the number of spent fuel assemblies stored in the Spent Fuel Storage Pool. Three assemblies were dropped during the total handling of all assemblies onsite (over 3000 assembly handlings), including those handlings associated with shipping of 180 stainless steel-clad assemblies off site. The calculated probability assumed a linear relationship between drop rate and the number of assemblies handled and assumed no decrease in probability as a result of experience. The probability thus calculated is unity, i.e., at least one assembly may be dropped during handling. Transfer of fuel for shipment is considered as the initial operation of DECON and the probability and consequence of an accident during these activities is discussed in Appendix A of the dismantlement plan.

During an inspection, the protective pool cover would be removed. Spent fuel handling will be accomplished in the same manner as during the operations of Unit 3. The implications of the radiological release, as a result of a fuel handling accident, to conditions at the site boundary were assessed using the assumptions and methodology suggested in Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." The assumptions applied to this analysis were:

- The dropped spent fuel assembly contained the greatest radiological inventory of those stored in the pool.

- ^{85}Kr is the single gaseous nuclide of significance for transport up the stack and into the environment. The maximum inventory of ^{85}Kr within the dropped assembly is calculated to be 98 Ci (July 1984).

- All of the fuel gap inventory of all rods in the assembly are released. The gap inventory is assumed to be 30 percent of the volume, or 29.4 Ci ^{85}Kr. For purposes of conservatism, the analysis was conducted as if the entire assembly inventory of 98 Ci was vented.

- No ^{85}Kr is retained in the pool. The entire 98 Ci of ^{85}Kr is released through the 50 foot stack over a 2-hour period.

- Still air conditions exist at the time of the accident and are maintained throughout the 2-hour venting of ^{85}Kr.

- The individual hypothetically exposed is located on the plant boundary or beyond at the point of maximum concentration of ^{85}Kr throughout the 2-hour time period.

1.1.2 A Rupture of the Spent Fuel Pool

The spent fuel storage pool and surrounding structure were designed to withstand an OBE of 0.25g and an SSE of 0.50g with the exception of the 75-ton bridge crane rails. The crane will not be stored over the pool and will not be used for lifting heavy loads over the pool.

A potential for a seismic event in the area exists due to the proximity of the Bay entrance and Little Salmon faults. The maximum event which is credible from either fault is 7.5 (Modified Richter Scale), (Woodward-Clyde Consultants, 1980). Because of the nature of seismic events, it is not useful to attempt calculation of a probability. However, if it is assumed that a seismic event of the maximum activity occurs during SAFSTOR, it is possible to compare the ground motion resulting from the event to the design basis of the spent fuel storage pool. Two results of earthquake activity which could be of structural consequence to Unit 3 are liquefaction and soil-structure interaction. Based on the Woodward-Clyde Consultants (1980) studies, liquefaction potential was assessed to be negligible. These studies also conclude that the only potential soil structure interaction effect resulting from earthquake activity is on the reactor caisson. Based on the data available, the design basis for the caisson (Bechtel, 1980) is conservative with respect to these effects. The resulting implication of a seismic event to the spent fuel storage pool's integrity are, therefore, negligible.

For the purpose of analysis, however, it was assumed a rupture of unknown origin occurred in the west side of the pool floor where it overhangs the suppression chamber. The water would then drain into the suppression chamber leaving the spent fuel completely exposed to the air. Further, for the purpose of analysis, it was also assumed that the pool cover was breached by the event.

1.1.3 Heavy Load Drop Into the Spent Fuel Pool

A protective cover has been installed over the spent fuel storage pool. As a precaution to minimize any potential for increased reactivity due to a heavy load drop, the spent fuel is configured in the pool to reduce collective reactivity of the stored array and to provide optimum physical separation of the individual assemblies. In addition, the fuel assemblies are individually enclosed in boral channels to further reduce the reactivity of each element.

The building structure, including support for the emergency condenser, was designed and built to seismic criteria considered sufficient at the time for an operating plant. Modifications were made to enhance the seismic design to more stringent standards in the mid 1970s. The structure is considered to be adequately protected against damage from natural events which could result in heavy load drops into the pool.

There are no plans for lifting or transferring heavy loads over the pool during SAFSTOR. Transfer of spent fuel to a shipping cask is considered an initial operation of DECON. No other operational

accident resulting in a heavy load drop into the pool is considered credible based upon the administrative controls and interlocks that will be in effect during SAFSTOR. The administrative controls include: (1) a prohibition to store the crane over the pool, (2) a requirement to not handle heavy loads over the spent fuel storage racks, (3) personnel training requirements, (4) procedural controls for load handling and crane operation, and (5) the crane will not be used to lift heavy loads over the pool. The circuit breaker for the crane power supply is locked and administratively controlled. Movement of the crane from its stored position is alarmed.

1.1.4 Uncontrolled Discharge of Radwaste Tankage

During the operating life of the plant, few incidents were encountered in the radwaste treatment system. In 1973, the storage capacity of the concentrated waste tanks was exceeded, resulting in overflow into the vault containing the tanks. No significant off-site release occurred.

In July 1977, overflow of the radwaste sump resulted in an unmonitored release to the discharge canal. The rate was approximately 1 gallon/min, and total discharge was approximately 2,000 gallons. No radiological release limits were exceeded.

Based on these experiences, it was determined that the worst case accident to the radwaste treatment system that could occur during SAFSTOR would be the loss of radwaste from the two concentrated waste storage tanks, a total volume of 10,900 gallons. For the purpose of the analysis it was assumed that the entirety of the spill would be lost to the discharge canal. Conversely it was assumed that all of the waste was retained in the soil near the tanks, requiring exhumation and subsequent waste handling as LLW.

1.1.5 Explosions, Fires, and Toxic Chemical Release

Offsite accidents could occur in Humboldt Bay or on the railroad tracks east of the HBPP resulting in explosions, fires, or releases of toxic chemicals. Based on the industry experience and the very low shipping rate by either rail or tanker in the area of the plant, the probability of these accidents has been established to be 10^{-7} per year.

The worst credible accident is the explosion and associated fire in the two large fuel oil storage tanks, assuming both were filled. The fuel stored onsite is combustible but non-explosive. Studies of industrial experience with similar tanks suggest that the probability of spontaneous explosion is negligible. For purposes of this analysis, it was assumed that the following conditions would occur as a result of this accident:

- Offices would be structurally destroyed.
- Fencelines would be breached on the south and east sides of the plant near the intake canal.
- Major superstructure damage would occur to Units 1 and 2.
- Rupture of the refueling building containment would occur.
- Damage would occur to the ventilation stack.
- Fire would surround the radwaste treatment facility.

The probability of rupture of the refueling building containment is small, even from a massive explosion of both oil storage tanks. Administrative controls and emergency procedures are sufficient to maintain surveillance and security of the fuel inventory throughout the emergency conditions.

1.2 CONSEQUENCES OF POTENTIAL ACCIDENTS DURING SAFSTOR

While accidents have an extremely small probability of occurrence during SAFSTOR, the consequences were analyzed to determine the potential worst case doses resulting from these potentialities.

1.2.1 Consequences of a Fuel Handling Accident

If a fuel assembly were damaged during handling (see assumptions in Section 1.1.1) such that the entire ^{85}Kr inventory was lost to the atmosphere, the maximally exposed individual would receive 0.13 mRem over the 2-hour release based on ^{85}Kr emission. Total iodine release is negligible. The resulting dose would be negligible. If a fuel handling accident resulted in the release of the entire ^{85}Kr inventory in the stored fuel, the dose the maximally exposed individual would receive would be 5.1 mRem, less than 1% of the 10 CFR Part 100 guideline value of 25 Rem to the whole body.

Occupational doses would be increased to collect fuel components, repackage and store them. The occupational dose is expected to not significantly impact annual personnel exposures or increase the number of personnel required for operations.

Environmental quality would be negligibly affected by such an accident. Air quality would not be affected by ^{85}Kr release and no other nuclides would be released in concentrations of any consequence. Water quality should not be impacted since the pool water cleanup system would remove the additional radioactivity (except for ^{3}H and ^{99}Tc) before any water was discharged to the liquid radwaste system. The maximum concentration of ^{3}H and ^{99}Tc in any assembly is 5.9 Ci and 0.24 Ci respectively. If the entire ^{3}H and ^{99}Tc inventory of the maximum activity assembly were mobilized and dispersed in the water in the pool (4.17×10^5 liters), the pool water concentration would be 1.41×10^{-2} and 5.76×10^{-4} μCi/ml for ^{3}H and ^{99}Tc, respectively. If this water were discharged into the discharge canal through the radioactive waste treatment system at a rate of 57 liters per minute, the dilution flow of the circulating water pumps (5.4×10^5 liters per minute) would result in a ^{99}Tc concentration of 6.08×10^{-8} μCi/ml, and a ^{3}H concentration of 1.49×10^{-6} μCi/ml.

Assuming the total core inventory of ^{99}Tc (67 Ci) and ^{3}H (1700 Ci) was released under these same conditions, the resulting effluent concentration of ^{99}Tc would be 1.70×10^{-5} μCi/ml and the effluent concentrations of ^{3}H would be 4.30×10^{-4} μCi/ml. Both of these values are well below the values given in 10 CFR, Part 20, Appendix B, Table 2.

The maximum quantity of ^{99}Tc in any fuel assembly is 0.24 Ci. Assuming an accident caused release of this activity to the spent fuel pool (110,000 gallon), the resulting pool concentration of ^{99}Tc would be 5.76×10^{-4} μCi/ml. If this water was discharged into the discharge canal through the radioactive waste treatment system at a rate of 57 liters per minute, the dilution flow of the circulating water pumps (5.4×10^5 liter per minute) would result in a ^{99}Tc concentration of 6.08×10^{-8} μCi/ml.

Assuming core inventory of ^{99}Tc (67 Ci) was released under these same conditions, the resulting

effluent concentration of ^{99}Tc would be 1.70 x 10^{-5} μCi/ml. Both of these values are well below the values given in 10 CFR, Part 20, Appendix B, Table 2.

The consequences of a fuel handling accident are minimal. Occupational dose would increase due to handling, fuel repackaging, and repairs. Public dose would not be significantly increased. Environmental quality would not be significantly impacted.

1.2.2 <u>Consequences of a Rupture of the Spent Fuel Storage Pool</u>

The loss of radiation shielding by complete loss of pool water would create high radiation levels (100 mR/hr) in the refueling building. Since the radiation would be directly above the pool interior and the pool is below grade, off-site radiation increase would be negligible. Recovery from the accident would increase external exposure of workers because of scattered radiation above the pool. Refilling of the pool would provide the necessary shielding for repair activities.

Reestablishment of the water shielding would require an estimated 3 person-rems. The ventilation efficiency might be decreased so that the inhalation dose might be slightly increased by approximately 1 person-rem over the assumed month's repair efforts.

A calculation has been performed to determine the dose rate at the exclusion area boundary assuming total loss of spent fuel pool water. The calculated dose rate for this postulated accident was 0.024 mRem/hr, at the exclusion area boundary. With this dose rate, there would be ample time to take mitigative action (e.g., reflooding the pool or otherwise shielding the spent fuel) before the design basis accident dose (5 rem) would be received offsite.

The calculation was based on the following information:

- Since the total activity in the spent fuel is several orders of magnitude greater than the activity in the pool water, the activity of the trapped pool water was disregarded for this calculation.

- The total activity in the spent fuel (8.5 x 10^5 Ci) was obtained using the inventory given in Table 10.4.5 of the HBPP Environmental Report.

- The average photon energy of approximately 0.5 MeV, with approximately 1 photon per disintegration, is based on the inventory given in Table 10.4.5 of the HBPP Report.

- The distance from the refueling building to the nearest site boundary (exclusion area boundary) is 700 feet, as discussed in Section V of the Final Hazards Summary Report.

- The spent fuel was assumed to be a point source located 7 feet (the approximate length of a fuel element) above the spent fuel pool floor. This point is 19 feet below ground level and 39 feet below the top of the concrete containment building walls. Due to the physical location of the spent fuel, there is no offsite dose from direct radiation.

- Per the guidance of ANSI/ANS-6.6.1-1976, "Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants", air-scattered radiation is only considered if it is shielded by less than 2 feet of concrete. Therefore, the only source of air-scattered radiation considered is that penetrating the 1-foot-thick concrete refueling building roof.

- The point source is geometrically modeled as being located behind a shield wall with a height of 39 feet and an internal radius of 20 feet. Skyshine is calculated at a receptor location 720 feet from the source.

- The skyshine calculational methodology was taken from Consumers Power's "Advanced Health Physics Training Manual," prepared by NUS Corporation.

- An attenuation calculation was performed (with no correction for buildup) to account for the shielding provided by the refueling building roof. This resulted in a reduction factor of 0.002 to the offsite dose rate.

The loss of water from the pool has no effect on criticality potential. Repositioning of the array configuration cannot impact criticality potential. This is discussed in Section 1.2.3. The decay heat rate from the fuel stored in Unit 3 is sufficiently low that the cladding is not subject to deterioration by thermal effects.

An analysis to address total loss of spent fuel storage pool water is provided in Appendix B, "Spent Fuel Heat Up Following Loss of Storage Pool Water."

Any releases of radioactive material should therefore be limited to relatively minor amounts of particulates that could become suspended in air from contaminants on pool surfaces when desiccated. These would lead to surface contamination in the refueling building but would not be significantly transported into the environment.

Ground water could be contaminated by the pool water, but the contamination would be very slight. Resulting concentrations of the three significant isotopes in ground water would approximate 1.2E-7 μCi/ml ^{137}Cs, 7.5E-9 μCi/ml ^{134}Cs, and 2.5E-10 μCi/ml ^{60}Co. This estimate is conservative since the water volume released from the pool would reach equilibrium with the very high water table in the site soil strata, resulting in less release than the entire pool volume.

All data at the site indicate that the net flow of groundwater in the vicinity of the spent fuel storage pool is towards Humboldt Bay. Flowmeter measurements made by Bechtel in 1984 indicated that the groundwater flow direction is affected by the tidal cycles in Humboldt Bay. During the summer months, and during flood tides, groundwater flow is generally landward. During ebb tides, groundwater flow is towards the bay. During winter months groundwater flow would also be affected by tidal actions but not as much as in the summer. During the winter, the controlling flow would be always toward Humboldt Bay even though, during flood tides there may be a minor temporary flow reversal.

The tidal cycle in Humboldt Bay for a 24-hour period consists of two flood tides and two ebb tides. Therefore, the flow reversals occur four times a day. Notwithstanding this flow reversal phenomenon, the controlling gradient of the flow of groundwater is still toward Humboldt Bay. The tidal cycle form July 8, 1979 to July 15, 1979 was studied for flow reversal effects near the spent fuel storage pool. The tide varied from a maximum elevation of 7.4 feet to a minimum of –1.6 feet compared to mean lower low water (MLLW). Assuming a constant summer groundwater elevation of 6 feet, it was calculated that a particle released underneath the spent fuel pool would travel a distance of 5.10 feet within the seven-day period toward Humboldt Bay, at a calculated average travel speed of 0.70 feet/day.

(a) Onsite Groundwater

The maximum volume of radioactive materials that can be released to the groundwater underneath the spent fuel storage pool was assumed to be equal to the volume from the top of the pool (elevation 12 feet above MLLW) to a mean tide level of 3.3 feet. A release of the radionuclides would be affected by the flow reversal pattern underneath the site, and the radionuclides would remain in that region for a long period of time due to the cyclic effect of the tides.

Two conditions were evaluated in order to determine the estimate radionuclide concentration of onsite groundwater at various depths due to a rupture of the spent fuel pool. Table A-1a shows a condition during the rainy season when the groundwater elevation was assumed to be at an elevation of 9 feet. Table A-1b shows a condition during the summer dry season when the groundwater elevation was assumed to be at an elevation of 6 feet.

(b) In the Humboldt Bay

As discussed earlier, the gradient of the groundwater flow is generally towards the Humboldt Bay. Even if there is a temporary reversal of flow due to tidal fluctuations, the effect of the reversal is not significant. The return flow (toward Humboldt Bay) is always greater than the reversal. Based on this concept, estimates of the concentration at various distances away from the rupture toward Humboldt Bay are shown in Tables A-2a and A-2b. Table A-2a shows a condition during the rainflood season with an assumed water elevation of 9 feet. Table A-2b shows a condition during the summer dry season when the groundwater level could be at an assumed elevation of 6 feet. Both of these tables were developed with average tide level for exit conditions. The concentration shown at a distance of 420 feet represents the exit point into the bay at which the isotopes are not yet subject to dilution in the bay. The concentration at Humboldt Bay will be diluted much more due to the flushing action of ocean water entering and leaving Humboldt Bay due to tidal action.

The only area which could be impacted by liquid discharges of radionuclides from Humboldt Bay Unit 3 is within the Humboldt Bay itself. Whenever possible, the following information on annual harvests is given in terms of Humboldt Bay. If Bay-specific data is not available, annual harvests are for the Bay and coastal areas for which data are available.

Information on the annual commercial harvest of some fish and shell fish is available specifically for Humboldt Bay. For fish such as salmon or surf perch, the harvested area for the port of Eureka ranges from the Mendocino coast to the Oregon border.

<u>Annual Commercial Yield</u>

Benthic	(Estimated yields within Humboldt Bay)		
	Oyster	-500,000 lbs	
	Crab	-2,000-3,000 lbs	
	Clam	-minor	
Pelagic	(Northern California yields brought into the port of Eureka)		
	Salmon	-130,000 lbs	(1984 yield)
	Perch	-12,000 lbs	(1984 yield)
	(Estimated yields within Humboldt Bay)		
	Herring	-40 to 60 tons	
	Anchovy	- 6 tons	(annual quota)

The available current information on the recreational harvest of marine fish is contained in a marine fisheries report covering the entire northern California coastline from Morro Bay to the Oregon border. Specific information on Humboldt Bay is not available directly from the report.

The following information on recreational yields has been derived from two sources: (1) the results of current species-specific surveys by the California Department of Fish and Game, and (2) Department of Fish and Game Fish Bulletin 130, dated 1965. The information contained in the Fish Bulletin was obtained in 1958 as part of a survey which estimated the annual yield from the major fishing pier within the bay (Lazio pier). The report does not include estimates of the total yield from Humboldt Bay which may be expected to be as great as two to three times that of the pier.

Annual Recreational Yield

Benthic (Estimated yields within Humboldt Bay)
Clams -123,000 ea.(based on random sampling)
Crab -data not available

Annual Recreational Yield

Pelagic - (Humboldt Bay and coastline within 25-mile range)
Salmon -78,000 lbs (estimate of 13,000 landings based on random sampling; average weight estimated to be 6 lbs) (Humboldt Bay Lazio pier)
Surf/Rock Fish -10,000 lbs (1958 yields at Lazio pier)

(c) At the Nearest Offsite Potable Water Supply Location

The two potable water wells nearest the spent fuel storage pool are owned by PG&E. Well No. 1 is about 650 feet east of the site and Well No. 2 is about 2,980 feet southeast of the site. These wells, which are sampled quarterly for activity, provide onsite water supplies.

As noted earlier, there is no radionuclide transport landward because of the persistent gradient of the groundwater level toward Humboldt Bay. As demonstrated earlier, even the cyclic variations of flow due to tidal effects would not cause a large enough reversal for a sufficient period of time to transport radionuclides eastward of the site.

Although not considered possible, an analysis was performed with an assumption that a severe drawdown at a particular well due to severe pumping resulted in a difference in head of 5 feet. Table A-3 shows the results of the analysis for Well Nos. 1 and 2.

The limit for total dissolved activity in the spent fuel storage pool water is given in DSAR Table 5-2. The limit is based on the anticipated levels that would be maintained following upgrading of the spent fuel pool water cleanup prior to SAFSTOR.

DSAR Section 2.3.1 states that the spent fuel pool circulation water pumps shall be operated as necessary to maintain spent fuel pool water within specified limits.

1.2.3 Consequences of a Heavy Load Drop Into the Spent Fuel Storage Pool

Based upon the discussion in Section 1.1.3, the potential for a heavy load drop into the spent fuel storage pool is extremely small. Administrative/procedural controls prevent the movement of heavy loads over the spent fuel, eliminating the potential for dropping a heavy load onto the spent fuel. If, however, design and/or administrative controls used for protection were to deteriorate and if a heavy load drop were to occur, the following consequences could result:

- The pool structure could crack. If the pool structure cracks, the worst crack could occur as postulated in Section 1.1.2 whereby water from the pool would drain to the suppression chamber. The consequences would be similar to those described in Section 1.2.2.

- A single fuel assembly or multiple fuel assemblies could be damaged such that the cladding would be breached. The consequences would be similar to those described in Section 1.2.1 for fuel handling accident.

- Spent fuel could be reconfigured/crushed such that an increase in reactivity could occur; however, with the fuel in Boral cans as discussed in Appendix C, there is no credible condition where criticality can occur.

PG&E has implemented design alternatives that would prevent possible criticality due to seismic and heavy load events. A report on this completed task is presented in Appendix C "Pacific Gas and Electric Company Humboldt Bay Power Plant Unit 3 Criticality Analysis for SAFSTOR Decommissioning."

1.2.4. Consequences of an Uncontrolled Release of Radwaste Tankage

In determining the quantities of radioactive materials that could be released in the worst case radwaste treatment system accident during SAFSTOR, it was assumed that due to radioactive decay, decontamination efforts and lower levels of operational activity, the radioactivity being added to the waste storage tank would be less than current additions. The concentrations of the four most significant radionuclides were assumed to be the maximum measured when the storage tank was sampled in 1984.

The tank was assumed to contain its maximum capacity (10,000 gallons) at its maximum concentration. The volume of the tank was assumed to be released to the discharge canal over an 11-hour period (15 gpm) through the 2-inch diameter waste discharge line.

The consequences of release of the 37,800 liters (10,900 gallons) of concentrated waste tank storage are measurable. If all the waste were discharged via the radwaste line to the canal, the concentrations of ^{137}Cs, ^{134}Cs, ^{60}Co, and ^{90}Sr, the four significant nuclides, would be less than water effluent concentration limits (10 CFR 20, Appendix B, Table 2) at the discharge point. If all the waste were spilled on the soil, the contaminated area would be approximately 94 m^2 to a depth of 40 cm, due to soil permeability. The approximate concentrations of the significant nuclides would be: 3.5 μCi/g ^{137}Cs, 1.2 μCi/g ^{134}Cs, 1.1 μCi/g ^{60}Co, and 0.01 μCi/g ^{90}Sr. If this contaminated area were exhumed for LLW disposal, 185 55-gallon drums would be required. Occupational dose is negligible from these operations. There is no impact on public dose or environmental quality.

1.2.5 Consequences of Explosion, Fire, and Toxic Chemical Release

An explosion and fire of the large fuel storage tanks on site would obviously cause damage to the plant facilities and incapacitate Units 1 and 2. The consequences to Unit 3 would be minor and could include:

Consequences to Security. Physical surveillance of any breached fences and gates would be required while repairs are completed.

Rupture of Refueling Building Containment. The working conditions in the refueling building during SAFSTOR will require personnel monitoring but no protective clothing under normal operating conditions. Negligible nuclide suspension to the air is therefore expected even if the building superstructure were entirely vented.

Damage to Ventilation Stack. Ventilation systems would be shut down and the suspended particulate dose to workers might increase slightly during repairs, estimated at less than 0.2 person-rem. No public exposure or environmental quality impact would result from radiological hazards.

Fire in the Unit 3 Restricted Area. There are no significant quantities of flammables or pressurized equipment in the area of the radwaste treatment and storage buildings. It is believed that no loss of stored wastes would result from a fire in their vicinity inside the Unit 3 restricted area. Although a calculation has not been performed to evaluate this particular sequence of events, it is not considered possible for a seismic event to rupture the spent fuel storage pool and the onsite fuel oil storage tank which then causes a fuel oil fire in the pool.

Each of the two main fuel oil storage tanks is surrounded by an earthen dike that has been in place for more than 20 years. The minimum dike cross-section is 10 feet top x 50 feet bottom x 10 feet high. The banks of the dikes are covered with vegetation and the tops are paved with asphalt. The capacity within each dike area is greater than the maximum available volume of the associated fuel oil storage tank (volume above the tank elevation which corresponds to the top of the dike). Therefore, even in the unlikely event of a tank rupture, all oil is expected to be contained within the fuel oil dike area.

In the unlikely event of rupture in the east side of the earthen dike, it is not expected that the fuel oil could reach the spent fuel pool since any flow in that direction would be impeded by the administration building and Units 1 and 2. It is more likely that a rupture of the dike in this area would result in flow to the intake canal.

Furthermore, the fuel oil stored in these tanks is extremely viscous, similar to the consistency of tar, and as such, it is not of a nature to flow freely. A fuel oil dike rupture in any other direction would result in flow away from Unit 3.

TABLE A-1a

ESTIMATED RADIONUCLIDE CONCENTRATION OF ONSITE GROUNDWATER AT VARIOUS DEPTH DUE TO POSTULATED RUPTURE OF SPENT FUEL STORAGE POOL (Rainy Season)

Given: Groundwater Level = 9 feet. Mean Tide Level = El. 3.3 feet. Postulated Volume of Water Released from Spent Fuel Storage Pool = 4524 feet^3. Hydraulic Conductivity K = 10,400 feet/year.

Isotope	Distance from Rupture (Feet)	Depth below Rupture (Feet)	Travel time (years)	Peak Concentration at location (μCi/ml)	Amount of radionuclide released (Ci)	Concentration NRC Limit (μCi/ml)*
^{137}Cs	30	0	6.95	2.7 X 10^{-4}	0.54	2 X 10^{-5}
		10		5.1 X 10^{-5}		
		20		3.5 X 10^{-7}		
		30		3.5 X 10^{-10}		
^{134}Cs	30	0	6.95	3.3 X 10^{-6}	0.055	9 X 10^{-6}
		10		6.2 X 10^{-7}		
		20		4.2 X 10^{-9}		
		30		4.2 X 10^{-12}		
^{90}Sr	30	0	0.58	4.6 X 10^{-6}	0.00067	3 X 10^{-7}
		10		8.8 X 10^{-7}		
		20		5.9 X 10^{-9}		
		30		5.9 X 10^{-12}		
^{60}Co	30	0	5.25	1.2 X 10^{-6}	0.0031	5 X 10^{-5}
		10		2.3 X 10^{-7}		
		20		1.6 X 10^{-9}		
		30		1.6 X 10^{-12}		

Note: Computations were based on point concentration model presented by Codell and Duguid (reference)

*From 10 CFR 20, App. B, Table 2, Column 2

TABLE A-1b

ESTIMATED RADIONUCLIDE CONCENTRATION OF ONSITE GROUNDWATER AT VARIOUS DEPTH DUE TO POSTULATED RUPTURE OF SPENT FUEL STORAGE POOL

Given: Groundwater Level = 6 feet. Mean Tide Level = El. 3.3 feet. Postulated Volume of Water Released from Spent Fuel Storage Pool = 4524 feet^3. Hydraulic Conductivity K = 10,400 feet/year.

(Dry Season)

Isotope	Distance from Rupture (Feet)	Depth below Rupture (Feet)	Travel time (years)	Peak Concentration at location (μCi/ml)	Amount of radionuclide released (Ci)	Concentration NRC Limit (μCi/ml)*
^{137}Cs	30	0	14.7	2.3×10^{-4}	0.54	2×10^{-5}
		10		4.3×10^{-5}		
		20		2.9×10^{-7}		
		30		2.9×10^{-10}		
^{134}Cs	30	0	14.7	2.5×10^{-7}	0.055	9×10^{-6}
		10		4.8×10^{-8}		
		20		3.2×10^{-10}		
		30		3.2×10^{-13}		
^{90}Sr	30	0	1.23	4.6×10^{-6}	0.00067	3×10^{-7}
		10		8.6×10^{-7}		
		20		5.8×10^{-9}		
		30		5.8×10^{-12}		
^{60}Co	30	0	11.1	5.7×10^{-7}	0.0031	5×10^{-5}
		10		1.1×10^{-7}		
		20		7.2×10^{-10}		
		30		7.3×10^{-13}		

Note: Computations were based on point concentration model presented by Codell and Duguid (reference)

*From 10 CFR 20, App. B, Table 2, Column 2

TABLE A-2a

PEAK CONCENTRATION AND TRAVEL TIME OF RADIONUCLIDES IN GROUNDWATER IN THE SITE AREA (Assumed Rainflood Season)

Given: Volume of water released from spent fuel storage pool = 4524 $feet^3$.
Groundwater Level = El. 9 feet; Mean Tide Level = El. 3.3 feet.
Hydraulic Conductivity K = 10,400 feet/year.

Distance from Rupture Source (ft)	Isotope	Travel time (years)	Concentration at Spent Fuel Pool (μCi/ml)	Peak Concentration at Specified Distance (μCi/ml)	NRC Limit (μCi/ml)*
50	^{137}Cs	11.6	4.22×10^{-3}	1.1×10^{-4}	2×10^{-5}
	^{134}Cs	11.6	4.27×10^{-4}	3.2×10^{-7}	9×10^{-6}
	^{90}Sr	0.97	5.2×10^{-6}	2.1×10^{-6}	3×10^{-7}
	^{60}Co	8.75	2.4×10^{-5}	3.6×10^{-7}	5×10^{-5}
100	^{137}Cs	23.1	4.22×10^{-3}	3.1×10^{-5}	2×10^{-5}
	^{134}Cs	23.1	4.27×10^{-4}	2.6×10^{-9}	9×10^{-6}
	^{90}Sr	1.94	5.2×10^{-6}	7.3×10^{-7}	3×10^{-7}
	^{60}Co	17.5	2.4×10^{-5}	4.0×10^{-8}	5×10^{-5}
150	^{137}Cs	34.7	4.22×10^{-3}	1.3×10^{-5}	2×10^{-5}
	^{134}Cs	34.7	4.27×10^{-4}	3.1×10^{-11}	9×10^{-6}
	^{90}Sr	2.92	5.2×10^{-6}	3.9×10^{-7}	3×10^{-7}
	^{60}Co	26.2	2.4×10^{-5}	7.0×10^{-9}	5×10^{-5}
200	^{137}Cs	46.3	4.22×10^{-3}	6.4×10^{-6}	2×10^{-5}
	^{134}Cs	46.3	4.27×10^{-4}	4.3×10^{-13}	9×10^{-6}
	^{90}Sr	3.9	5.2×10^{-6}	2.5×10^{-7}	3×10^{-7}
	^{60}Co	35.0	2.4×10^{-5}	1.4×10^{-9}	5×10^{-5}
250	^{137}Cs	57.9	4.22×10^{-3}	3.7×10^{-6}	2×10^{-5}
	^{134}Cs	57.9	4.27×10^{-4}	7.1×10^{-15}	9×10^{-6}
	^{90}Sr	4.86	5.2×10^{-6}	1.8×10^{-7}	3×10^{-7}
	^{60}Co	43.7	2.4×10^{-5}	3.5×10^{-10}	5×10^{-5}

TABLE A-2a (Con'd)
PEAK CONCENTRATION AND TRAVEL TIME OF RADIONUCLIDES IN GROUNDWATER IN THE SITE AREA
(Assumed Rainflood Season)

Distance from Rupture Source (ft)	Isotope	Travel time (years)	Concentration at Spent Fuel Pool (μCi/ml)	Peak Concentration at Specified Distance (μCi/ml)	NRC Limit (μCi/ml)*
300	^{137}Cs	69.5	4.22×10^{-3}	2.2×10^{-6}	2×10^{-5}
	^{134}Cs	69.5	4.27×10^{-4}	1.2×10^{-16}	9×10^{-6}
	^{90}Sr	5.83	5.2×10^{-6}	1.4×10^{-7}	3×10^{-7}
	^{60}Co	52.5	2.4×10^{-5}	8.6×10^{-11}	5×10^{-5}
**420	^{137}Cs	97.3	4.22×10^{-3}	7.1×10^{-7}	2×10^{-5}
	^{134}Cs	97.3	4.27×10^{-4}	7.8×10^{-21}	9×10^{-6}
	^{90}Sr	8.17	5.2×10^{-6}	8.2×10^{-8}	3×10^{-7}
	^{60}Co	73.5	2.4×10^{-5}	3.4×10^{-12}	5×10^{-5}

* From 10 CFR 20, App. B, Table 2, Column 2

** Distance to the exit point at bay

Note: Computations were based on simplified analytical methods for minimum dilution presented in the reference (Codell and Duguid)

TABLE A-2b
PEAK CONCENTRATION AND TRAVEL TIME OF RADIONUCLIDES IN GROUNDWATER IN THE SITE AREA
(Assumed Rainflood Season)

Given: Volume of water released from spent fuel storage pool = 4524 $feet^3$.
Groundwater Level = El. 6 feet; Mean Tide Level = El.3.3 feet.
Hydraulic Conductivity K = 10,400 feet/year.

Distance from Rupture Source (ft)	Isotope	Travel time (years)	Concentration at Spent Fuel Pool (μCi/ml)	Peak Concentration at Specified Distance (μCi/ml)	NRC Limit (μCi/ml)*
50	^{137}Cs	24.5	4.22×10^{-3}	8.4×10^{-5}	2×10^{-5}
	^{134}Cs	24.5	4.27×10^{-4}	4.6×10^{-9}	9×10^{-6}
	^{90}Sr	2.06	5.2×10^{-6}	2.1×10^{-6}	3×10^{-7}
	^{60}Co	18.5	2.4×10^{-5}	9.9×10^{-8}	5×10^{-5}
100	^{137}Cs	49.0	4.22×10^{-3}	1.7×10^{-5}	2×10^{-5}
	^{134}Cs	49.0	4.27×10^{-4}	5.0×10^{-13}	9×10^{-6}
	^{90}Sr	4.1	5.2×10^{-6}	7.0×10^{-7}	3×10^{-7}
	^{60}Co	37.0	2.4×10^{-5}	3.1×10^{-9}	5×10^{-5}
150	^{137}Cs	73.4	4.22×10^{-3}	5.2×10^{-6}	2×10^{-5}
	^{134}Cs	73.4	4.27×10^{-4}	8.6×10^{-17}	9×10^{-6}
	^{90}Sr	6.17	5.2×10^{-6}	3.6×10^{-7}	3×10^{-7}
	^{60}Co	55.5	2.4×10^{-5}	1.5×10^{-10}	5×10^{-5}
200	^{137}Cs	97.9	4.22×10^{-3}	1.9×10^{-6}	2×10^{-5}
	^{134}Cs	97.9	4.27×10^{-4}	1.9×10^{-20}	9×10^{-6}
	^{90}Sr	8.22	5.2×10^{-6}	2.2×10^{-7}	3×10^{-7}
	^{60}Co	74.0	2.4×10^{-5}	8.7×10^{-12}	5×10^{-5}
250	^{137}Cs	122.4	4.22×10^{-3}	8.3×10^{-7}	2×10^{-5}
	^{134}Cs	122.4	4.27×10^{-4}	4.0×10^{-24}	9×10^{-6}
	^{90}Sr	10.28	5.2×10^{-6}	1.6×10^{-7}	3×10^{-7}
	^{60}Co	92.5	2.4×10^{-5}	5.9×10^{-13}	5×10^{-5}

TABLE A-2b (Con'd)
PEAK CONCENTRATION AND TRAVEL TIME OF RADIONUCLIDES IN GROUNDWATER IN THE SITE AREA
(Assumed Rainflood Season)

Distance from Rupture Source (ft)	Isotope	Travel time (years)	Concentration at Spent Fuel Pool (μCi/ml)	Peak Concentration at Specified Distance (μCi/ml)	NRC Limit (μCi/ml)*
300	^{137}Cs	146.9	4.22×10^{-3}	3.7×10^{-7}	2×10^{-5}
	^{134}Cs	146.9	4.27×10^{-4}	9.7×10^{-28}	9×10^{-6}
	^{90}Sr	12.33	5.2×10^{-6}	1.2×10^{-7}	3×10^{-7}
	^{60}Co	111.0	2.4×10^{-5}	4.1×10^{-14}	5×10^{-5}
**420	^{137}Cs	205.6	4.22×10^{-3}	5.9×10^{-8}	2×10^{-5}
	^{134}Cs	205.6	4.27×10^{-4}	2.3×10^{-36}	9×10^{-6}
	^{90}Sr	17.3	5.2×10^{-6}	6.6×10^{-8}	3×10^{-7}
	^{60}Co	155.4	2.4×10^{-5}	7.6×10^{-17}	5×10^{-5}

* From 10 CFR 20, App. B, Table 2, Column 2

** Distance to the exit point at bay

Note: Computations were based on simplified analytical methods for minimum dilution presented in the reference (Codell and Duguid)

TABLE A-3
ESTIMATED RADIONUCLIDE CONCENTRATION AT PG&E WELLS NO. 1 AND 2 FOR POSTULATED SEVERE DRAWDOWN ONLY AT WELLS

Assumed Difference in Head = 5 feet

Location	Isotope	Concentration at Spent Fuel Pool (μCi/ml)	Travel time (Year)	Peak Concentration at Well (μCi/ml)	NRC Concentration Limit (μCi/ml)*
PG&E Well #1 (650 ft. East)	^{137}Cs	4.22×10^{-3}	256.4	3.25×10^{-8}	2×10^{-5}
	^{134}Cs	4.27×10^{-4}	256.4	2.15×10^{-43}	$9 X 10^{-6}$
	^{90}Sr	5.2×10^{-6}	21.5	1.1×10^{-7}	3×10^{-7}
	^{60}Co	2.4×10^{-5}	193.7	9×10^{-19}	5×10^{-5}
PG&E Well #2 (2980 ft. SE)	^{137}Cs	4.22×10^{-3}	1175.3	3.8×10^{-8}	2×10^{-5}
	^{134}Cs	4.27×10^{-4}	1175.3	0	9×10^{-6}
	^{90}Sr	5.2×10^{-6}	98.7	3.3×10^{-9}	3×10^{-7}
	^{60}Co	2.4×10^{-5}	888.2	6.4×10^{-57}	5×10^{-5}

*From 10 CFR 20, App. B, Table 2, Column 2

NOTE: The following assumptions were used in all calculations for the tables shown in the response.

1. Hydraulic conductivity was based on Bechtel's report, K = 10,400 ft/year (28.5 ft/day)

2. Other parameters used are:

   Effective porosity = 0.25
   Total porosity = 0.40
   Longitudinal dispersion = 1.0 ft
   Lateral dispersion = 0.5 ft
   Distribution coefficients

   ^{137}Cs and ^{134}Cs = 20 ml/g
   ^{90}Sr = 1.5 ml/g
   ^{60}Co = 15 ml/g

APPENDIX B

Spent Fuel Heat Up Following Loss of Storage Pool Water

Rev. 5
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SUMMARY

Plant specific fuel heatup analyses (Ref. 4 and 6) were made for Humboldt Bay Power Plant's spent fuel storage configuration for the unlikely event of loss of spent fuel storage pool water. The analysis concluded that the Humboldt Bay spent fuel heatup would not challenge the integrity of the fuel cladding, the fuel rack, or the Boral cans. It can be concluded, therefore, that there would be no adverse impact to the health and safety of the public as a result of this postulated accident at Humboldt Bay.

1. INTRODUCTION

Humboldt Bay's nuclear unit, a 65 MWe boiling water reactor (BWR), has been shut down since July 1976, and its nuclear fuel is now stored in the unit's spent fuel storage pool (Appendix H, Figures 1 and 2). Currently, the decay heat generation is so low that the fuel pool coolers are not used. During the review of Humboldt Bay's Decommissioning Plan, the NRC staff requested that a plant-specific calculation be performed to demonstrate that the spent fuel will not be damaged in the case of loss of water from the spent fuel storage pool (Ref. 1). This appendix summarizes the spent fuel heatup analyses (Ref. 4 and 6) under this scenario to show that the Humboldt Bay spent fuel cladding, storage rack (Appendix H, Figures 3 and 4), and Boral cans will remain intact in the unlikely event of loss of water from the spent fuel storage pool.

Humboldt Bay's spent fuel storage pool is below ground level. The lowest ground water level is about 7 feet above the top of the fuel. A spent fuel storage pool leak to the suppression chamber, however, could potentially result in complete drainage of the pool. A rupture of the spent fuel storage pool drain line could potentially result in a partial draindown, though it is considered very unlikely (Ref. 3). Low spent fuel storage pool water level and high radiation alarms are available to warn the operators of any abnormal conditions, allowing action to be taken early enough to avoid complete loss of water from the pool.

Some fuel assemblies at Humboldt Bay have channels. Each fuel assembly is housed within a Boral can (Appendix H, Figure 6) with the exception of damaged fuel assembly UD-6N, which is housed in an aluminum box. Currently in the spent fuel storage pool there are three types of fuel assemblies. All fuel assemblies have Zircaloy cladding. The fuel data are shown in Table B-1.

Postulating a draindown of the spent fuel storage pool, the spent fuel assemblies will heat up and will eventually reach a steady-state temperature when the decay heat generation rate is balanced by the heat loss out of the pool. If the decay heat generation rate is sufficiently high relative to the heat loss, the cladding may reach a high enough temperature to melt the storage rack or Boral cans or, if high enough, may cause the clad to rupture as a result of internal pressure or to undergo rapid exothermic oxidation (i.e., zirc fire) leading to clad melting. It has been postulated that a partial draindown of the spent fuel storage pool, where the inlet to the assemblies remains covered, would create a worst-case scenario relative to fuel heatup (Ref. 2).

A spent fuel heatup analysis (Ref. 4 and 6) has been performed by PG&E. Section 2 of this Appendix describes the analysis approach, results, and conclusions. The Humboldt Bay fuel rods have a deposit of iron oxides giving them a fairly uniform reddish color. The effect of the iron oxide deposits on fuel heatup in a postulated spent fuel storage pool draindown accident is less than 1 ^{o}C (Ref. 2) and, therefore, was not taken into account in the analyses.

2. HBPP SPENT FUEL HEATUP ANALYSES

A. General Case

Approach

PG&E performed a conservative site-specific spent fuel heatup analysis for a loss of water from the fuel storage pool (Ref. 4) with input from the results of a site-specific spent fuel decay heat analysis (Ref. 5). As the fuel decays over time, the results of the spent fuel heatup analysis become more conservative.

The analysis was based upon guidance in NUREG/CR-0649 (Ref. 2) and the SFUEL computer code (contained in this NUREG). The intent of this NUREG is to assess the effect of decay time, fuel element design, storage rack design, packing density, room ventilation, and other variables on the heatup characteristics of the spent fuel and to predict the conditions under which clad failure will occur. The likelihood of clad failure due to rupture or melting is dependent on the spent fuel decay time period and the storage configuration.

The heat removal analysis for the drained spent fuel storage pool is considered in two parts: (1) the heat transfer from the spent fuel assemblies within spent fuel storage pool, and (2) the removal of heat from the refueling building.

Heat produced by decay within the spent fuel assemblies and by chemical oxidation of the clad is removed, in part, by buoyancy-driven airflows circulating in the open channels. Calculation of this airflow uses an iterative, finite-difference solution of conservation equations. Transient conduction equations are solved in the axial direction of the fuel element to determine the heatup of the fuel rod. Radiation heat transfer between structural elements is accounted for, as is transient conduction into the pool's concrete encasement.

The refueling building air temperature will increase as a result of the natural convection heat transfer process from the fuel in the pool. The SFUEL code computes the amount of heat that is removed by a combination of forced ventilation (in this case 'none' since it is conservatively assumed there is no active ventilation), leakage of air through the building structure (in this case no leakage was conservatively assumed), heat storage by the structural heat sinks, and radiation/ natural convection from the building exterior to the outside.

Results and Conclusions

As would be expected with the long decay time for the Humboldt Bay fuel (i.e., from 1976), fuel heat up as a result of storage pool draindown is insignificant.

The results of the spent fuel heatup analysis can be summarized as follows:

- Maximum fuel temperature occurs in the postulated partial draindown condition where water level falls to just above the opening near the bottom of the Boral cans thereby blocking airflow in the fuel assemblies.

- Maximum fuel temperature for a postulated partial draindown accident at HBPP is 217° C.

As discussed in Reference 7, the onset of rapid oxidation of Zirconium (cladding) is 800° C. A cladding temperature of 570° C is used as a thermal limit under accident conditions for spent fuel dry storage casks (Ref. 8). The melting temperature of aluminum (the rack material and limiting constituent of the Boral cans) is approximately 640° C. The maximum fuel temperature, therefore, is approximately 353° C below the most limiting of these temperatures. Therefore, uncertainties related to the usage of the SFUEL code as discussed in Reference 7 are bounded and do not affect the conclusion that the fuel heatup, as a result of a postulated pool draindown, is insignificant.

B. Fuel Assembly UD-6N Case

Fuel assembly UD-6N is damaged and is housed in an aluminum box in the spent fuel storage pool. A separate specific fuel heatup analysis (Ref. 6) was performed for this unique case based on the decay heat of this particular fuel assembly.

Approach

A conservative simplified bounding analysis was performed (Ref. 6) that only credits conduction (through air) and radiation heat transfer. The spent fuel storage pool is conservatively assumed to be completely drained, since complete drainage in this case will result in the poorest heat transfer and, thus, highest fuel temperature. The fuel rods are modeled as square rods with equivalent circumference (identical heat transfer area). The pitch between the rods is varied in the analysis to determine the worst case. Meanwhile, the "bundle" of rods is assumed to be centered in the storage box to minimize heat transfer between the rods and the box. The heat transfer rate from the outer surface of the aluminum wall to the ambient air is calculated based on natural convection. Since one side of the aluminum box is very close to the spent fuel pool wall, the natural convection heat transfer is assumed for only three surfaces of the box. Each fuel rod is assumed to have a uniform temperature.

Results and Conclusions

As a result of a postulated complete draindown of the spent fuel storage pool, the maximum fuel temperature for assembly UD-6N was determined to be approximately 262°C, which is more limiting than the General Case described above in Section A of this Appendix. The maximum fuel temperature for this assembly is approximately 308° C below the most limiting of the temperature limits (cladding temperature of 570° C thermal limit under accident conditions for spent fuel dry storage casks (Ref. 8)). The fuel heatup in the assembly UD-6N specific case is still well below the most limiting temperature limit. The conclusion, therefore, that fuel heatup is insignificant remains valid for a postulated spent fuel storage pool draindown.

3. REFERENCES

1. NRC Letter, J. A. Zwolinski to J. D. Shiffer, dated January 23, 1985, "Request for Additional Information," Question No. 84, Docket No. 50-133, LS05-85-01-021.

2. "Spent Fuel Heatup Following Loss of Water During Storage," NUREG/CR-0649, SAND77-1271, March 1979.

3. NRC Inspection Report 50-133/94-02 – Humboldt Bay Power Plant Unit 3 (May 9, 1994)

4. PG&E Calculation N-269, Rev. 0, "Spent Fuel Heatup with Spent Fuel Pool Water Drained"

5. PG&E Calculation N-270, Rev. 0, "Spent Fuel Decay Heat Analysis"

6. PG&E Calculation N-271, Rev. 0, "Maximum Fuel Cladding Temperature (UD-6N)"

7. Nuclear Regulatory Commission, " Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, October 2000".

8. "Standard Review Plan for spent Fuel Dry Storage Facilities," NUREG-1567, March 2000

Table B-1

HUMBOLDT BAY FUEL ASSEMBLY DATA

	Type II	Type III	Type IV
Rod array	7 x 7	6 x 6	6 x 6
Active fuel height (in.)	79	77.5	77.125
Rod pitch (in.)	0.631	0.74	0.74
Rod OD (in.)	0.486	0.563	0.5625
Pitch/OD ratio	1.298	1.314	1.315
Cladding thickness (in.)	0.033	0.032	0.035
UO_2 weight/assembly (lbs)	192	191	181
Assemblies per core	172	172	172
Channel inside dimension (in.)	4.542	4.542	4.542
Net flow area per assembly (in.^{2})	11.54	11.67	11.68
Number of fuel assemblies currently in the spent fuel storage pool	88	179	123

APPENDIX C

Pacific Gas and Electric Company
Humboldt Bay Power Plant Unit 3

Criticality Analysis
for SAFSTOR Decommissioning

Rev 5
March 2004

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A. INTRODUCTION

Item 75 of NRC letter dated January 23, 1985, requested the following information:
"Discuss the likelihood of a reactivity accident in the spent fuel storage pool due to heavy load drop or seismic event. If sufficient likelihood ($<10^{-6}$ per year) of such events exists, then, assuming step and/or ramp reactivity insertions in the stored spent array due to reduction in undermoderation of stored fuel in the pool, in turn due to fuel reconfiguration initiated by a heavy load drop or strong seismic event, calculate offsite radiological consequences assuming:

a) upward spray of all pool water without the presence of the building roof, and
b) pool boiling without spray and without the presence of the building roof."

PG&E's response dated February 28, 1985, stated the following: "PG&E is actively evaluating design alternatives that would prevent possible criticality due to seismic and heavy load events." This report provides a complete response to Item 75.

This report describes the design, fabrication, and safety analysis performed for the addition of neutron-absorbing material in the Humboldt Bay Power Plant (HBPP) Unit 3 spent fuel storage pool. The purpose of the modification is to ensure subcriticality following any event, which results in a rearrangement of fuel assemblies from the existing criticality safe storage rack configurations.

This modification consists of enclosing each fuel assembly (with the exception of UD-6N) in a can fabricated from a neutron-absorbing material, so that a k-effective greater than 0.95 cannot be achieved for any possible fuel configuration.

The criticality analysis associated with this project was prepared by an outside consultant for Pacific Gas and Electric Company (PG&E). The analysis shows that up to two assemblies in addition to UD-6N may be outside of the Boral cans and ensure subcriticality for any possible configuration of the fuel.

B. OVERALL DESCRIPTION

1. Existing Rack Configuration

The HBPP spent fuel storage racks have a total capacity of 486 fuel assemblies. This includes 351 central pool locations in 88 groups of 4*, and 135 peripheral pool locations in 45 groups of 3. The central racks are designed to individually support each fuel assembly. The peripheral racks support fuel assemblies in groups of three.

The central storage racks (Appendix H, Figure 5) are constructed of aluminum and consist of pairs of storage units approximately 5 feet high and 12 inches square. Each storage unit is able to hold four fuel assemblies. The peripheral racks are similarly constructed except that they can hold either three fuel assemblies or one full fuel storage can.

* (However, pool location 64-07 cannot be used due to a bolt protruding into the bottom of this location and inadvertent use of this location is prevented by a triangular plate welded over the top.)

The fuel storage racks are welded and/or bolted to cross members of aluminum channels. The fuel storage racks are spaced to be "criticality safe."

There are currently 390 irradiated fuel assemblies in the HBPP spent fuel storage pool, with exposures ranging from 1,307 to 22,876 MWD/MTU.

2. Implemented Modifications

In order to preclude criticality in the spent fuel storage pool following an event which results in movement or damage to the fuel assembly storage racks, each fuel assembly (except for assembly UD-6N, which is in a separate can, and cannot be enclosed in a Boral can) is enclosed in a can fabricated from a neutron-absorbing material. The can contains an areal density (0.005 gm/cm^2) of boron (^{10}B) such that a k-effective greater than 0.95 cannot be achieved for any possible configuration.

A drawing of the can is shown in Appendix H, Figure 6. The walls of the can are fabricated from Boral™. Bands are attached at the top and bottom of the can to provide structural strength. The band attached to the bottom of the can is designed to prevent the fuel assembly from coming out of the bottom, and still allow for convection cooling. The top band is fabricated with locking tabs, which are bent over to prevent inadvertent removal of the fuel assembly from the can. This design will ensure that the poisoned material is an integral part of the fuel assembly. The tabs can be bent and unbent as required to remove the fuel assembly for inspection and other similar activities. There is also a spare set of tabs that can be used should the other set be damaged.

C. MATERIAL CONSIDERATIONS

Most of the material used in fabrication of the fuel bundle enclosure can is Boral, which is a thermal neutron poison material composed of boron carbide and 1100-alloy aluminum. Boron carbide is a compound having high boron content in a physically stable and chemically inert form. The 1100-alloy aluminum is a lightweight metal with high tensile strength, which is protected from corrosion by a highly resistant oxide film. The boron carbide and aluminum are chemically compatible and suited for long-term use in the radiation, thermal, and chemical environment of the HBPP spent fuel storage pool.

The Boral is provided in flat sheets and is formed to enclose the full length of each of the four sides of each individual fuel assembly. Physical integrity of the poisoned can is maintained by use of stainless steel bands, which are attached to the Boral with rivets and encircle the can at the bottom and the top.[1]

The materials contained in the Boral, as well as the stainless steel, are compatible with all parts of the spent fuel storage system, including the fuel assemblies, the cooling system, the cleanup system, the pool liner, and the storage racks. The useful life of the Boral will exceed 40 years when in contact with the storage pool water. The corrosion resistance of Boral is provided by the protective film on the aluminum cladding that is an integral part of the Boral panels. Testing performed by the Boral supplier confirms that the effects are negligible from general corrosion, galvanic corrosion of the Boral/stainless steel interface, pitting corrosion, stress corrosion, and intergranular corrosion.

[1] Revised as per Safety Evaluation Number 2000-10.

Boral is manufactured under the control and surveillance of a computer-aided quality assurance/quality control program that conforms to the requirements of 10 CFR 50, Appendix B, entitled "Quality Assurance Criteria for Nuclear Power Plants."

Boral has been licensed by the USNRC for use in BWR and PWR spent fuel storage racks, and is also used around the world for spent fuel shipping and storage containers. Table C-1 is a listing of high density rack experience as of 1985.

Table C-1

SUMMARY OF GENERAL ELECTRIC HIGH DENSITY FUEL STORAGE RACK EXPERIENCE

Plant	Scope of Work	Status
Monticello	13 racks, storage capacity 2,237 spaces	Licensed and in use since April 1978
Browns Ferry 1, 2, and 3	57 racks, storage capacity 10,413 spaces	Licensed and in use since Sept. 1978
Hatch 1 and 2	30 racks, storage capacity 6,026 spaces	Licensed and in use since April 1980
Brunswick 1 and 2	10 racks, storage capacity 3,642	Licensed and in use December 1983
Hartsville A1, A2, B1, B2	60 racks, storage capacity 11,804 spaces (Plant canceled)	Approved for installation through GESAR II FDA July 1983
Phipps Bend 1 and 2	30 racks, storage capacity 5,902 spaces (Plant cancelled)	Approved for installation through GESAR II FDA July 1983
Kuosheng 1 and 2	6 racks, storage capacity 1,326 spaces	Scheduled for 1985 installation

D. CRITICALITY ANALYSIS

1. OVERVIEW

The purpose of this analysis is to demonstrate that spent fuel pool (SFP) evolutions at the Humboldt Bay Power Plant, Unit 3, will not result in a criticality event. These evolutions may include the removal of up to two spent fuel assemblies from their neutron absorbing Boral cans. The two assemblies are in addition to fuel assembly UD-6N, which cannot be stored in a Boral can. The analysis addresses the unrestricted movement of two fuel assemblies to any location of the fuel pool for purposes of inspection and/or reshuffling to different locations. To encompass all credible situations, normal conditions as well as off-normal events and accidents will be analyzed. This analysis is documented in calculation N-265 (Ref. 18).

The analysis will show that k-effective (k_{eff}) for the spent fuel pool will not exceed 0.95, with a 95% probability and a 95% confidence limit (95/95), as per NRC design criteria. That is, there is a 95% probability that the calculations bound 95% of the possible outcomes.

The criticality analysis is performed with the SCALE 4.4 code package (Ref. 1) from Oak Ridge National Laboratories (ORNL). This code package was procured from ORNL on a single CD-ROM. The package was validated in accordance with ORNL instructions (Ref. 2). A benchmark study with data from known critical experiments was done to determine a code bias and the results are attached in Attachment A to this document.

2. ASSUMPTIONS AND CONSERVATISMS

There are several sources of conservatism involved in the modeling of the SFP. These include SFP geometry, fuel assembly arrangement, the poison can dimensions, fuel pin arrangement within the fuel assemblies, and the burnup of the fuel.

The spent fuel pool currently contains 390 exposed fuel assemblies of mixed General Electric and Exxon Nuclear Corporation design. These are GE Type II and Type III, and Exxon Type III and Type IV (Ref. 4). For the purposes of this analysis, all elements were modeled as GE Type III (a 6 x 6 pin assembly). The pool consists of a rectangular pool of water lined by a stainless steel liner enclosed by concrete at least 24 inches thick on the bottom and sides. The pool provides coverage of the fuel assemblies to a depth of 26 feet (Ref. 14) or to an approximate depth of 17 feet above the storage racks.

In 1985, PG&E installed neutron-absorbing cans on each fuel assembly (except UD-6N, which is stored in a separate box and would not physically accommodate a can). The purpose of the can was to maintain the shutdown margin of the SFP in the event of a loss of normal assembly separation due to a seismic or other event. With the absorber cans in place under normal conditions the SFP is maintained in a highly sub-critical condition. The water between the storage units functions to thermalize fast neutrons. A flux trap is created because the Boral is highly efficient at absorbing thermal neutrons and thus prevents the neutrons from going back into the storage units containing the fuel.

2.1 SFP Geometry and Fuel Assembly Arrangement

The current HBPP SFP design consists primarily of racks with storage units of four cells (2x2 arrays) held in place by aluminum "T" beams (Ref. 5). Two storage units are mounted on a rack, at a pitch of 18". The racks are placed in the pool to maintain at least an 18" pitch throughout the spent fuel pool.

The SFP has been modeled as a rectangular array of 9 columns of racks arrayed in six rows, all of whose locations are assumed to be occupied (9 x 6 x 8 or 432 cells). In reality, several locations are empty, including an area used for fuel assembly hardware storage near the center of the pool. The racks on the periphery only accept three assemblies per storage unit. A conservative assumption of denser packing of fuel and a higher assembly loading (432 vs. 390 assemblies, with no empty storage locations) was made in the analysis.

To simulate the effects of infinite length fuel assemblies (with higher k_{eff}, since there is no axial neutron leakage) and an infinitely large Spent Fuel Pool, the boundary conditions in KENO-Va were set to *perfectly reflective using the* "**ALL=SPECULAR**" *statement.*

This conservative assumption is similar to that made in the earlier CASMO-2E analysis (Ref. 6 & 12). This provides the most conservative geometry because with the infinite length fuel rod, all neutron leakage is into the pool and because of the infinitely large pool, all neutrons will be absorbed within the confines of the pool thereby conservatively accounting for the maximum number of neutronic reactions.

The most reactive condition that would normally exist in the pool is with the largest possible number of assemblies without poison cans (assumed to be two) plus the assembly UD-6N. For this analysis, assembly UD-6N is assumed to be as reactive as a regular fresh assembly without a poison absorber. The fuel pins may be in an altered geometry, but random pin rearrangement is less reactive due to non-uniform moderation versus a lattice with an optimized pin pitch.

The grids that provide structural support to the fuel assembly were not modeled. A zero grid assumption is conservative since the grids displace moderator and have some neutron absorption.

2.2 Fuel Rods

The current model assumes that all fuel contains fresh 2.52 wt% ^{235}U and is free from fission product poisons and burnable absorbers. This is a very conservative assumption since fuel depletion reduces the amount of fissionable material, some fuel rods contain gadolinium burnable absorber and ^{149}Sm produced during power operation is a neutron absorber.

All the fuel rods were assumed to have the same ^{235}U enrichment as the highest average assembly enrichment. Previous CASMO simulations (Ref. 12) have shown this to be conservative. All of the fuel in the SFP was exposed to at least one in-core operating cycle. The minimum batch average burnup is *approximately 5000 MWD/MTU for batch XC fuel. The total fissile enrichment of these assemblies is* 1.90 wt% ^{235}U and 0.21 wt% fissile Pu. The reactivity of this fuel is significantly less than the fresh fuel assumed to be located in every cell of SFP in the current model.

2.3 Boral Can Design

The SAFSTOR technical specifications require that all fuel assemblies be stored in Boral containers.

The active absorber portion of the Boral cans extends approximately 4-6 inches above the top of the active fuel to the top of the fuel plenum region, causing an additional "flux trap" region above the top of the active fuel. To a lesser extent this situation also exists at the bottom of the active fuel. This design impact results in significant neutron absorption in a region where the greatest moderation and reflection of thermal and epithermal neutron flux would otherwise occur. This greatly reduces the effectiveness of the water above the assemblies as a reflector of neutrons, resulting in a higher neutron leakage at the top and bottom of the fuel. Although this region could be modeled using KENO Va, for conservative purposes the fuel assemblies are considered to have specular reflection conditions (maximum) at both the top and bottom, corresponding to fuel assemblies of infinite length as noted in Section 2.1 above.

2.4 Channels

BWR fuel assemblies normally have a 60 mil thick Zircaloy-2 channel. Fuel inspection activities have resulted in the removal of most of the channels. The presence of channels attached to the fuel was not modeled since calculations indicate that this is the more conservative condition. While Zircaloy-2 is not a significant neutron absorber, it replaces an equal volume of water inside the Boral containers that would otherwise be more effective in moderating and reflecting neutrons back into the fuel pin array.

3. DESIGN DATA

All dimensions in inches unless noted

3.1 Fuel Rod Data (all from Ref. 4)

Active Length, GE Type III	77.5
Rod Pitch	0.740
Fuel pellet Diameter	0.488
Fuel Rod O.D.	0.563
Fuel Rod Cladding Thickness	0.032± 0.003
Fuel Cladding Material	Zr-2
Average Enrichment	2.50 wt%

3.2 Fuel Assembly Data (all from Ref. 4)

Number of Active Fuel Rods per Assembly	36
Assembly Width	4.70
Fuel Channel Material	Zr-2
Channel Thickness	60 mils±0.025
Channel O.D.	4.70

3.3 Spent Fuel Pool Rack Data (all from Ref. 10)

Number Cells per Storage Unit	4
Number of Storage Units per Rack	2
Fuel Rack Material and Construction	Aluminum (6061-T6)
Height of Racks	59.75
Inner Box Dimension of Storage Units	11.375

3.4 Poison Can Data (all from Ref. 16)

Length of Neutron Absorber	92
Can O.D./I.D	5.0/4.8
Poison Can Material	Boral
Boron-10 content	5 mg/cm^2
Boral Sheet Thickness	0.075 to 0.100

4. METHODOLOGY

4.1 Calculation of K_{eff} and Uncertainties

Reference 13 provides a methodology for calculating k_{eff} and accounting for uncertainties in the values:

$K_{eff\text{-}total} \leq 0.95$

Where $k_{eff\text{-}total}$ represents the system k_{eff} including all statistical uncertainties and effects of mechanical tolerances, in accordance with Reference 15 & 17:

$$K_{eff} + \Delta k + B_m + \Delta B_m + (\Sigma \Delta K (\text{tolerances})^2)^{1/2} < 0.95$$

Where:

K_{eff} is the eigenvalue from KENO
Δk is uncertainty addition due to bounding conditions (e.g. pool temperature)
B_m = method bias from KENO benchmark cases (Attachment A)
ΔB_m is the uncertainty in the method bias, also obtained from the statistical analysis of the benchmark runs (Attachment A)

$(\Sigma \Delta K (\text{tolerances})^2)^{1/2}$ is the root mean-square (rms) value of uncertainties in the K_{eff} that are due to physical parameters (enrichment, pellet density and mechanical tolerances such as cladding thickness, etc. below) and calculational uncertainty.

The following are physical parameter uncertainties considered in this calculation:

- Fuel enrichment
- Fuel density
- Clad thickness uncertainty
- Fuel rod pitch uncertainty
- Channel thickness uncertainty
- Fuel assembly pitch uncertainty (within a cell)
- Cell and storage unit pitch tolerances
- Temperature swings in the SFP
- Moderator Intrusion into damaged fuel rod gaps

Except temperature, all of these uncertainties are uncorrelated with each other and can therefore be added in quadrature (i.e. the square-root of the sum of each squared uncertainty). If not, then the corresponding Δk must be added arithmetically to the KENO eigenvalue. The KENO calculational uncertainty for each parameter uncertainty case is added statistically in the results to create a 95% confidence level. The base KENO calculational uncertainty is included also. Including these statistical calculational uncertainties assures the results will have a 95 percent probability at a 95 percent confidence level of being below the criticality limit of 0.95.

4.2 Normal Scenarios

The baseline case was used to analyze reactivity additions due to the tolerances and changes listed in Section 4.1 above. Fuel rods and assembly parameters were assumed to have base-line values as listed above with the exception of:

Fuel pellet density = 10.33 g/cc (94.3% of theoretical density vs. 10.30 g/cc (Ref. 4))
Enrichment = 2.52 wt% (vs. 2.50 wt% listed (Ref. 4))
Fuel rack height of 55" (vs. 59.75" (Ref. 10))
Boral wall thickness of 0.075" (vs. 0.100" (Ref. 16))
Pin Pitch = Optimal versus .740" (optimized to 100% of maximum in KENO runs)

Higher fuel pellet density and enrichment were used because the nominal values listed are usually "guaranteed minimum values" by the fuel vendor, and the "as built" enrichment and pellet densities are usually slightly higher.

For the fuel assembly itself, computer runs at different fuel pitches were made to determine an optimum fuel pitch within the neutron absorber can. This was done because of the history of fuel pins being removed for hot-cell testing by GE (potentially changing the lattice) and the potential warping of the rod array within the absorber container to produce a more reactive cell. The most reactive pitch for a 5 in. width (outer dimension) Boral container was found to be 100% of the maximum pitch. Thus, the four corner pins are pushed against the corners of the Boral can to maximize the moderator volume between them.

The Boral container itself was assumed to be 75 mils instead of 100 mils thick since it is more conservative to assume a thinner wall thickness. The thinner wall displaces less moderator and is more reactive.

The fuel assembly was assumed to be un-channeled, centered in the absorber container and at its nearest distance to cell neighbors given the storage unit dimensions (0.375" nearest neighbor spacing).

The rack height was assumed to be slightly less than the as-built value (adding reactivity) to account for slightly different rack heights in the active fuel regions of different fuel designs.

Since a fuel assembly without an absorber container is more reactive than one in a container, it was assumed that the maximum number of assemblies without a container (three, one of which is UD-6N) were in close proximity to one another that is, in the same 4-assembly storage unit.

This storage unit was then placed in different positions in the SFP to find the optimal location. It was found that a non-corner edge storage unit location was the most reactive, as shown below.

Optimal SFP configuration of Storage Rack

Edge of Pool

X	O
X	X

O	O
O	O

Where **X** denotes an assembly with no absorber container
O denotes an assembly in a Boral container

Using the assumptions and input discussed above the baseline case was created. The pool configuration consists of an entire pool of fuel in Boral cans with the exception of one storage unit containing three assemblies without Boral cans.

Case Description	K_{eff}
Base case	0.81119

4.3 Off Normal Conditions

4.3.1 Reactivity Addition Due to Loss of Normal Temperature Control.

At HBPP the heat load of the SFP is so minimal that conduction and surface convection alone is sufficient to maintain temperature at about 72 degrees F. At the same time, the decay heat production of the fuel is sufficient to maintain water temperature above 32 degrees F even if the plant experienced a prolonged period of cold weather.

To model the reactivity effects of a loss of temperature control, pool and fuel temperature was assumed to be initially 80.6 F (27 C) at baseline. Pool and fuel temperatures were then assumed to rise by an additional 48.6 F (27 C) to 129.2 F (54 C).

For a corresponding drop in SFP temperature, SFP and fuel temperatures were assumed to fall to 32 F (0 C). The methods used bound the maximum water density conditions that occur when water is at 39.2 F (4 C). Since the pool has a negative moderator temperature coefficient (MTC), lowering the pool temperature introduces positive reactivity as shown below for the normal case:

Description	K_{eff}
Baseline k_{eff} (80.6 F)	0.81119
Minimum Pool Temp (32 F) k_{eff}	0.81252
Δk	+0.00133

A similar calculation was performed for the accident case and the result is listed in Section 6.2.

4.3.2 Fuel Assembly Mis-loading

A mis-loading of a fuel assembly at the HBPP SFP cannot occur since the pool does not have separate fuel regions with prescriptive restrictions on fuel placement.

4.4 <u>Accident Conditions</u>

4.4.1 Loss of Spent Fuel Pool Coolant

The scenario of partial to complete uncovering of the fuel was investigated. As expected, uncovering of the fuel leads to a less, rather than a more reactive condition owing to the loss of effective neutron moderation within the fuel storage cell. This condition of reduced moderation more than compensates for the reduced thermal neutron absorption by the boron cans. The core k_{eff} decreases continuously, starting with the loss of the water as reflector when the level reaches the top of active fuel (TAF):

Effect on K_{eff} of Spent Fuel Uncovering

Amount of core uncovering	K_{eff}
None (Base Case--SFP full)	0.81119
None (Level at TAF)	0.78312
100% Uncovered	0.52639
Δk_{eff}	-0.28480

4.4.2 Loss of Normal SFP Storage Geometry

Considerable positive reactivity is added assuming that the water gaps between the storage units and the cells are lost. This is different from the loss of coolant situation since water is assumed to remain in the storage cell for this condition. When the Boral cans are touching (no flux trap water gap), the effectiveness of the boron in the Boral, while still present, is reduced.

The base case was for an optimized assembly pitch with all assemblies in neutron absorbing containers, as discussed above. In accordance with the previous CASMO (Ref. 12) analysis and the discussion in the NRC's review of SFP "compression events" (Ref. 7) the inter-assembly space was assumed to be freely variable from the base case spacing down to zero.

Case Description	K_{eff}
Base rod spacing, 18-20" storage unit pitch	0.81119
All racks collapsed, 0" between all assemblies	0.88094
Δk_{eff}	+0.06975

A few cases were also examined to consider asymmetrical spacing of the fuel without Boral cans. The reactivity impact was very small and negative indicating that the geometry assumptions of this case are relatively conservative.

4.4.3 Dropped Fuel Assembly Accident

It is assumed that two assemblies without a Boral can are in the storage unit side by side and that a third assembly drops either in a neighboring water gap or directly onto the other two assemblies. The resultant configuration is conservatively bounded by modeling the fuel rods of the 3 fuel assemblies (108 fuel rods combined) occupying an optimal pitch centered in two storage locations.

Case Description	K_{eff}
Base cell spacing, .375" between assemblies	0.81119
Extra assembly forms 12 x 9 array	0.76500
Δk_{eff}	-0.04619

Interestingly, this case is less reactive than the base case because the array of three assemblies cannot arrange into as reactive a geometry as if they were in three fuel assembly locations in the storage module. The new lattice is under-moderated.

The base case and the loss of spent fuel pool geometry case already bound an assembly falling to a vertical position beside a storage unit. An assembly falling to a horizontal position on top of the storage unit is bounded by the infinite axial length assumption used in the model.

5. COMPUTATIONS

The following computations are for the normal case. Similar computations were performed for the worst accident case and the results are listed in Section 6.2.

5.1 Effects of Enrichment Uncertainties

The assembly average enrichment of the GE Type III assembly was specified to be 2.50 wt% (Ref. 4). Although a tolerance for this was not specified, the average enrichment typically has an uncertainty of + 0.25-0.5 %. Because of the age of the fuel, a conservative 2 sigma value of 2% over nominal (or 1.02 x 2.50 = 2.55 wt%) was used. The base case used 2.52 wt% because the nominal enrichment is usually a guaranteed minimum value from the vendor and is usually slightly higher in the as-built fuel.

Case Description	K_{eff}
Base case, 2.52 wt%	0.81119
Maximum enrichment, 2.55 wt%	0.81486
Δk_{eff}	0.00367

5.2 Effects of Fuel Pellet Density Tolerance

Fuel pellets are sintered UO_2 with a nominal density of 10.3 $\pm$ 0.2 g/cc (Ref. 4). Because of

manufactured-in pellet "voids" (dished pellet end and edge chamfering to reduce pellet-cladding interactions), it is actually difficult to achieve a density of greater than about 10.5 (+1σ value). As with fuel enrichment, the nominal density is usually a minimum value guaranteed by the fuel vendor, so a slightly higher value of 10.33 was used as the base case

Case Description	K_{eff}
Base Case, 10.33 Density	0.81119
Max Density, 10.5 g/cc	0.81454
Δk_{eff}	+0.00335

5.3 Effects of Clad Thickness Tolerance/Uncertainty

The cladding has a nominal thickness of 0.032 + 0.003 in. KENO Va runs were done to establish the optimal cladding thickness relative to nominal, as shown in the table below.

Case Description	K_{eff}
Base Case, 0.032" clad	0.81119
Min clad thickness, .029"	0.81385
Max clad thickness, .035"	0.81032
Δk at .029 clad thickness	+0.00266

5.4 Effects of Fuel Rod Pitch (within an assembly)

Using the minimum assembly separation of 0.375" as discussed above; the pitch was varied within the assembly to determine the most reactive SFP k_{eff}. As shown in the table below, the most reactive configuration for normal assembly spacing was the 100% of maximum pitch case.

Case Description	K_{eff}
Base Case, Maximum Pitch	0.81119
98% of max pitch	0.80810
96% of max pitch	0.79947
Δk_{eff}	-0.00309

5.5 Effects of Channel Wall Thickness

The HBPP channels are made of Zircaloy, an alloy having low neutron absorption cross-section. The effect of the channel on the fuel assembly neutronics is essentially to take up space that would otherwise be occupied by moderator as discussed in Section 2.4.

There is a small negative reactivity resulting from a channel being present. Because the random eigenvalue deviations obscure the small reactivity effects, a least squares analysis was done using different channel thickness. The reactivity increase of not having a channel was determined to be .00072 (using the least squares fit line difference between 0 and 60 mil channels). Thus, the case without a channel is a conservative configuration and no uncertainty bias is required.

5.6 Fuel Assembly Pitch Uncertainty (within a cell)

The aluminum rack members have a 0.0625" (Ref. 10) tolerance in thickness. Because the fuel assemblies could be slightly cocked within the location, this tolerance was doubled to account for the cocking effect. Base case was for a .375" thick aluminum member, and cases were run with 0.25" and 0.5 " thick members (and therefore, assembly spacing) to represent the effects on minimum and maximum tolerances.

Case Description	K_{eff}
Base Case, 0.375" spacing	0.81119
Minimum spacing, 0.25"	0.80993
Maximum spacing, 0.5"	0.81103
Δk_{eff} rel to Base Case	-0.00126

5.7 Effects of Tolerances of the Fuel Storage Unit Pitch

Based on the stated tolerances on the spacing between adjacent 2 x 2 storage units (.0625", (Ref. 10)), the storage unit pitch was decreased by .125" (2σ). The effect on k_{eff} is shown in the table below. As expected, moving the storage units closer introduces positive reactivity:

Case Description	K_{eff}
Base Case, 18-20" spacing	0.81119
Minimum spacing	0.81256
Δk_{eff}	+0.00137

5.8 Effects of Moderator Intrusion into Damaged Fuel Rod Gaps

The presence of some failed fuel (Ref: 4) leads to the possibility that a significant number of the fuel rods have some water intrusion into the gap volume of the rods, at least the older, higher burnup assemblies. Although uranium pellet cracking/swelling may have eliminated these gaps, this condition also allows water impregnation of the pellet itself.

For sensitivity and uncertainty calculations the entire gap region of all rods is assumed to be 100% full of water and the moderator density is adjusted in the range from 0.01 to 1.0 to give maximum reactivity addition.

Case Description	K_{eff}
Base Case, (no mod in gaps)	0.81119
100% density H_2O in gap	0.81609
Δk	+0.00490

5.9 Effects of Boral Absorber Container Thickness on Reactivity

Assuming that the base boron content is always present, the reactivity effect of the different thicknesses at maximum tolerances was studied.

The 100 mil-container is assumed to be at a maximum tolerance of 0.106 in (from Section 3 above). The 75 mil container is assumed to be at a minimum tolerance of 0.70 (Ref. 11).

Case Description	K_{eff}
Base Case, (75 mil container at min tolerance)	0.81119
100 mil container at maximum tolerance	0.80629
Δk	-0.00490

Thus, the base case is already a more reactive configuration than one with a thicker Boral container (for the same B^{10} loading).

6. RESULTS AND CONCLUSIONS

6.1 Normal Case

Description: Optimal pin pitch (maximum for 5 inch container), 2.52 wt%, density =10.33 g/cc, normal moderator density

Nominal KENO Va Reference K_{eff}:		0.81119
Calculational & Methodology Biases:		
Methodology (benchmark) bias, (App. A)		0.00684
Methodology bias uncertainty 95/95	(App. A)	0.00386
Pool Temperature Bias (Section 4.3.1)		0.00432*
Total Biases		0.01502
Tolerances and Uncertainties:		
Fuel Enrichment tolerance (+0.03, -0.0 wt% ^{235}U)		0.00708*
UO_2 density tolerance (+0.2, -0.0 g/cc)		0.00645*
Clad wall thickness (±0.003 in)		0.00584*
Asymmetric Assembly positioning (optima1 pitch)		0.00000
Storage unit pitch (± 0.25 in)		0.00458*
Moderator Intrusion		0.00797*
Calculational Uncertainty		0.00298
Total Uncertainty (Statistical)		0.01480
Final 95/95 k_{eff} Including Uncertainties and Tolerances		0.84101

*Values have been increased to a 95% confidence level by adding the square root of the sum of the squared standard deviations and multiplying by 1.65, the one-sided 95% confidence multiplier.

6.2 Worst Accident Case

Description: Optimal pin pitch (maximum for 5 inch container), 2.52 wt%, density =10.33 g/cc, normal moderator density. Loss of inter-assembly spacing.

Nominal KENO Va Reference K_{eff}:		0.88094
Calculational & Methodology Biases:		
Methodology (benchmark) bias (App. A)		0.00684
Methodology bias uncertainty 95/95	(App. A)	0.00386
Pool Temperature Bias (Section 4.3.1)		0.01084*
Total Bias		0.02154
Tolerances and Uncertainties:		
Fuel Enrichment tolerance (+0.03, -0.0 wt% ^{235}U)		0.00408*
UO_2 density tolerance (+0.2, -0.0 g/cc)		0.00527*

Clad wall thickness (±0.003 in)	0.00764*
Asymmetric Assembly positioning (optimal pitch)	0.00000
Storage unit pitch (collapsed)	0.00000
Moderator Intrusion	0.00525*
Calculational Uncertainty	0.00176
Total Uncertainty (Statistical)	0.01156
Final 95/95 k_{eff} Including Uncertainties and Tolerances	0.91404

*Values have been increased to a 95% confidence level by adding the square root of the sum of the squared standard deviations and multiplying by 1.65, the one-sided 95% confidence multiplier.

In conclusion, these calculations demonstrate that up to two Boral cans may be removed from the fuel without exceeding the K-effective criticality limit of 0.95, based on conservative calculations of K-effective that show significant margin to the 0.95 limit on a 95/95 statistical basis.

7. REFERENCES

1. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG/CR-0200, Rev. 6, 1998

2. Oak ridge National Laboratory, M.B. Emett and W.C. Jordan, A Guide to Verification and Validation of the SCALE-4 Criticality Safety Software, NUREG/CR-6483 (ORNL/TM-12834), November, 1996

3. Altran Corporation Verification Report, VR AV-26, Rev. 0, date 3/3/2000

4. Pacific Gas and Electric Corporation file, "ISFSI Cask RFP - Appendix B, Spent Fuel Parameters, Humboldt Bay Power Plant", 1/13/2000

5. Pacific Gas and Electric Corporation HBPP Drawing #420612, Sheet 1/2, Change 3, "Modification to Spent Fuel Storage Facility, Unit No. 3, Humboldt Bay Power Plant"

6. R. M. Johnson, Pacific Gas and Electric Corporation Calculational File Number 850723-0, HBPP#3 SFP Poison Tube analysis, July 23, 1985

7. U.S. Nuclear Regulatory Commission Internal Memorandum, J. Wermiel to J. Hannon, "Draft Criticality Assessment of Criticality Events in Decommissioned Spent Fuel Pools", 12/14/99

8. American Society of Mechanical Engineers, "Steam Tables" 11th Edition, ASME, 1984

9. U.S. Nuclear Regulatory Commission, HBPP Unit No. 3 Decommissioning, Safety Evaluation Report, April 29, 1987

10. Pacific Gas and Electric Corporation HBPP Drawing #643839, Change 1, "Fuel Storage Rack"

11. "Boral, the Neutron Absorber" Technical Report. Brooks and Perkins Advanced Structures, Product Performance Report 624

12. R. M. Johnson, Pacific Gas and Electric Corporation Calculational File Number 850613-0, HBPP#3 SFP Poison loading, June 25, 1985

13. U.S. Nuclear Regulatory Commission, Generic Letter of April 14, 1978, "NRC Guidance Regarding Review and Acceptance of Spent Fuel Storage..."

14. Pacific Gas and Electric Company, Humboldt Bay Power Plant Unit #3 DSAR, August 1998, Revision 2.

15. ANSI/ANS Standard 8.17.1984, "Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors" American Nuclear Society", 1984

16. Pacific Gas and Electric Corporation, HBPP Drawing #6019924-17, "Poison Can"

17. U.S. Nuclear Regulatory Commission Internal Memorandum, L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", 8/19/98

18. Calculation N265 (Altran report No. 00138-TR-001) Dated 5/31/200

Attachment A. Benchmark Critical Experiment Validation Study on CSAS25/KENO Va with Statistical Analysis of Bias and Uncertainties Using the USLSTATS Code

A.1 General

Criticality Codes are of two general types: deterministic codes, which solve the Boltzmann Transport equation for a system with a multiplicative medium, and probabilistic (a.k.a. Monte-Carlo), which attempt to model the histories of neutrons from birth until absorption within the system or escape from it. The latter type use random numbers to determine where the neutron will be born, the direction and velocity (energy) it will have, and the distance it will travel before it collides with a nucleus. Once a collision occurs, a random number is used to determine whether (based on the cross sections involved) the type of nucleus involved and whether the collision will involve a scattering or absorption, and what type.

For a multiplying medium, at least some of the absorptions result in additional neutrons. After a group, or "generation" of neutrons of sufficient size for statistical analysis (500-1500 neutrons) have had their life histories tracked, a new generation of the same size is started in the locations where they were created. The ratio of neutrons created by multiplication in the medium to the number in the generation that precedes and creates them is the generation k_{eff}. This is essentially the average ratio over a large number of generations (usually 300-500), generally skipping the first 5-25 generations (during which the eigenvalue—the generation k_{eff}—is fluctuating by relatively large amounts).

To assure that results are accurate, a simulation of a critical benchmark experiment, in which fissile material is brought together into a critical configuration with k_{eff}=1.00 is performed. If the computer code calculates a k_{eff} above or below this, this is called a bias (positive or negative, respectively). When the results of simulations of many different critical experiments are compared, then it can be determined if the biases are always the same, or if there is random statistical variation in the results (the variance of the bias). Finally, calculated k_{eff} may depend on parameters of the critical system (such as fuel enrichment, rod pitch, or moderator-to-fuel volume ratio). A spreadsheet or an auxiliary code can do the analysis of these comparisons to the benchmark critical experiments.

A.2 Summary of Benchmark Cases

Reference 1A & 6A documents the physical description of 180 LWR benchmark critical experiments performed by several NSSS vendors and NRC contractors, primarily by Babcock and Wilcox (B&W), Westinghouse, and Pacific Northwest Laboratory (PNL). Of this set, 125 were chosen to model for the purpose of benchmarking Keno Va for a typical LWR spent fuel pool. These were chosen because parameters such as ^{235}U enrichment, fuel rod pitch, assembly separation distance, and moderator to fuel ratio bracketed what would typically be seen in a LWR spent fuel pool. Those not picked were primarily because 1) they used absorbers other than b_4c (i.e. containing cadmium, gadolinium), 2) had plutonium oxide or mixed-oxide (MOX) fuel, or 3) because they had non-square lattice geometries. Results of each benchmark case are presented in the attached table.

A.3 Statistical Analysis of Results

Reference (5A) recommends that calculational methods used to determine criticality safety limits for

applications outside reactors are validated by comparison with appropriate critical experiments. The goal is to identify any biases in the calculational method along with the uncertainties associated with those biases. Once these are known then a maximum allowable value for the calculated k_{eff} can be established. This "Upper Safety Limit" (USL) includes the calculational bias, bias uncertainty and any desired additional safety margin to ensure sub-criticality. This method includes all of the uncertainties related to the methodology, but not the uncertainties that come from a particular computer run or the mechanical tolerances (rod pitch, assembly spacing, pellet enrichment, etc.).

USLSTATS is an auxiliary code that determines the calculational bias (β), and the uncertainty in the bias ($\Delta\beta$). To adequately cover characteristics of the fuel and its environment the following five statistical studies were conducted using the output from the benchmark models:

1. K_{eff} versus Average Energy Group (AEG)
2. K_{eff} versus ^{235}U Enrichment
3. K_{eff} versus Moderator/Fuel Volume Ratio
4. K_{eff} versus Fuel Rod Pitch
5. K_{eff} versus Fuel Assembly Separation

Each KENO computer run calculates the average energy group causing fission. The other four parameters of interest are physical characteristics of the lattice, fuel and moderator. The first statistical analysis examines the correlation of the given parameter to k_{eff}. Correlation results are presented at the bottom of the attached table. AEG has a negative correlation with k_{eff}, which implies that, the k_{eff} vs. AEG points are randomly scattered. The other parameters have low to moderate correlation with k_{eff}, and none had a high correlation (i.e. >0.67).

USLSTATS should then be run for every parameter that shows at least moderate correlation with k_{eff}. It is conservative to run USLSTATS for every parameter of interest although this may result in an unnecessarily restrictive USL.

The USL is calculated as a linear equation of k_{eff} vs. X (where X is the parameter value) over the range of X, and alternately, put in table form. There are two ways to use this information: a) most precisely, for every KENO Va run, consult the five tables and pick the lowest USL for each of the five tables based on the five parameter values for a particular run:

$$USL = \min [USL(X1_1), USL(X2_1), USL(X3_1), USL(X4_1), USL(X5_1)]$$

Where $X\#_1$ is the specific parameter value for a particular KENO run. More generically, the minimum value of USL can be picked for each table for the range of parameters given, that is:

$$USL = \min [\min(USL(X1)), \min(USL(X2)), \min(USL(X3)), \min(USL(X4)), \min(USL(X5))]$$

Where X1 through X5, the parameters of interest, are allowed to run over all possible values in their ranges.

This is conservative but not necessarily restrictive because in practice the slope of the regression line k_{eff} vs. X is so small that the penalty for talking a generic minimum as above is typically a few tenth of 1-% Δk. The output of USLSTATS is discussed in greater detail in Reference 6A.

The most conservative bias was determined generically by using the minimum USL based parameter. This was k_{eff} versus Fuel Rod Pitch. The calculation bias (β) was determined to be .00684 and the uncertainty in the bias ($\Delta\beta$) was determined to be .00386. The addition of the two results in an overall bias of 0.01070. This establishes a benchmark bias for the k_{eff} value calculated by CSAS25/KENO Va that will have a 95 percent probability at a 95 percent confidence level.

Table C-A1. Benchmark Critical Experiment Summary

Run ID	U Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1s
B1645SO1	2.46	1.41	1.015		32.8165	0.99673	0.00159
B1645SO2	2.46	1.41	1.015		32.7349	0.99734	0.00156
BW1231B1	4.02	1.511	1.139		31.1369	0.9979	0.00143
BW1231B2	4.02	1.511	1.139		29.8855	0.99752	0.00143
BW1273M	2.46	1.511	1.376		32.2060	0.99608	0.00124
BW1484A1	2.46	1.636	1.841	1.636	34.5468	0.99581	0.00158
BW1484A2	2.46	1.636	1.841	4.908	35.1582	0.9903	0.0016
BW1484B1	2.46	1.636	1.841		33.9534	0.99922	0.0013
BW1484B2	2.46	1.636	1.841	1.636	34.5785	0.99688	0.00137
BW1484B3	2.46	1.636	1.841	4.908	35.2463	0.9979	0.00137
BW1484C1	2.46	1.636	1.841	1.636	34.6495	0.99223	0.00171
BW1484C2	2.46	1.636	1.841	4.908	35.2374	0.99223	0.00186
BW1484S1	2.46	1.636	1.841	1.636	34.5097	0.99898	0.00156
BW1484S2	2.46	1.636	1.841	1.636	34.5493	1.00021	0.00142
BW1484SL	2.46	1.636	1.841	6.544	35.4144	0.99223	0.00158
BW1645S1	2.46	1.209	0.383	1.778	30.1098	1.00146	0.00165
BW1645S2	2.46	1.209	0.383	1.778	29.9731	1.00276	0.00172
BW1810A	2.46	1.636	1.841		33.9627	0.99911	0.00127
BW1810B	2.46	1.636	1.841		33.9677	0.99773	0.00127
BW1810C	2.46	1.636	1.841		33.1621	1.00055	0.00156
BW1810D	2.46	1.636	1.841		33.1274	0.99681	0.00132
BW1810E	2.46	1.636	1.841		33.1341	0.99924	0.00129
BW1810F	2.46	1.636	1.841		33.9669	1.00422	0.00131
BW1810G	2.46	1.636	1.841		32.9519	0.99987	0.00147
BW1810H	2.46	1.636	1.841		32.9475	0.99719	0.00178
BW1810I	2.46	1.636	1.841		33.9559	1.00539	0.00135
BW1810J	2.46	1.636	1.841		33.1249	1.00092	0.00146
DSN399-1	4.74	1.6	3.807	1.8	33.9780	1.00811	0.00178
DSN399-2	4.74	1.6	3.807	5.8	34.4310	0.99956	0.0018
DSN399-3	4.74	1.6	3.807		35.3231	1.00447	0.00194
DSN399-4	4.74	1.6	3.807		35.3800	1.00079	0.00204
EPRU65	2.35	1.562	1.196		33.9269	0.99867	0.00181
EPRU65B	2.35	1.562	1.196		33.3979	1.00256	0.00141
EPRU75	2.35	1.905	2.408		35.8569	0.99573	0.00157
EPRU75B	2.35	1.905	2.408		35.3204	1.00254	0.00149

Table C-A1. Benchmark Critical Experiment Summary, continued

Run ID	U Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
EPRU87	2.35	2.21	3.687		36.6308	0.99886	0.00143
EPRU87B	2.35	2.21	3.687		36.3458	1.00072	0.00141
NSE71SQ	4.74	1.26	1.823		33.7672	0.99728	0.00175
NSE71W1	4.74	1.26	1.823		33.9994	0.99631	0.00174
NSE71W2	4.74	1.26	1.823		34.3729	0.99786	0.00188
P2438BA	2.35	2.032	2.918	5.05	36.2319	1.0001	0.00174
P2438SLG	2.35	2.032	2.918	8.39	36.2910	0.99728	0.00162
P2438SS	2.35	2.032	2.918	6.88	36.2786	0.99823	0.00167
P2438ZR	2.35	2.032	2.918	8.79	36.2919	0.99888	0.00186
P2615BA	4.31	2.54	3.883	6.72	35.7316	0.99805	0.00157
P2615SS	4.31	2.54	3.883	8.58	35.7532	0.99928	0.00165
P2615ZR	4.31	2.54	3.883	10.92	35.7741	0.99755	0.00165
P2827L1	2.35	2.032	2.918	13.27	36.2283	1.00178	0.00161
P2827L2	2.35	2.032	2.918	11.25	36.2940	0.99849	0.00173
P2827L3	4.31	2.54	3.883	20.78	35.6779	1.01113	0.00177
P2827L4	4.31	2.54	3.883	19.04	35.7177	1.00646	0.00166
P2827SLG	2.35	2.032	2.918	8.31	36.3006	0.99695	0.0015
P3314BA	4.31	1.892	1.6	2.83	33.1745	1.00055	0.00183
P3314BC	4.31	1.892	1.6	2.83	33.2212	1.00202	0.00163
P3314BF1	4.31	1.892	1.6	2.83	33.2560	1.00526	0.00157
P3314BF2	4.31	1.892	1.6	2.83	33.2025	0.99955	0.00169
P3314BS1	2.35	1.684	1.6	3.86	34.8619	0.99518	0.00177
P3314BS2	2.35	1.684	1.6	3.46	34.8276	0.99451	0.00155
P3314BS3	4.31	1.892	1.6	7.23	33.4557	0.99634	0.00162
P3314BS4	4.31	1.892	1.6	6.63	33.3966	1.00207	0.00172
P3314SLG	4.31	1.892	1.6	2.83	34.0201	1.00002	0.00199
P3314SS1	4.31	1.892	1.6	2.83	33.9611	1.00141	0.00171
P3314SS2	4.31	1.892	1.6	2.83	33.7666	0.9996	0.00203
P3314SS3	4.31	1.892	1.6	2.83	33.8930	0.99911	0.00189
P3314SS4	4.31	1.892	1.6	2.83	33.7682	0.99809	0.00158
P3314SS5	2.35	1.684	1.6	7.8	34.9281	0.99245	0.00196
P3314SS6	4.31	1.892	1.6	10.52	33.5314	1.00265	0.00196
P3314W1	4.31	1.892	1.6		34.3935	1.00121	0.00185
P3314W2	2.35	1.684	1.6		35.2303	0.9983	0.00171
P3314ZR	4.31	1.892	1.6	2.83	33.9777	0.99731	0.0019

Table C-A1. Benchmark Critical Experiment Summary, continued

Run ID	U Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1s
P3602BB	4.31	1.892	1.6	8.3	33.3454	1.00431	0.00175
P3602BS1	2.35	1.684	1.6	4.8	34.7814	1.00095	0.00185
P3602BS2	4.31	1.892	1.6	9.83	33.3724	1.0051	0.002
P3602N11	2.35	1.684	1.6	8.98	34.7272	1.00368	0.00159
P3602N12	2.35	1.684	1.6	9.58	34.8196	1.00414	0.00166
P3602N13	2.35	1.684	1.6	9.66	34.9455	1.00383	0.00158
P3602N14	2.35	1.684	1.6	8.54	35.0345	0.99743	0.00158
P3602N21	2.35	2.032	2.918	11.2	36.2799	0.99847	0.00158
P3602N22	2.35	2.032	2.918	10.36	36.1878	1.00086	0.0013
P3602N31	4.31	1.892	1.6	14.87	33.2018	1.00859	0.00194
P3602N32	4.31	1.892	1.6	15.74	33.2950	1.00727	0.00181
P3602N33	4.31	1.892	1.6	15.87	33.3933	1.00729	0.00184
P3602N34	4.31	1.892	1.6	15.84	33.4592	1.00631	0.00199
P3602N35	4.31	1.892	1.6	15.45	33.5075	1.00186	0.00201
P3602N36	4.31	1.892	1.6	13.82	33.5903	0.99992	0.00208
P3602N41	4.31	2.54	3.883	12.89	35.5080	1.00973	0.0016
P3602N42	4.31	2.54	3.883	14.12	35.6567	1.00707	0.00223
P3602N43	4.31	2.54	3.883	12.44	35.7484	1.00402	0.00166
P3602SS1	2.35	1.684	1.6	8.28	34.8794	1.00167	0.00174
P3602SS2	4.31	1.892	1.6	13.75	33.4332	1.00291	0.00158
P3926L1	2.35	1.684	1.6	10.06	34.8423	0.9992	0.00194
P3926L2	2.35	1.684	1.6	10.11	34.9393	1.00273	0.00173
P3926L3	2.35	1.684	1.6	8.5	35.0687	0.99848	0.0016
P3926L4	4.31	1.892	1.6	17.74	33.3187	1.00749	0.00165
P3926L5	4.31	1.892	1.6	18.18	33.3915	1.00777	0.00194
P3926L6	4.31	1.892	1.6	17.43	33.5319	1.00696	0.00162
P3926SL1	2.35	1.684	1.6	6.59	35.0623	0.99296	0.00156
P3926SL2	4.31	1.892	1.6	12.79	33.5795	0.99989	0.00168
P4267B1	4.31	1.8901	1.59		31.7976	0.99949	0.00131
P4267B2	4.31	0.89	1.59		31.5472	1.00069	0.00184
P4267B3	4.31	1.715	1.09		30.9817	1.00272	0.0016
P4267B4	4.31	1.715	1.09		30.4769	0.99866	0.00161
P4267B5	4.31	1.715	1.09		30.1042	0.99865	0.00127
P4267SL1	4.31	1.89	1.59		33.4641	1.00236	0.00165
P4267SL2	4.31	1.715	1.09		31.9602	1.00074	0.00203

Table C-A1. Benchmark Critical Experiment Summary, continued

Run ID	U Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1s
P62FT231	4.31	1.891	1.6	5.19	32.9151	1.00133	0.002
P71F14F3	4.31	1.891	1.6	5.19	32.8273	1.00227	0.00183
P71F14V3	4.31	1.891	1.6	5.19	32.8705	0.99865	0.00198
P71F14V5	4.31	1.891	1.6	5.19	32.8738	1.00112	0.00163
P71F214R	4.31	1.891	1.6	5.19	32.8698	0.99592	0.00206
PAT80L1	4.74	1.6	3.807	4.9	35.0222	0.99973	0.00183
PAT80L2	4.74	1.6	3.807	4.9	35.1008	0.99916	0.00199
PAT80SS1	4.74	1.6	3.807	4.9	35.0197	1.00184	0.0016
PAT80SS2	4.74	1.6	3.807	4.9	35.0824	0.99441	0.00234
W3269A	5.7	1.422	1.93		33.1435	0.9990	0.00189
W3269B1	3.7	1.105	1.432		32.3396	0.99551	0.0017
W3269B2	3.7	1.105	1.432		32.3742	0.99703	0.00163
W3269B3	3.7	1.105	1.432		32.2665	0.99737	0.00167
W3269C	2.72	1.524	1.494		33.7632	0.99876	0.00153
W3269SL1	2.72	1.524	1.494		33.3813	0.99739	0.00176
W3269SL2	5.7	1.422	1.93		33.1039	1.00198	0.00178
W3269W1	2.72	1.524	1.494		33.4883	0.99493	0.0017
W3269W25.7		1.422	1.93		33.1448	1.00044	0.00197
W3385SL1	5.74	1.422	1.932		33.2171	0.99774	0.00201
W3385SL2	5.74	2.012	5.067		35.9044	1.00393	0.00169
Correlation of K_{eff} with parameter	**0.340**	**0.357**	**0.180**	**0.578**	**-0.045**		

A.4 REFERENCES FOR ATTACHMENT A

1A. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG/CR-0200, Rev. 6, 1998

2A. Oak ridge National Laboratory, M.B. Emett and W.C. Jordan, A Guide to Verification and Validation of the SCALE-4 Criticality Safety Software, NUREG/CR-6483 (ORNL/TM-12834), November, 1996

3A. Altran Corporation Verification Report, VR AV-26, Rev. 0, date 3/3/2000

4A. U.S. Nuclear Regulatory Commission, Generic Letter of April 14, 1978, "NRC Guidance Regarding Review and Acceptance of Spent Fuel Storage..."

5A. ANSI/ANS Standard 8.17.1984, "Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors" American Nuclear Society", 1984

6A. Oak Ridge National Laboratory, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages", NUREG/CR-6361, March 1997.

APPENDIX D

Humboldt Bay Power Plant Unit 3

SAFSTOR

Baseline Radiation Study

SAFSTOR BASELINE RADIATION STUDY

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SAFSTOR BASELINE RADIATION STUDY

1.0 SCOPE

This document describes the radiological status of Humboldt Bay Power Plant (HBPP) Unit 3 and its environs at the beginning of SAFSTOR.

2.0 DISCUSSION

Section 5.0 of the SAFSTOR Decommissioning Plan for the HBPP, Unit No. 3, dated July 1984, describes a radiation survey to establish activity levels and nuclide concentrations in the plant and its environs at the beginning of SAFSTOR.

This SAFSTOR Baseline Radiation Study is a review of surveys of plant areas and system components, to provide a baseline for comparison with surveys during the SAFSTOR period. The review of each area summarizes radiation and radioactivity measurements. The usual summary items are removable (beta-gamma) contamination, gamma radiation levels, beta radiation levels (if found), date(s) of survey, a general description of the potential for personnel contamination, and any additional remarks.

The study includes a section that discusses the inventory of on-site radioactive waste, and a section, which describes previous measurements of environmental radioactivity.

3.0 SUMMARY

The radiological status of various plant areas ranges from those areas which require no significant control to those where personnel internal/external exposure might be significant.

Areas (that can be entered without tools or mobile lifting equipment) considered to have a high potential for personnel contamination are:

+ Access Shaft -66' elevation, REDT Area
+ Access Shaft -14' elevation, Shutdown Heat Exchanger Room
+ Access Shaft -2' elevation, Cleanup Heat Exchanger Room
+ Access Shaft -2' elevation, Cleanup Demineralizer Cell
+ Valve Gallery -14' elevation
+ Valve Gallery - 8' elevation
+ Air-Ejector Room
+ Condensate Demineralizer Regeneration Room
+ Specific Portions of the Refueling Building +12' elevation
+ New Fuel Storage Vault + Turbine Building Drain Tank Vault
+ Radwaste Treatment Building - Waste Concentrator Area
+ Radwaste Treatment Building - Sump Area
+ Radwaste Treatment Building - Filter Area (during filter changes)
+ Radwaste Treatment Building - Concentrated Waste Tank Room
+ Radwaste Treatment Building - Resin Disposal Tank Room
+ Offgas Filter/Holdup Pipe Tunnel

Areas (that can be entered without tools or mobile lifting equipment) considered to have a potential for significant personnel exposures are:

+ Access Shaft -66' elevation, REDT Area
+ Access Shaft -14' elevation, Shutdown Heat Exchanger Room

+ Access Shaft -2' elevation, Cleanup Heat Exchanger Room
+ Access Shaft -2' elevation, Cleanup Demineralizer Cell
+ Condensate Demineralizer Cubicle
+ Condensate Demineralizer Regeneration Room
+ Radwaste Treatment Building - Waste Concentrator Area
+ Radwaste Treatment Building - Sump Area
+ Radwaste Treatment Building - Resin Disposal Tank Room
+ Offgas Filter/Holdup Pipe Tunnel
+ Radwaste Treatment Building - Resin Disposal Tank Room

4.0 STATUS OF SPECIFIC AREAS

4.1 Access Shaft -66' elevation, REDT area

4.1.1 Most surfaces in this area show levels of removable contamination that range from 2,000 to 20,000 dpm/dm^2. Higher activity levels are located around the REDT pumps (70,000 dpm/dm^2) and the drainage trench to the sump (60,000 dpm/dm^2). The sump interior should be considered highly contaminated, although numerical activity levels are unknown.

4.1.2 Gamma radiation levels are generally 20 mR/hr to about 40 mR/hr, increasing toward the piping on the SW wall (about 120 mR/hr at 18"). There are numerous hot spots on the piping, with highest contact dose rates ranging from 250 to 600 mR/hr.

4.1.3 Beta radiation fields have been found in this area, associated with the drainage trench and the REDT (80 mRad/hr).

4.1.4 This summary is based on surveys taken during the period of 11/18/86 through 6/16/88.

4.1.5 This area has a high potential for personnel contamination, as well as the potential for undesirable radiation exposures.

4.1.6 Note that some of the particulate radioactive contamination found in the sump came directly from the reactor (during control rod drive changes and incore 'flushing'). Also, the loop seal in the vent line from the REDT may show increasing radiation levels if there is any vertical migration of material from higher elevations.

4.2 Access Shaft -66' elevation (exclusive of REDT area)

4.2.1 Most of this area is free of removable contamination (less than 500 dpm/dm^2). Areas of contamination have been found on the walls and on the overhead surfaces. The highest accessible removable contamination (up to 16,000 dpm/dm^2) was found in the drainage area around No. 2 Core Spray Pump.

4.2.2 Gamma radiation levels in the generally accessible area range from about 5 mR/hr near the manlift to about 15 mR/hr under the Drywell. There are numerous hotspots on the piping, with higher contact dose rates ranging from 70 to 300 mR/hr. The contact dose rate on the control rod drive lifting machine cable spool is 60 mR/hr.

4.2.3 Beta radiation fields have been found around the Core Spray pump drainage area (up to 20 mRad/hr). Beta radiation fields have been found on the control rod drive lifting machine (160 mRad/hr on the cable spool).

4.2.4 This summary is based on surveys taken during the period of 11/18/86 through 6/16/88.

4.2.5 This area has moderate potential for personnel contamination from the sources discussed above, as well as from the contamination in the REDT area.

4.2.6 Note that in addition to the sources discussed above, the painted surfaces contain fixed contamination. During almost every refueling outage, this area was contaminated when control rod drives were exchanged. When decontamination was not successful, the floor was repainted.

4.3 Access Shaft -54' elevation

4.3.1 Floor surfaces in this area show low levels of removable contamination (at or below 500 dpm/dm^2 as of 6/7/88). Surfaces and piping around the control rod drive hydraulic accumulators show levels from 2,000 to 6,000 dpm/dm^2 with occasional spots to 16,000 dpm/dm^2.

4.3.2 Gamma radiation levels currently range from about 6 mR/hr in the center of the area to about 10 mR/hr in front of the accumulators on the east and west walls. Except for a hot spot on an overhead valve west of the manlift (350 mR/hr contact in 1986), the hot spots on the accessible piping are less than 20 mR/hr. The accumulators have contact dose rates of 250 to 400 mR/hr (1986).

4.3.3 No beta radiation fields were reported in this area.

4.3.4 This summary is based on surveys taken during the period of 1/16/86 through 6/7/88. Over this time period, dose rates have appeared to decay by approximately 25 percent.

4.3.5 This area has moderate potential for personnel contamination from the various piping surfaces. The potential for personnel radiation exposure is low for routine inspection, but moderate for any work in contact with the accumulators/piping.

4.3.6 Note that the surfaces below the grating on the west side of the area were not surveyed. Because this area collected contamination whenever the accumulators were exchanged, there may be sources of removable contamination there.

4.4 Access Shaft -44' elevation

4.4.1 Floor surfaces in this area generally do not have detectable levels of removable contamination (less than 500 dpm/dm^2 as of 6/7/88). Typical surface contamination on piping and equipment is about 2,000 dpm/dm^2, with occasional hot spots to 12,000 dpm/dm^2.

4.4.2 Gamma radiation levels currently range from about 3 mR/hr in the center of the area to about 5 mR/hr (waist high) at the east and west walls. Higher levels of 10 mR/hr can be found at floor level above the accumulators, and contact dose rates up to 15 mR/hr were reported on piping on the east side of the area.

4.4.3 No beta radiation fields were reported in this area.

4.4.4 This summary is based on surveys taken during the period of 1/16/86 through 6/7/88. Over this time period, dose rates appeared to decay by approximately 25 percent.

4.4.5 This area has moderate potential for personnel contamination from the various piping surfaces.

4.5 Access Shaft -34' elevation

4.5.1 Floor surfaces in this area generally do not have detectable levels of removable contamination (less than 500 dpm/dm^2). Most other surfaces have low contamination levels (at or below 1,000 dpm/dm^2), except for a few spots of contamination (e.g. 8,000 dpm/dm^2 on the edge of the ladder hatch way).

4.5.2 Gamma radiation levels are currently about 1 mR/hr in the east side of the area, decreasing to 0.2 mR/hr in the west side of the area. Highest contact dose rates on piping in this area are 5 mR/hr (SE corner) and 3 mR/hr (NE wall).

4.5.3 No beta radiation fields were reported in this area.

4.5.4 This summary is based on surveys taken during the period of 1/6/86 through 6/3/88.

4.5.5 This area has low to moderate potential for personnel contamination.

4.6 Access Shaft -24' elevation

4.6.1 Surfaces in this area generally do not have detectable levels of removable contamination (less than 500 dpm/dm^2), and except for a few measurements (2,000 to 3,000 dpm/dm^2), contamination levels are at or below 1,000 dpm/dm^2.

4.6.2 Gamma radiation levels are currently about 0.6 mR/hr in the east side of the area, decreasing to 0.2 mR/hr in the west side of the area. Highest contact dose rates on piping in this area are 5 mR/hr (SE corner, east side) and 3 mR/hr (SE wall, west side).

4.6.3 No beta radiation fields were reported in this area.

4.6.4 This summary is based on surveys taken during the period of 1/16/86 through 6/3/88.

4.6.5 This area has low to moderate potential for personnel contamination.

4.7 Access Shaft -14' elevation (exclusive of Shutdown Heat Exchanger Room)

4.7.1 Surfaces in this area are either "clean" or have barely detectable levels of removable contamination. Except for a few measurements of approximately 2,000 dpm/dm^2, contamination levels are at or below 1,000 dpm/dm^2.

4.7.2 Gamma radiation levels in the main room vary from a low of 0.2 mR/hr (near stair) to 1 mR/hr at the east and west ends. There is a hot spot of 10 mR/hr (contact with overhead pipe) at the east end of the area. The highest dose rate at contact with the Low Pressure Core Flooding line (shielded elbow on south wall) is 20 mR/hr. The general area dose rate in the access room to the piping chase is about 3 mR/hr, with hot spots on the piping (20 mR/hr in pipe chase, and 90 mR/hr in SE corner).

4.7.3 No beta radiation fields were reported in this area.

4.7.4 This summary is based on surveys taken during the period of 1/16/86 through 5/20/88.

4.7.5 This area has moderate potential for personnel contamination.

4.8 Access Shaft -14' elevation, Shutdown Heat Exchanger Room

4.8.1 Based on a limited number of samples, the removable contamination level of the floor in this room is approximately 3,000 to 6,000 dpm/dm^2.

4.8.2 Gamma radiation levels in this area vary considerably due to the complexity of the piping. The general area is expected to be about 30 to 50 mR/hr, with hot spots on piping of a few R/hr.

4.8.3 Beta radiation has been reported for this area, but except for open systems (e.g. drains) it does not appear to be significant as compared to the gamma fields.

4.8.4 This summary is from data collected 9/26/88, as well as from data collected May, 1984.

4.8.5 Because of the scarcity of information, this area is considered to have a moderate to high potential for personnel contamination.

4.9 Access Shaft -2' elevation (exclusive of Cleanup Heat Exchanger Room)

4.9.1 Surfaces in this area are either "clean" or have barely detectable levels of removable contamination. Except for one measurement of approximately 2,000 dpm/dm^2, contamination levels are at or below 1,000 dpm/dm^2.

4.9.2 Gamma radiation levels in the main room vary from a low of 0.3 mR/hr (foot of stair) to about 2 mR/hr elsewhere. Dose rates of up to 10 mR/hr can be found on a few pipes. There is some "shine" (5 mR/hr) at the door to the Cleanup Heat Exchanger Room.

4.9.3 No beta was observed in this area, except for a floor drain toward the East end of the room (8 mRad/hr beta).

4.9.4 This summary is based on surveys taken during the period of 1/16/86 through 5/20/88.

4.9.5 This area has low to moderate potential for personnel contamination.

4.10 Access Shaft -2' elevation, Cleanup Heat Exchanger Room

4.10.1 Based on a limited number of samples, the removable contamination level of surfaces in this room may be as high as 20,000 dpm/dm^2 (as of May 1984), although subsequent data suggests a decreasing trend to about 3,000 dpm/dm^2 (September 1988).

4.10.2 Gamma radiation levels in this area vary considerably due to the complexity of piping. The general area is expected to be about 40 to 60 mR/hr, with hot spots on piping of a few R/hr.

4.10.3 Beta radiation was not reported for this area, but might be expected at any open systems (e.g. drains).

4.10.4 This summary is from data collected 9/26/88, as well as from data collected May, 1984.

4.10.5 Because of the scarcity of information, this area is considered to have a high potential for personnel contamination.

4.10.6 To aid interpretation of surveys, it should be noted that the new (PG&E West) heat exchangers are relatively free of contamination and internal radioactivity, as compared to the old ones that were abandoned in place (about 1976).

4.11 Access Shaft -2' elevation, Cleanup Demineralizer Cell

4.11.1 The removable contamination level of surfaces in this room is highly variable, ranging from 1,000 to 100,000 dpm/dm^2 (beta-gamma). Removable alpha contamination was also reported at about 200 dpm/dm^2.

4.11.2 Gamma radiation levels in the cell are generally highest at floor level (due to radiation from components placed here for SAFSTOR). Highest contact dose rates are the Demineralizer Inlet piping (250 mR/hr) and the drum containing Scram Dump Tank Level instrumentation (420 mR/hr at floor level).

4.11.3 Beta radiation has been reported for this area, but except for open systems (e.g. drains) it does not appear to be significant as compared to the gamma fields.

4.11.4 This summary is based on a survey taken 12/30/86, with other data from May 1984.

4.11.5 This area is considered to have high potential for personnel contamination and undesirable personnel radiation exposure.

4.11.6 Note that there are 6 drums stored here, containing contaminated components (such as the hydraulic filter housings, the cleanup pump, and scram dump tank level instrumentation).

4.12 Escape Hatch -66' to +12' elevation

4.12.1 The removable contamination levels range from less than 500 to 1,000 dpm/dm^2 from the +12' to the -14' elevations. Contamination levels range from 1,000 to 3,500 dpm/dm^2 from the -34' to -66' elevations.

4.12.2 No gamma or beta readings are available, but because there are no radioactive systems in this area, no significant radiation is expected.

4.12.3 This summary is based on a survey taken on 9/9/86.

4.12.4 This area has low to moderate potential for personnel contamination.

4.13 Refueling Building +12' elevation

4.13.1 This area has varying levels of removable contamination. Contamination levels are generally less than 500 dpm/dm^2, with spots of removable contamination at levels of 1,000 to 3,000 dpm/dm^2. Higher levels are found at the following locations:

a) Removable contamination at levels of 25,000 to 60,000 dpm/dm^2 was found on the crane main hook cable spool, traveling block and main hook. Note that the more highly contaminated portions are enclosed in a vinyl bag.

b) The surfaces of the Spent Fuel Pool 'overflow scuppers' (in the Southwest and Southeast corners) had 200,000 to 400,000 dpm/dm^2 of removable contamination. Note that these areas are now below the sheet metal flashing around the pool.

c) The Spent Fuel Pool Recirc. Pumps have indicated varying smear results, ranging from a typical level of 10,000 to 20,000 dpm/dm^2 to a high of 260,000 dpm/dm^2.

d) Some of the piping below the West end of the Emergency Condenser has been found to have removable contamination levels in the range of 15,000 to 25,000 dpm/dm^2.

e) Smears of the two hydraulic pumps (prior to decontamination) indicated removable contamination levels of 10,000 to 30,000 dpm/dm^2. Although more recent smears show reduced levels (e.g. 1,000 dpm/dm^2), it is not clear that all of the contamination has been removed.

f) The 'Washdown Area' has removable contamination in the range of 2,000 to 20,000 dpm/dm^2 (including a smear of the top of the concrete block wall, that indicated 15,000 dpm/dm^2) .

4.13.2 Gamma radiation levels are generally less than 1 mR/hr, increasing to about 3 to 5 mR/hr around the pool, the recirculation pumps, and the hydraulic system pumps. Various hotspots are located on piping, particularly piping near the hydraulic pumps (about 50 mR/hr), on piping near the Recirc. Pumps (20 to 50 mR/hr) and on the recirc. pump discharge line, about 8' above floor level (100 mR/hr).

4.13.3 Beta radiation fields have been measured at the floor drains (12 to 32 mRad/hr) and at the SFP Recirc. Pumps (survey data illegible).

4.13.4 This summary is based primarily on surveys performed from 8/18/86 through 6/17/88.

4.13.5 Most of this area has a moderate potential for personnel contamination, but specific locations have high potential for personnel contamination.

4.13.6 It should be noted that this area was contaminated throughout the life of the plant, and that the floor was routinely repainted to fix any contamination remaining after decontamination. A similar treatment was applied to the exterior of the Fuel Transfer Cask. There is also a variety of previously contaminated components stored in this area, including the (interior of) the Fuel Transfer Cask, the fuel handling tools (in the washdown area), the Extension Tank (inside the sheet metal can on top of the reactor shield plug), and some of the slings (Northeast corner of area). For a final note, the 4" steel floor plate (between the pool and the reactor) was installed under contaminated conditions, so that the floor underneath it should be assumed contaminated.

4.14 New Fuel Storage Vault

4.14.1 Floor surfaces have typical removable contamination levels of from 1,000 to 10,000 dpm/dm^2. Higher values have been observed on a valve on the Spent Fuel Pool Cooler (18,000 dpm/dm^2).

4.14.2 General area gamma dose rates are 10 to 25 mR/hr. Contact dose rates on the 'cans' containing the spare control rod drives show hotspots (up to 160 mR/hr).

4.14.3 A beta radiation field of 20 mRad/hr was found at the sample scupper in the Northwest corner of the area.

4.14.4 This summary is based on surveys performed from 4/10/86 through 1/22/88.

4.14.5 This area has a high potential for personnel contamination.

4.14.6 Note that this area was used to store contaminated (but not irradiated) fuel bundles. The spare control rod drives are stored (in 'cans') in this area.

4.15 Turbine Building Drain Tank Vault

4.15.1 Floor surfaces have removable contamination levels ranging from 3,000 to 9,000 dpm/dm^2. Walls have removable contamination levels ranging from 10,000 to 170,000 dpm/dm^2. The West wall has the highest readings (170,000 dpm/dm^2 beta-gamma, 40 cpm alpha) The West end of the tank has the highest readings (90,000 dpm/dm^2, 20 cpm alpha. The pump located in the SE corner has removable contamination levels of 100,000 dpm/dm^2.

4.15.2 General area gamma dose rates are 10 to 25 mR/hr. Contact dose rates on the bottom of the tank range from 30 to 150 mR/hr. There is a 60 mR/hr hot spot on a pipe in the overhead (in the middle of the South wall).

4.15.3 No beta radiation fields were found with the exception of the floor drain (80 mRad/hr).

4.15.4 This summary is based on surveys performed from 4/10/86 through 1/22/88.

4.15.5 This area has a high potential for personnel contamination.

4.16 Valve Gallery -14' elevation

4.16.1 The removable contamination levels range from less than 500 to 2,500,000 dpm/dm^2 with a general average of about 2,000 to 10,000 dpm/dm^2. The highest contamination was found at a valve (assumed to be the 8" Shutdown System Return to Reactor Motor Operated Valve). At this location, the smear indicated 0.5 mR/hr gamma, 60 mRad/hr beta, and 1,200 dpm alpha.

4.16.2 Gamma radiation levels in this area are generally 10 to 20 mR/hr. The major hotspot is at the valve (and line) indicated above, with a contact reading of 40 mR/hr.

4.16.3 Beta radiation was primarily found at the hotspot at the valve described above (200 mRad/hr). In addition, the two floor drains were observed to be beta sources (12 to 64 mRad/hr).

4.16.4 This summary is primarily based on a survey taken on 5/22/86 and 5/23/86.

4.16.5 This area has moderate to high potential for personnel contamination.

4.17 Valve Gallery -8' elevation

4.17.1 The removable contamination levels range from less than 500 to 30,000 dpm/dm^2 with a general average of about 5,000 dpm/dm^2. One smear was found to have alpha with 200 dpm/dm^2.

4.17.2 Gamma dose rates in this area are generally 10 to 30 mR/hr. A variety of hotspots were observed on piping and valves, up to 220 mR/hr.

4.17.3 Beta radiation was primarily found at a hotspot at a valve near the center of the area (possibly the feedwater isolation valve) with a reading of 320 mRad/hr.

4.17.4 This summary is primarily based on a survey taken on 5/21/86 and 5/23/86.

4.17.5 This area has moderate to high potential for personnel contamination.

4.18 Valve Gallery -2' elevation

4.18.1 The removable contamination levels range from less than 500 to 18,000 dpm/dm^2 with a general average of about 2,000 dgm/dm^2. No removable alpha activity was observed.

4.18.2 Gamma dose rates in this area are generally 10 to 30 mR/hr, except around the Emergency Condenser Condensate Return Valve (East of Reactor). A variety of hotspots were observed on piping and valves, up to 280 mR/hr.

4.18.3 No beta radiation was observed in this area.

4.18.4 This summary is primarily based on a survey taken on 5/21/86 and 5/23/86.

4.18.5 This area has moderate potential for personnel contamination.

4.19 Pipe Tunnel North +6' elevation

4.19.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2).

4.19.2 Gamma radiation dose rates are approximately 1 mR/hr. Higher dose rates (2 to 6 mR/hr) are found near piping.

4.19.3 No beta radiation fields have been detected.

4.19.4 This summary is based on surveys performed from 1/14/86 through 6/2/88.

4.19.5 This area is considered to have a low potential for personnel contaminations.

4.19.6 Note that although most of the area is clean, there is contamination in the drainage trench which underlies the 'seismic patch plate' in the middle of the room.

4.20 Pipe Tunnel South +6' elevation

4.20.1 This area has low to moderate levels of removable contamination (generally at or below 1,000 dpm/dm^2, with a few locations up to 4,000 dpm/dm^2).

4.20.2 General gamma radiation dose rates in the area under the turbine steam exhaust trunk range from 5 mR/hr up toward 35 mR/hr closer to the condenser. Dose rates near the North end of the feedwater heaters and associated piping are approximately 25 mR/hr (at 18") with hotspots up to 150 mR/hr.

4.20.3 No beta radiation was noted in this area on the most recent survey, but some beta was observed during earlier surveys (at a floor drain and at the packing of valves located along the West wall).

4.20.4 This summary is based on surveys performed from 7/4/86 through 7/14/86.

4.20.5 This area has low to moderate potential for personnel contamination.

4.21 Pipe Tunnel Condenser Area +6' elevation

4.21.1 The surfaces in this general area have levels of contamination of about 1,000 to 4,000 dpm/dm^2. The contamination levels under the condenser have not been well defined, but are expected to be in the range of 1,000 to 10,000 dpm/dm^2.

4.21.2 Away from the condenser, the area dose rates vary from about 10 to 50 mR/hr, depending on the radiation levels of the nearby piping. The bottom of the condenser has contact readings varying from 100 to 250 mR/hr, typically dropping to less than 100 mR/hr at 18". The space under the condenser probably has dose rates in the range of 100 to 200 mR/hr.

4.21.3 A reading of 15 mRad/hr for Beta radiation was observed at the floor drain near the center of the East wall.

4.21.4 This summary is primarily based on surveys performed from 7/16/86 through 7/17/86.

4.21.5 This area is considered to have a moderate potential for personnel contamination.

4.22 Pipe Tunnel Condenser Area +12' elevation

4.22.1 This area has generally low levels of removable contamination (less than 500 dpm/dm^2), except that 1,000 dpm/dm^2 of removable beta contamination was observed on the North condenser waterbox doors (associated with 2 dpm/dm^2 of removable alpha activity).

4.22.2 Gamma radiation dose rates range from 2 to 15 mR/hr, with the higher dose rates directly in front of the access door on the East side of the area.

4.22.3 No beta radiation fields were detected.

4.22.4 This information is primarily based on surveys performed on 4/7/86 through 7/16/86, and on 5/25/88.

4.22.5 This area is considered to have a low potential for personnel contamination.

4.23 Air Ejector Room

4.23.1 This area is considered to be a contaminated area. Removable contamination levels on most surfaces are generally 1,000 to 10,000 dpm/dm^2. The highest

removable contamination levels are associated with the turbine gland seal exhaust fans and condenser. One smear of the equipment pedestal read 24 mRad beta. A different survey at the same point indicated 165,000 dpm/dm^2 beta-gamma and 150 dpm/dm^2 alpha.

4.23.2 Gamma radiation levels in the general area are about 1 to 5 mR/hr, with equipment/piping contact readings up to 30 mR/hr. There is one spot on (in) the concrete floor under the offgas flow meter loop seal (North side of the air ejector) that produced a contact reading of 100 mR/hr.

4.23.3 Beta radiation has been observed on the equipment pedestal noted above (72 mRad/hr), on a floor drain (24 mRad/hr), and at the hot spot on the floor (1200 mRad/hr).

4.23.4 This information is based on surveys performed between 5/15/86 and 4/25/87.

4.23.5 This area has a high potential for personnel contamination.

4.24 Condensate Pump Room

4.24.1 This area is generally considered to be free of removable contamination (less than 500 dpm/dm^2), except for some portions of the electric drive vacuum pump (6,000 to 10,000 dpm/dm^2) and the pits for the two condensate pumps (30,000 dpm/dm^2).

4.24.2 Gamma radiation fields are in the range of 1 to 5 mR/hr. Hot spots on piping are typically 20 mR/hr, with the exception of one hot spot on piping of 60 mR/hr.

4.24.3 Recent survey information on beta fields is not available. However, a survey performed on 5/20/86 showed beta fields of 20 mRad/hr in one of the pits for the condensate pumps.

4.24.4 This information is based on surveys performed from 1/17/86 through 3/4/88.

4.24.5 This area is considered to have a low potential for personnel contamination.

4.25 Instrument Vault in Demineralizer Pipe Gallery

4.25.1 This area is considered to be essentially free of removable contamination. Contamination levels are at or below 1,000 dpm/dm^2.

4.25.2 Gamma radiation readings in this area are less than 0.5 mR/hr.

4.25.3 No beta radiation was found in this area.

4.25.4 This summary is based on surveys performed 12/29/86 and 6/13/88.

4.25.5 This area has a low potential for personnel contamination.

4.26 Condensate Demineralizer Pipe Gallery

4.26.1 This area is not considered to have significant removable contamination (less than 500 dpm/dm^2).

4.26.2 Gamma radiation levels about 1 to 2 mR/h in the Southern side of the area, increasing toward the door to the Regeneration Room, and toward the Condensate piping. There are hot spots on the bottom of the condensate demineralizer strainers (of about 15 to 30 mR/hr).

4.26.3 No beta radiation fields exist with the exception of the West floor drain which measures about 1 mRad/hr.

4.26.4 This summary is based on surveys performed from 1/17/86 through 8/4/88.

4.26.5 This area has a low potential for personnel contamination.

4.27 Condensate Demineralizer Cubicle

4.27.1 Removable contamination levels in this area range from less than 500 to 2,000 dpm/dm^2.

4.27.2 Gamma radiation levels in the general area are 20 to 50 mR/hr when the resin in #1 Demineralizer is not contaminated (see below). Otherwise, the general levels can be about 200 to 300 mR/hr. Typical levels for the contact dose rate profile of the demineralized tank are approximately 40 mR/hr at the top and bottom and 2 to 3 R/hr at the middle.

4.27.3 No beta radiation fields were reported in this area.

4.27.4 This information is based on surveys performed from 1/9/87 through 8/4/88.

4.27.5 This area has a low to moderate potential for personnel contamination, and has the potential for undesirable radiation exposures.

4.27.6 Note that the #1 (nearest to the entrance) demineralizer tank was converted to serve as the Spent Fuel Pool demineralizer. Because of radioactivity continuously removed from the recirculated water, the contact radiation levels will increase during the lifetime of each resin replacement. The contact dose rates appear to change at about 200 mR/hr for each month of operation.

4.28 Condensate Demineralizer Regeneration Room

4.28.1 This area is considered to be a highly contaminated area, with contamination levels ranging from 2,000 to 200,000 dpm/dm^2.

4.28.2 General area gamma radiation levels range from 6 to 30 mR/hr. Contact dose rates on the laundry waste tank (Northwest corner of room) range from 2 to 45 mR/hr. Contact dose rates on the cation tank (Northeast corner of room) range

from 30 to 800 mR/hr. Contact dose rates on the anion tank (Southeast corner of room) range from 6 to 40 mR/hr. The higher dose rates generally are found at or under the bottom of the tanks. Contact dose rates on the overhead resin piping are generally 30 mR/hr, with a hot spot of 100 mR/hr near the cation tank.

4.28.3 Beta radiation fields show 40 mRad/hr at the floor drain, 60 mRad/hr at the sight glass and 600 mRad/hr at a scupper (due to approximately 1/2 gallon of resin, noted to be in the scupper).

4.28.4 This summary is based on surveys performed from 9/8/86 through 12/22/86.

4.28.5 This area has a high potential for personnel contamination, as well as the potential for undesirable radiation exposures.

4.29 Radwaste Treatment Building - Tankage Area

4.29.1 This area has generally low levels of removable contamination (less than 500 dpm/dm^2) with the exception of the level instrumentation piping (for the CWTs and the RDT), near No. 3 WRT (3,000 to 6,000 dpm/dm^2). Removable contamination has also been found (20,000 to 50,000 dpm/dm^2) on the concentrator feed line valves (at the corner of the concrete wall, near the entrance to the tankage area).

4.29.2 Gamma radiation area doses rates range from 2 to 14 mR/hr. Highest readings are in No. 3 WRT area. Lowest readings are in No. 1 WRT and No. 1 WHT area. Contact dose rates on Nos. 1 and 2 WHTs are less than 5 mR/hr. Contact dose rates on No. 1 WRT are 3 to 12 mR/hr. Contact dose rates on No. 2 WRT are 35 to 70 mR/hr. Contact dose rates on No. 3 WRT are 35 to 150 mR/hr.

4.29.3 Beta radiation fields have been detected at the scupper near No. 3 WRT (52 mRad/hr) and on the Concentrator Drip Pump (22 mRad/hr).

4.29.4 This summary is based on surveys performed from 1/8/86 through 8/26/88.

4.29.5 This area has a low to moderate potential for personnel contamination.

4.29.6 There are contaminated hoses (reserved for future solidification projects) stored in plastic at the West side of the area. Note that although the typical liquid radioactive waste (in Nos. 1 and 2 WRT or in Nos. 1 and 2 WHT) is at about 1 x 10-5 μCi/ml, the material in No. 3 CWT has radioactivity levels of about 0.3 μCi/ml (^{137}Cs and ^{60}Co). Note also that the pipe trench (under the walkway at the South side of the area) was decontaminated, and then painted to fix remaining contamination.

4.30 Radwaste Treatment Building - Waste Concentrator Area

4.30.1 This area has high levels of removable contamination, with surfaces in this area showing widely varying levels of contamination. The levels range from 1,000 to 330,000 dpm/dm^2, with the highest levels near the concentrator.

4.30.2 Gamma radiation dose rates are approximately 20 mR/hr. Dose rates near the concentrator are approximately 45 mR/hr.

4.30.3 Beta radiation fields have been measured on both pumps (30 to 50 mRad/hr) and the floor drain (400 mRad/hr).

4.30.4 This summary is based on surveys performed from 2/4/86 through 7/1/88.

4.30.5 *This area has a high potential for personnel contamination, and it has the potential for undesirable personnel beta exposure .*

4.31 Radwaste Treatment Building - Sump Area

4.31.1 This area has high levels of removable contamination. Levels at the sump range from 26,000 to 70,000 dpm/dm^2 and 46 dpm/dm^2 alpha. Levels at the floor drain show 350 dpm/dm^2 beta-gamma and 250 dpm/dm^2 alpha. Levels near the Concentrator Feedpump range from 6,000 to 26,000 dpm/dm^2.

4.31.2 Gamma radiation dose rates are variable. Dose rates are highest at floor level (20 mR/hr) and decrease with height (3 mR/hr). A hot spot at the sump measures 40 mR/hr. The floor drain measures 55 mR/hr. Piping below the CCW Heat Exchanger has contact dose rates of about 30 to 50 mR/hr.

4.31.3 Beta radiation fields have been measured at the floor drain (2,000 mRad/hr), the sump (60 mRad/hr), and the Concentrator Feedpump (280 mRad/hr).

4.31.4 This summary is based on surveys performed from 1/8/86 through 7/1/88.

4.31.5 This area has a high potential for personnel contamination and undesirable personnel beta exposures.

4.32 Radwaste Treatment Building - Operating Area and Hallway

4.32.1 This area shows removable contamination levels from less than 500 dpm/dm^2 to about 2,000 dpm/dm^2.

4.32.2 Gamma radiation dose rates are generally less than 2 mR/hr. There are several hot spots (4 to 9 mR/hr).

4.32.3 Beta radiation was found at the floor drain between the radwaste pumps (110 mRad/hr), at the sample scupper (220 mRad/hr) and on the pump bases (60 to 70 mRad/hr).

4.32.4 This summary is based on surveys performed from 1/8/87 through 3/2/88.

4.32.5 This area has a moderate potential for personnel contamination.

4.33 Radwaste Treatment Building - Filter Room

4.33.1 This area shows removable contamination levels from less than 500 dpm/dm^2 to 1,000 dpm/dm^2.

4.33.2 Gamma radiation dose rates are generally 1 to 5 mR/hr. At the time of these surveys, the filter 'pig' had a surface reading of about 30 to 50 mR/hr. Note that with current (1989) filters, the 'pig' contact dose rate is about 2 to 5 mR/hr.

4.33.3 Beta radiation has been detected at the floor drain (40 mRad/hr).

4.33.4 This summary is based on surveys performed from 3/15/88 through 8/8/88.

4.33.5 This area has low potential for personnel contamination during normal conditions, the area is considered to have moderate to high potential for personnel contamination during the process of replacing expended filter cartridges.

4.34 Radwaste Treatment Building Roof

4.34.1 This area has low contamination levels (less than 500 dpm/dm^2).

4.34.2 Gamma radiation dose rates are less than 2 mR/hr. The resin transfer line from the Radwaste Demineralizer to the Resin disposal tank has general contact reading of about 5 mR/hr. The hotspot (about 100 mR/hr) that had been indicated on the Resin Disposal Tank vent line has been removed (by flushing back into the tank).

4.34.3 No beta radiation fields have been detected.

4.34.4 This summary is based on surveys performed from 2/5/86 through 9/26/88.

4.34.5 This area has low potential for personnel contamination, except possibly in the area of the trash compactor when waste handling is in progress.

4.35 Radwaste Treatment Building - CWT Tank Room

4.35.1 This area has very high removable contamination levels. Smears of walls and piping measured by 'pancake GM' indicate 40,000 to 100,000 dpm/dm^2, but most smears of the floor were measurable only with dose rate instruments (reading 8 to 15 mRad/hr per dm^2).

4.35.2 Gamma radiation dose rates decrease with height. The ceiling has the lowest dose rate. General area dose rates are on the order of 40 mR/hr. Contact dose rates on the tanks range from 30 to 150 mR/hr. Contact dose rates on piping range from 30 to 120 mR/hr.

4.35.3 Beta radiation was noted (320 mRad/hr) for a survey taken while the floor was flooded with concentrated liquid waste. After washing the area, no beta radiation fields were detected (using an instrument bagged to prevent contamination).

Based on the readings from the smears described above, beta fields of 20 to 200 mRad/hr would probably be observed near the floor.

4.35.4 This summary is based on surveys performed from 6/30/86 through 7/31/86.

4.35.5 This area has a high potential for personnel contamination, as well as personnel beta and gamma exposure.

4.35.6 Note that this area has been flooded with concentrated liquid waste at least twice in the history of the plant. The tanks were left essentially empty, so that this problem is not expected to reoccur.

4.36 High Level Storage Vaults

4.36.1 No. 1 Vault has removable contamination levels which range from less than 1,000 to 2,000 dpm/dm^2. No. 2 Vault has removable contamination levels which range from less than 1,000 to 7,000 dpm/dm^2. No. 3 Vault has removable contamination levels which range from less than 1,000 to 9,000 dpm/dm^2. One smear showed 20 dpm/dm^2 alpha.

4.36.2 No. 1 Vault gamma radiation dose rates are less than 0.2 mR/hr with the exception of a hot spot on the floor of 1.2 mR/hr gamma and 60 mRad/hr beta. No. 2 Vault gamma radiation dose rates are less than 0.2 mR/hr with the exception of the floor drain which shows 5 mR/hr gamma and 52 mRad/hr beta. No. 3 Vault gamma radiation dose rates are less than 0.6 mR/hr. Beta fields up to 140 mRad/hr were detected on the east floor and wall.

4.36.3 This summary is based on surveys performed from 1/31/86 through 10/20/86.

4.36.4 This area has moderate potential for personnel contamination.

4.36.5 Note that these vaults will be used for storage of higher dose rate waste (in 55-gallon drums).

4.37 Low Level Storage Building

4.37.1 This area is considered free of removable contamination (less than 500 dpm/dm^2).

4.37.2 Gamma radiation dose rates are generally less than 1 mR/hr. Note that dose rates in this building depend on the materials stored in the area, as there are no other significant radiation sources in the area.

4.37.3 No beta radiation fields have been detected.

4.37.4 This summary is based on surveys performed from 1/7/86 through 6/24/88.

4.37.5 This area has a low potential for personnel contamination.

4.37.6 The West end of the structure is used as a staging area for the accumulation of waste for packaging. This part of the area is also used for storage of ladders and scaffoldings that have fixed contamination. The East end of the structure is primarily used for 'SAFSTOR' storage of tools/equipment that is contaminated.

4.38 Radwaste Handling Building

4.38.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2) with the exception of the reactor head shield (4,000 dpm/dm^2 beta-gamma and 43 dpm/dm^2 alpha).

4.38.2 Gamma radiation dose rates are generally less than 0.2 mR/hr except for roped area in SE corner (5.0 mR/hr).

4.38.3 No beta fields have been detected with the exception of the reactor head shield (20 mRad/hr).

4.38.4 This summary is based on surveys performed from 1/31/86 through 8/9/88.

4.38.5 This area has low potential for personnel contamination.

4.39 Yard - 'Upper Area'

4.39.1 This area is considered to be generally free of removable contamination (less than 500 dpm/dm^2).

4.39.2 In most of the area, gamma dose rates are less than 0.2 mR/hr. Higher dose rates are found between the Low Level Storage Building and the Solid Waste Handling Building (about 3 mR/hr) due to material stored in both structures.

4.39.3 No beta radiation fields have been detected.

4.39.4 This summary is based on surveys performed from 4/3/86 through 9/26/88.

4.39.5 This area is not considered to have significant potential for personnel contamination.

4.39.6 Most of this area (with the exception of an area about 100' x 100' in the Northwest corner of the yard) had the pavement removed, soil sampled and discarded (if contaminated). The remaining soil was graded, selectively back-filled and then paved. In general, the soil below the pavement in this area is considered to have less than 1 pico-Curie/gm of ^{137}Cs or ^{60}Co.

4.40 Yard - From Liquid Radwaste Building to Refueling Building

4.40.1 This area is considered to be generally free of removable contamination (less than 500 dpm/dm^2).

4.40.2 In most of the area, gamma dose rates are less than 0.2 mR/hr. Higher dose rates are found along the South face of the Liquid Radwaste Treatment Facility

Enclosure (generally 0.5 to 2.5 mR/hr, with 3.4 mR/hr at ground level, in front of the middle roll-up door), around the Condensate Storage Tank (up to 10 mR/hr at contact with tank base) and around the amputated piping Southeast of the Stack (about 5 mR/hr). Hot spots were identified at the piping South of the CST (100 mR/hr) and on the amputated piping by the Stack (30 mR/hr), but some of this piping was removed since the survey (3/30/88).

4.40.3 No beta radiation fields have been detected.

4.40.4 The summary is based on surveys performed from 1/1/86 through 9/26/88.

4.40.5 This area is not considered to have significant potential for personnel contamination.

4.40.6 Most of this area (with the exception of concreted sections) had the pavement removed, soil sampled and discarded (if contaminated). The remaining soil was graded, selectively back- filled and then paved. The dirt bank at the East end of the area was coated with concrete to stabilize the bank. In general, the soil below the pavement/concrete in this area is considered to range from less than 1 to about
10 pico-Curie/gram of ^{60}Co or ^{137}Cs.

4.41 Yard - Southeast Area

4.41.1 This area is considered to be generally free of removable contamination (less than 500 dpm/dm^2).

4.41.2 In most of the area, gamma dose rates are less than 0.2 mR/hr. Higher dose rates are found at the entrance to the Condensate Pump Room (0.5 mR/hr), to the Condensate Demineralizer Pipe Gallery (0.8 mR/hr) and to the Condenser Bay (4.8 mR/hr).

4.41.3 No beta radiation fields have been detected.

4.41.4 The summary is based on surveys performed from 1/9/86 through 8/31/88.

4.41.5 This area is not considered to have significant potential for personnel contamination.

4.41.6 Most of this area had the pavement removed, soil sampled and discarded (if significantly contaminated). The railroad tracks and ties between the buildings and the gate were removed. The remaining soil was graded, selectively back-filled and then paved. In general, the soil below the pavement/concrete in this area is considered to range from less than 1 to about 100 pico-Curie/gram of ^{60}Co or ^{137}Cs.

4.42 Calibration Facility

4.42.1 This area is considered to be essentially free of removable contamination (less than 500 dpm/dm^2).

4.42.2 Gamma radiation dose rates are generally low (ranging from 0.2 mR/hr to 0.6 mR/hr), with hot spots on the source locker (6 mR/hr) and the cask for the 'old' ^{60}Co source (4 mR/hr).

4.42.3 No beta, radiation fields have been detected. Note there is a Uranium slab source in the source locker that reads about 120 mRad/hr.

4.42.4 This summary is based on surveys performed from 1/22/86 through 9/18/86.

4.42.5 This area has a low potential for personnel contamination.

4.42.6 There is a high potential for undesirable personnel exposures due to the presence of the two ^{60}Co sources (one in indefinite storage pending disposal, and the other installed in the source calibration well). Note that when the source in the well is unshielded and moved to its highest location there is a dose rate of about 25 R/hr at the opening of the well, and the potential of several hundred mR/hr on top of the roof.

4.43 Hot Shop

4.43.1 The area on the clean side of the step-off-pad is considered to be essentially free of removable contamination (less than 500 dpm/dm^2). Most of the remaining area of the shop has generally low levels of removable contamination (less than 1,000 dpm/dm^2) with the exception of the lathe (up to 3,000 dpm/dm^2 beta-gamma and 64 dpm/dm^2 alpha) and with exception of the washdown area (up to 40,000 dpm/dm^2 and 20 dpm/dm^2 alpha).

4.43.2 Gamma radiation dose rates are less than 0.2 mR/hr, with the exception of the floor drains (5 mR/hr).

4.43.3 No beta radiation fields have been detected, with the exception of the lathe (0.4 mRad/hr) and the floor drains (12 mRad/hr at the one in the washdown area).

4.43.4 This summary is based on surveys performed from 1/10/86 through 8/12/88.

4.43.5 Most of this area has a low potential for personnel contamination. The lathe has a moderate potential for personnel contamination, and the washdown area has a high potential for personnel contamination.

4.44 Seal Oil Room

4.44.1 This area is generally considered to be free of removable contamination with the exception of the center floor drain (2500 dpm/dm^2 beta-gamma and 11.8 dpm/dm^2 alpha).

4.44.2 Gamma radiation levels are less than 0.2 mR/hr.

4.44.3 No beta radiation fields have been detected with the exception of the center floor drain (0.4 mRad/hr).

4.44.4 This summary is based on surveys performed from 1/6/86 through 9/10/88.

4.44.5 This area has a low potential for personnel contamination.

4.45 Hydrogen Yard

4.45.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2), with the exception of the instrument vault, where 12,000 dpm/dm^2 has been observed.

4.45.2 Gamma radiation levels are less than 0.2 mR/hr.

4.45.3 No beta radiation has been detected.

4.45.4 This summary is based on surveys performed from 1/6/86 through 9/10/88.

4.45.5 This area is generally considered not to have potential for personnel contamination.

4.45.6 Note that the Battery Room (batteries removed) was constructed about 1976, and is believed to be not contaminated. At the time of the latest survey, the generator Hydrogen Cooler (Closed Cooling Water heat exchanger) is stored in a box to the West of the Battery Room. The Hydrogen Cooler is believed to be uncontaminated.

4.46 Reactor Feedwater Pump Room

4.46.1 This area is generally considered to be free of removable contamination (less than 500 dpm/dm^2). An exception is the troughs and scuppers of the two reactor feedwater pumps where some removable contamination was found (approximately 2,000 dpm/dm^2).

4.46.2 Gamma radiation levels are generally less than 0.2 mR/hr, except in the vicinity of the feedwater pumps and piping. Radiation levels near the pumps and piping are 1 mR/hr. Contact dose rates on the piping are 2 to 5 mR/hr. An overhead elbow SE of No. 2 Reactor Feedpump has a contact dose rate of 20 mR/hr.

4.46.3 No Beta radiation fields have been detected in this area.

4.46.4 This summary is based on various surveys taken during the period of 1/6/86 through 9/10/88.

4.46.5 This area is considered to have low potential for personnel contamination, exclusive of any material retained on the interior of the feedwater piping. No significant contamination has been found in the overhead portions of the room.

4.46.6 Note that the below grade conduit run from pullbox 02 towards the radwaste areas has been contaminated.

4.47 Propane Engine Generator Room

4.47.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2).

4.47.2 Gamma radiation levels are less than 0.1 mR/hr.

4.47.3 No beta radiation fields have been found in this area.

4.47.4 This summary is based on surveys performed from 2/4/86 through 7/6/88.

4.47.5 This area is not considered to have potential for personnel contamination.

4.47.6 During operation, this area was outside the Radiological Control Area, but access was changed during preparation for SAFSTOR. To expedite maintenance, this area will probably be released for uncontrolled access, since no radioactivity has been detected.

4.48 Recombiner Vault

4.48.1 This area is considered to be free of removable contamination (measurements show less than 500 dpm/dm^2).

4.48.2 Gamma radiation dose rates are less than 0.1 mR/hr except for a field of 0.1 mR/hr at the end of the tunnel toward the original offgas filter. This field is probably due to radiation levels south of the wall.

4.48.3 No beta radiation fields have been found in this area.

4.48.4 This summary is primarily based on a survey performed 6/23/88.

4.48.5 This area is not considered to have personnel contamination potential, since the area was constructed clean and the condenser off-gas treatment equipment was neither connected to the condenser off-gas line nor operated. (The high-pressure steam supply, cooling water, vents, and drains were connected to Unit 3 systems after shutdown in 1976, and disconnected for SAFSTOR in 1984. The sump pump discharge was connected in 1976, and remains connected). No radioactivity has been detected on accessible surfaces in this area. Traces of removable contamination, below 1000 dpm/ dm^2, had been detected inside some steam and drain piping when the system equipment and piping were removed in 2002.

4.49 Former Stack

4.49.1 The normally accessible levels of the former stack (+0', +12' and +26'6") are considered to be free of removable contamination (at or below 500 dpm/dm^2).

4.49.2 Gamma radiation levels are generally less than 0.05 mR/hr on the +12' elevation. The +26'6" elevation radiation levels are generally less than 0.2 mR/hr, increasing

toward a P-trap on the south wall of the area. Contact dose rates on the P-trap are 1 mR/hr. The +0, elevation radiation levels are generally less than 0.5 mR/hr, except toward a hot spot on the elbow of the Drywell Purge line on the south wall of the area. This line has a contact dose rate of 80 mR/hr.

4.49.3 Beta radiation has been detected on the Drywell purge fan and on the floor drains.

4.49.4 This summary is based on various surveys taken during the period of 1/11/86 through 7/8/88.

4.49.5 This area has low potential for personnel contamination from the interior surfaces of the liquid/gaseous portions of the gas scrubber system.

4.49.6 Note that there appears to be a significant amount of radioactive material collected in the Drywell Purge line. The dose rate at the P-trap on the upper elevation is probably the result of material washed down from the upper level of the stack. This trap is supposed to have water in it, but since it is probably dry, any air flow through it may be a source of airborne contamination.

4.50 Generator and Exciter Housing

4.50.1 This area (including the access on both sides, and the catwalk to the control room) is free of removable contamination (less than 500 dpm/dm^2).

4.50.2 Gamma radiation dose rates are less than 0.2 mR/hr.

4.50.3 No beta radiation fields have been found in this area.

4.50.4 This summary is based on surveys performed from 1/20/86 through 2/8/88.

4.50.5 This area is not considered to have potential for personnel contamination. The area is now outside the Radiological Controlled Area (released on 2/8/88).

4.51 Turbine Enclosure

4.51.1 This area is considered to be generally free of removable contamination (less than 500 dpm/dm^2). The contamination levels within the protective cover over the turbine control valve mechanism are not available.

4.51.2 Gamma radiation levels range from <0.2 mR/hr (South end of area) to about 1.5 mR/hr (at the grating toward the North end of the area).

4.51.3 No beta fields have been found in this area.

4.51.4 This summary is based on surveys performed from 1/9/86 through 4/27/88.

4.51.5 This area is considered to have low potential for personnel contamination.

4.51.6 Although smear surveys of portions of the turbine surface do not show contamination, the turbine is roped off as a contaminated area, because of the complexity of the piping precluded a completely detailed survey.

4.52 Turbine Laydown Area

4.52.1 This area is considered to be generally free of removable contamination (less than 500 dpm/dm^2).

4.52.2 Gamma radiation levels are less than 0.2 mR/hr.

4.52.3 No beta fields have been found in this area.

4.52.4 This summary is based on surveys performed from 9/8/86 through 4/27/88.

4.52.5 This area is considered to have low potential for personnel contamination.

4.52.6 The materials in this area ('Rail Car' and the I-beams) have low levels of fixed contamination, but do not appear to have significant removable contamination.

4.53 Hot Lab

4.53.1 This area is considered to be reasonably free of removable contamination (less than 500 dpm/dm^2) with occasional exceptions, such as the sinks (2,000 and 4,000 dpm/dm^2) and at least one drawer (2,500 dpm/dm^2).

4.53.2 Gamma radiation dose rates are less than 0.2 mR/hr.

4.53.3 No beta radiation fields have been detected other than the sink in the Northeast corner of the room (12 mRad/hr).

4.53.4 This summary is based on surveys performed from 1/11/86 through 6/30/88.

4.53.5 This area is considered to have a low potential for personnel contamination.

4.54 Laundry

4.54.1 This area is considered to be reasonably free of removable contamination (less than 1,000 dpm/dm^2) with the exception of the internal parts of the washers, dryers, and the exhaust to the ventilation filter. Some contamination has been detected, such as the inside of No. 1 Dryer (5,000 dpm/dm^2) and an area in front of No. 2 Dryer (1,000 dgm/dm^2).

4.54.2 Gamma radiation dose rates are leas than 0.2 mR/hr.

4.54.3 No beta radiation fields are detected.

4.54.4 This summary is based on surveys performed from 1/11/86 through 6/30/88.

4.54.5 This area is considered to have low potential for personnel contamination.

4.54.6 Note that several bags of slightly contaminated (but laundered) cloth coveralls are stored in this area.

4.55 Security Area (Eastern portion of +27 elevation)

4.55.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2).

4.55.2 Gamma radiation dose rates are less than 0.2 mR/hr.

4.55.3 No beta radiation fields have been detected, with the exception of the floor drains (less than 0.8 mRad/hr).

4.55.4 This summary is based on surveys performed from 1/11/86 through 7/21/88.

4.55.5 This area is not considered to have significant potential for personnel contamination.

4.56 Access Control (Western portion of RCA at +27' elevation)

4.56.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2).

4.56.2 Gamma radiation dose rates are less than 0.2 mR/hr.

4.56.3 No beta radiation fields have been detected. Note that there is a potential for beta exposure from the ^{90}Sr check source mounted on the Turbine Shield wall.

4.56.4 This summary is based on surveys performed from 1/11/86 through 8/3/88.

4.56.5 This area is not considered to have significant potential for personnel contamination.

4.57 Control Room and Instrumentation Shop

4.57.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2).

4.57.2 Gamma dose rates are less than 0.2 mR/hr.

4.57.3 No beta radiation fields have been detected.

4.57.4 This summary is based on surveys performed from 2/6/86 through 8/8/88.

4.57.5 This area is not considered to have potential for personnel contamination.

4.58 Count Room

4.58.1 This area is considered to be free of removable contamination (less than 500 dpm/dm^2).

4.58.2 Gamma radiation dose rates are less than 0.2 mR/hr.

4.58.3 No beta radiation fields have been detected.

4.58.4 This information is based on surveys performed from 2/21/86 through 8/19/88.

4.58.5 This area is considered to have a low potential for personnel contamination.

4.59 Batteries, Multi-Zone Fan Area and Hot Lab 'Attic'

4.59.1 The portion of the area accessible from the Counting Room is considered to be free of removable contamination (less than 500 dpm/dm^2). The portion of the area accessible from the ladder near.the Hot Lab door is considered to have low levels of removable contamination (less than 1,000 dgm/dm^2).

4.59.2 Gamma radiation dose rates are less than 0.2 mR/hr.

4.59.3 No beta radiation fields have been detected.

4.59.4 This summary is based on surveys performed from 10/7/86 through 8/29/88.

4.59.5 This area is not considered to have significant potential for personnel contamination.

4.60 Suppression Chamber

4.60.1 The suppression chamber is closed with bolted hatchways.

4.60.2 This is a highly contaminated area. Contamination levels average 40,000 dpm/dm' with highest measurements of 100,000 to 2,000,000 dpm/dm^2.

4.60.3 Gamma radiation levels are on the order of 3 to 11 mR/hr. An I-beam showed a hot spot of 140 mR/hr.

4.60.4 There are a few beta radiation fields of 4 mRad/hr. The I-beam hot spot measured 560 mRad/hr.

4.60.5 This summary is based on surveys performed from 1/27/86 through 3/25/86.

4.60.6 This area has a high potential for personnel contamination, for high airborne contamination and undesirable personnel exposures.

5.0 ONSITE RADIOACTIVE WASTE INVENTORY

5.1 As of January, 1989, the most significant inventory of radioactive waste is the resin in the resin disposal tank. The contents of the tank are considered to be about 70 Curies (primarily ^{137}Cs), distributed in 196 cubic feet of resin. At the currently estimated rate at which the Spent Fuel Pool Demineralizer resin requires replacement, the tank ^{137}Cs inventory will increase at a rate of about 15 Curies per year (and about 40 cubic feet per year).

5.2 The inventory (1/89) of liquid concentrated waste in No. 3 Waste Receiver Tank is about 2500 gallons of liquid totaling about 1.7 Curies for ^{137}Cs and ^{60}Co (approximately 60:40 for ^{137}Cs: ^{60}Co).

5.3 There are a variety of waste packages onsite, pending shipment. As of January, 1989, this consisted of 27 drums and 12 boxes. One of the drums contains filter cartridges from the original Spent Fuel Pool Filter, and has a contact dose rate of about 1.3 R/hr. Two other drums (containing radwaste system filter cartridges) have contact dose rates of 110 mR/hr and 12 mR/hr. These three drums are stored in the No. 3 High Level Storage Vault. The rest of the packages are stored in the Solid Waste Handling Building. The estimated activity of the drums in the vault is 1.5 Curies, and the estimated activity of the other packages is less than 0.5 Curies.

5.4 These activity estimates are made on the basis of gamma emitting nuclides. A review of the mixtures used for shipping indicates that the combined quantities of other nuclides (^{3}H, ^{14}C, ^{63}Ni and ^{90}Sr) will normally be less than the combined activity of ^{137}Cs and ^{60}Co (except that after about 5 years, ^{63}Ni may be about equal to ^{60}Co).

6.0 ENVIRONMENTAL RADIOACTIVITY LEVELS

6.1 The Annual Environmental Monitoring Report and the SAFSTOR Environmental Report (Attachment 6 to the SAFSTOR license amendment request) provide comprehensive documentation of environmental radioactivity levels.

6.2 A concrete 'pad' adjacent to the Unit 2/3 fenceline was found to have fixed contamination (^{137}Cs). Further surveys/sampling will be required to define the extent of the contamination.

Appendix E

RADIOLOGICAL CHARACTERIZATION

RADIONUCLIDE INVENTORY

The largest percentage of the onsite radionuclide inventory is contained in the spent fuel, with the reactor vessel and internals containing the next largest percentage. Radionuclides are also present in corrosion films within various in-plant systems.

These radionuclide sources are not readily dispersible in their present condition and will continue to decay during the SAFSTOR period.

Additional contributions to the radionuclide inventory are those sources external to the closed systems addressed above. These sources include fixed and removable surface contamination; the radionuclides contained in the spent fuel pool water, and associated systems.

Spent Fuel. During January and February 1984, all fuel and fission chambers were removed from the reactor vessel. Currently there are 390 spent fuel assemblies stored in the spent fuel pool. Incore fission chambers with a total of less than 1 gram of ^{235}U are also stored in the pool. The minimum decay and cooling time for any element stored in the pool is slightly less than 18 years. Characteristics of the stored spent fuel are summarized in Table E-1. Miscellaneous items that are stored in the spent fuel pool are listed in Table E-2.

The 140 spent fuel assemblies removed from the reactor in February 1984 have exposures ranging from 5,009 to 15,492 megawatt-days per metric ton of uranium. The 250 fuel assemblies previously stored had exposures ranging from 14,400 to 19,481 megawatt-days per metric ton of uranium. The fuel types include GE Types II and III-1 through III-4, as well as Exxon Types III and IV. There are no GE Type I, stainless steel-clad assemblies remaining on site. However, cladding failure of the Type I assemblies during the early years of operation has contaminated the spent fuel pool.

The radioactive deposits on the spent fuel pool walls, racks, and related equipment are described in Table E-5. The total activity is estimated at 3.2 Ci with ^{137}Cs (2.1 Ci) comprising 70 percent. ^{55}Fe is present at 27 percent (690 mCi) and gamma-emitting ^{60}Co (120 mCi) is present at 4 percent.

The pool contains 110,000 gallon of water with an average concentration of 0.005 μCi/ml. ^{137}Cs (1.76 Ci) comprises 94 percent of the activity, while transuranics (12 μCi) comprise less than 0.007 percent of the spent fuel pool water inventory of 2.0 Ci.

Reactor Vessels and Internals. Table E-3 shows the radionuclide inventory calculated by Gibbs and Hill (1982) and PG&E for the reactor vessel, corrected for decay to mid-1984 conditions. The highest nuclide inventory is contained in the chimney guide and chimney with an inventory of 4,900 Ci. The 32 control rod blades and the core support grid structure were estimated to have the next greatest inventory, estimated at 2,200 Ci each. These

structures account for 78 percent of the total inventory. Within the reactor vessel and internals, the primary controlling nuclide is gamma-emitting ^{60}Co. Because of the predominance of ^{60}Co and ^{55}Fe and their relatively short half-lives (5.27y and 2.7y, respectively), the activity of the reactor vessel and internals will decrease to 2,900 Ci over the 30-year SAFSTOR period.

In-Plant Systems. Internal and external contamination of plant systems have been characterized by samples and surveys taken by Pacific Northwest Laboratory (PNL-4628) and by recent characterization efforts, as well as routine PG&E characterization and radiological monitoring surveys.

The corrosion film radionuclide inventory estimates are presented in Table E-4. These estimates have been corrected for decay from values determined by PNL's 1981 surveys (PNL-4628). The most abundant radionuclides in the corrosion films associated with reactor piping and components are ^{55}Fe, ^{60}Co, ^{137}Cs and ^{63}Ni. Transuranics are estimated to constitute about 0.04 percent of the total residual radionuclide inventory, or about 30 mCi.

Within the piping systems, the major radionuclide repository is contained in the shutdown cooling system, where the radionuclide inventory is about two times greater than in the radwaste or reactor water cleanup system piping. The main steam lines contain an order of magnitude less ^{55}Fe than in the shutdown cooling system and relatively minor inventory resides in the feedwater, condensate, and emergency condenser piping.

In the nonpiping systems, the largest radionuclide inventory is in the main condenser (26 Ci ^{55}Fe), which is about 2.5 times greater than the inventory estimate for the reactor shutdown cooling system cooler. The largest transuranic inventory is contained in the reactor shutdown cooling system cooler (4.5 mCi of ^{241}Am). Table E-5 summarizes the radionuclide inventory estimates by system components.

Surface Contamination. Analyses of concrete cores conducted by PNL in 1981 indicate that the radionuclide contamination resides mainly in the top centimeter of concrete (see Table E-6). The exception to this is where cracks have occurred and the contaminants have migrated into the concrete. The primary radionuclides in the concrete surfaces are ^{137}Cs and ^{60}Co. The ^{137}Cs appears to have penetrated painted surfaces and migrated into cracks to a greater extent than ^{60}Co. The greatest concentrations of ^{137}Cs are in the radwaste treatment building area. ^{60}Co concentrations are highest at the -66 foot elevation of the reactor caisson access shaft and in the condensate demineralizer room.

All radionuclide concentrations in the concrete are two-to-three orders of magnitude below the maximum allowable concentrations for Class A waste under 10 CFR 61. These concentrations will be further reduced due to decay during the SAFSTOR period, particularly ^{60}Co with a half-life of 5.3 years. The highest total concentrations of transuranics analyzed by PNL were 2.9 nanocuries/g.

The activity in the external surface contamination is estimated to be 1.1 Ci assuming the ratio of internal to external contamination reported for NUREG/CR-0672, "Technology, Safety and Cost of Decommissioning a Reference Boiling Water Reactor." Surveys of radiation and contamination levels in Unit 3 and related structures are presented in Table E-7. An inventory estimate based on core and swipe analyses for radionuclides associated

with concrete or other surficial sources has not been attempted. Such an estimate would be unreliable since it would contain sources of extreme variability of radionuclide concentrations and distributions.

Sealed Sources. A 4 Ci ^{60}Co source and a 14 Ci ^{60}Co source are stored in the calibration facility. Several other small check/calibration sources are also routinely used.

REFUELLING BUILDING/POWER BUILDING

Decontamination of the Unit 3 facilities will be an ongoing process, with systems and components drained, flushed, and partially decontaminated as appropriate. These measures and maintenance of the integrity of contamination barriers will minimize radiation and contamination levels throughout the SAFSTOR period. Tables E-7 and E-8 describe the radiological conditions existing within the refueling building and the power building, respectively.

YARD STRUCTURES

Structures within the yard of Unit 3 are the hot machine shop/calibration facility, radwaste treatment and storage buildings, the 40-foot base of the former ventilation stack, and the recombiner vault. Table E-9 presents radiation and contamination level surveys of various structures in the yard.

CROSS-CONNECTIONS

Existing cross-connections provide a possible pathway for cross-contamination between Units 1/2 and Unit 3. In response to NRC's IE Bulletin No. 80-10 of May 6, 1980, a review of possible cross-connections was undertaken and identified several systems where cross-contamination may occur. A routine sampling analysis program was initiated to ensure that potential problems would be readily identified and that releases would be minimized and documented.

Table E-10 presents the radionuclide analysis of systems with cross-contamination potential. In all systems, the contamination has been well below limits established by 10 CFR 20.

ENVIRONMENTAL RADIOLOGICAL CHARACTERISTICS

Background activity levels are currently found in the soils, canal and slough sediments, bay sediments, terrestrial and aquatic plants, and bay mussels (see Environmental Report, Section 5.6) sampled from outside the Unit 3 restricted area.

Within the Unit 3 restricted area, total beta-gamma dose rates range from background to 100 times background, primarily due to shine from the refueling building and the radwaste treatment building. Concentrations of ^{137}Cs, ^{60}Co, and other isotopes are slightly above

background in core samples of soils very near the refueling building (see the Environmental Report, Section 5.6). These concentrations are primarily the result of spills on the soil surface, leakage from buried transfer pipes, and leakage from the spent fuel storage pool. However, the concentrations are low and do not require remediation for the SAFSTOR period. No significant airborne re-suspension is associated with the minimal contamination.

SUMMARY

As of July 1984, the largest inventory of radionuclides is contained in the spent fuel rods with an estimated activity of 1.2E+6 Ci. Within 30 years, this activity will be less than 5.0E+5 Ci due to the decay of ^{137}Cs and ^{90}Sr with half-lives of approximately 30 years.

The largest radionuclide inventory outside of the spent fuel pool consists of activation products in the reactor vessel and surrounding structures. The total activity is estimated at 12,000 Ci. The primary radionuclide is ^{60}Co with an activity of 7,100 Ci (61 percent of the total.) During the SAFSTOR period the ^{60}Co activity will decrease to 110 Ci, and ^{63}Ni with an activity of 2,700 Ci will be the most abundant radionuclide in the reactor vessel system. Since the primary source of exposure is currently from ^{60}Co, the exposure rate will continue to decrease during the SAFSTOR period.

Other than several sealed sources and surface contamination, the remaining inventory is estimated at 100 Ci contained in corrosion films in various piping and components. The primary radionuclides are ^{55}Fe (81 percent) and ^{60}Co (14 percent). After 30 years, the primary radionuclides will be ^{63}Ni and ^{137}Cs, which will comprise 90 percent of the estimated remaining 2.5 Ci. The transuranic radionuclide inventory is estimated at 32 mCi (0.04 percent). The most abundant transuranic nuclide is ^{241}Am, which comprises 38 percent of the corrosion film transuranic inventory.

Table E-1
Spent Fuel Inventory July 1984[a]

Isotope	Half-Life (Years)		Activity (Ci)		Isotope	Half-life (Years)		Activity (Ci)	
^{137}Cs	3.0	E+1	4.5	E+5	^{125}Sb	2.7	E+0	2.3	E+2
^{90}Sr	2.9	E+1	3.3	E+5	^{99}Tc	2.1	E+5	6.7	E+1
^{241}Pu	1.5	E+1	2.9	E+5	^{242}Cm	4.5	E-1	1.8	E+0
^{134}Cs	2.1	E+0	3.4	E+4	^{113m}Cd	1.5	E+1	5.8	E+0
^{85}Kr	1.1	E+1	2.7	E+4	^{93}Zr	1.5	E+6	1.4	E+0
^{244}Cm	1.8	E+1	2.2	E+4	^{151}Sm	9.3	E+1	3.8	E-1
^{238}Pu	8.8	E+1	1.3	E+4	^{126}Sn	1.0	E+5	1.7	E-1
^{106}Ru	1.0	E+0	3.8	E+3	^{79}Se	6.5	E+4	1.3	E-1
^{147}Pm	2.6	E+0	3.6	E+3	^{135}Cs	2.3	E+6	8.4	E-2
^{240}Pu	6.5	E+3	2.0	E+3	^{152}Eu	1.3	E+1	8.1	E-2
^{144}Ce	7.8	E-1	1.8	E+3	^{121m}Sn	5.0	E+1	3.9	E-2
^{3}H	1.2	E+1	1.7	E+3	^{93m}Nb	1.4	E+1	5.0	E-2
^{154}Eu	8.2	E+0	1.3	E+3	^{107}Pd	6.5	E+6	3.4	E-2
^{239}Pu	2.4	E+4	1.3	E+3	^{119m}Sn	8.0	E-1	1.2	E-2
^{241}Am	4.3	E+2	7.2	E+2	^{129}I	1.6	E+7	9.4	E-3
^{155}Eu	4.8	E+0	5.6	E+2	^{127m}Tm	3.0	E-1	7.5	E-4

a Activities based on the following: Westinghouse BLWE-1154, April 17, 1980, adjusted for burnup fraction and power; General Electric, 6EB-80/027, October 31, 1980, adjusted for burnup fraction and power; and W. B. Wilson, et al, Extended Burnup Calculations for Operating Reactor Reload Reviews, NUREG/CR-3108, LA-9563-MS, Los Alamos National Laboratory, February 1983, adjusted for burnup fraction and power.

Table E-2
Spent Fuel Pool Miscellaneous Inventory - 1984

	ESTIMATED CURIE CONTENT				
COMPONENT	^{55}Fe	^{60}Co	^{63}Ni	Other Nuclides	Total Curies
Incore Instrument Strings[a] (< 1 gram ^{235}U)	90	98	69	<1	260
Stellite Rollers[b]	< 15	8,000	11	-	8,200
Canned Waste	-	-	-	-	-
Sealed Sources					
Sb-Be Operating Sources (2)					< 1E-6

a Based on Oaks, H.D., Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, Vols. 1 and 2.

b Based on neutron activation calculations and sample analysis

Table E-3
Reactor Vessel Inventory of Radionuclides Corrected for Decay for Conditions Mid-1984

	Estimated Curie Content				
Reactor Internal Components	^{60}Co	^{55}Fe	^{63}Ni	Other Nuclides[a]	Total Curies
Chimney Guide and Chimney[b]	3.1E+3	1.7E+2	1.6E+3	1.7E+1	4.9E+3
Core Shroud[b]	5.3E+2	2.9E+1	2.7E+2	3.0E+0	8.3E+2
Core Support & Grid[b]	1.4E+3	7.7E+1	7.3E+2	7.0E+0	2.2E+3
Fuel Support Plates[b]	9.9E+2	5.4E+1	5.1E+2	5.0E+0	1.6E+3
Control Rod Guide Tubes[b]	6.3E+1	3.7E+0	3.3E+1	< 1E+0	1.0E+2
Control Rod Blades[c]	9.4E+2	1.0E+3	2.3E+2	3.1E+0	2.2E+3
Reactor Vessel & Clad[b]	6.9E+1	5.0E+1	9.0E+0	3.0E+0	1.3E+2
Drywell Vessel Wall[b]	< 1E+0	< 1E+0	< 1E+0	< 1E+0	< 1E+0
Drywell Concrete & Rebar	< 1E+0	< 1E+0	< 1E+0	< 1E+0	< 1E+0
Totals	7.1E+3	1.4E+3	3.4E+3	3.8E+1	1.2E+4

a Not corrected for decay since identities not reported.

b Gibbs and Hill, Inc. Decommissioning and Decontamination Study Humboldt Bay Unit 3, March 1982

c Calculated from Oak, H.D., et.al. Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR 0672, Vol.2, June 1980

Table E-4
Corrosion Film Radionuclide Inventory[a]
Corrected to July 1984
(PNL - 4628)

Radionuclide	Half-Life (years)	Inventory (curies)
^{55}Fe	2.7E+0	6.9E+1
^{60}Co	5.3E+0	1.2E+1
^{137}Cs	3.0E+1	2.1E+0
^{63}Ni	1.0E+2	1.4E+0
^{90}Sr	2.9E+1	1.7E-2
^{241}Am	4.3E+2	1.2E-2
^{238}Pu	8.8E+1	6.8E-3
$^{239, 240}Pu$	2.4E+4	6.1E-3
^{244}Cm	1.8E+1	4.4E-3
^{54}Mn	8.6E-1	3.3E-3
^{242}Cm	4.5E-1	1.1E-6

a Excluding the reactor vessel, biological shield, concrete surfaces, and residues in tanks and sumps

Table E-5
Radionuclide Inventory Estimates for Humboldt Bay Reactor Systems, July 1984 (mCi)
(Based on Data from PNL, 1983)

	^{54}Mn	^{60}Co	125 Sb	134 Cs	137 Cs	155 Eu	^{55}Fe mCi	^{63}Ni	^{90}Sr	^{99}Tc	239, ^{240}Pu	238 Pu	^{241}Am	244 Cm	TOTAL
NUCLEAR STEAM SUPPLY															
<u>Reactor Cleanup</u>															
Piping	1.1E+0	1.2E+3	3.4E+0	3.5E-1	3.1E+0	6.1E-1	3.1E+3	1.5E+2	1.1E+0	8.5E-2	7.0E-1	6.9E-1	1.3E+0	3.5E-1	4.5E+3
Regenerative Heat Ex.	1.4E+0	1.6E+3	4.4E+0	3.0E-1	4.0E+0	6.0E-1	4.0E+3	1.9E+2	1.7E+0	1.2E-1	9.2E-1	9.2E-1	1.7E+0	4.6E-1	5.8E+3
Resin Storage Tank	1.6E-2	6.0E+0	7.5E-2	3.0E-3	5.3E-1	5.0E-3	2.4E+3	6.3E-1	1.6E-2	1.6E-1	3.1E-3	3.7E-3	7.6E-3	8.9E-3	2.4E+3
<u>Reactor Shutdown Cooling System</u>															
Piping	2.0E+0	2.3E+3	6.1E+0	6.4E-1	5.7E+0	1.2E+0	5.6E+0	2.7E+2	2.3E+0	1.7E-1	1.3E+0	1.3E+0	2.5E+0	6.6E-1	2.6E+3
Reactor Shutdown Cooler	3.6E+0	4.2E+3	1.1E+1	1.0E+0	1.0E+1	2.0E+0	1.0E+4	5.0E+2	4.2E+0	3.0E-1	2.3E+0	2.4E+0	4.5E+0	1.2E+0	1.5E+4
<u>Emergency Condenser System</u>															
Piping	1.5E-4	8.3E-1	2.0E-3	1.1E-3	3.7E-3	5.2E-4	7.9E+0	8.0E-2	2.5E-3	9.0E-3	2.0E-4	4.3E-4	3.5E-4	3.5E-4	8.8E+0
Emergency Condenser	3.1E-4	1.7E+0	4.0E-3	1.4E-3	7.0E-3	1.0E-3	1.6E+1	2.0E-1	4.0E-3	2.0E-2	4.1E-4	8.6E-4	7.4E-4	7.4E-4	1.8E+1
<u>Suppression Tank Cooling and Core Spray System</u>															
Suppression Chamber	6.7E-3	4.4E+0	3.0E-3	1.4E-2	6.5E-1	1.0E-3	9.0E+0	3.8E-1	1.0E-1	3.0E-1	1.5E-3	1.9E-3	2.5E-3	1.3E-3	1.5E+1
Suppression Cooler	3.0E-4	1.8E-1	5.0E-4	7.0E-4	2.7E-2	6.0E-4	3.9E-1	1.6E-2	5.0E-3	1.3E-2	6.3E-5	8.0E-5	1.1E-4	5.6E-5	6.3E-1
TURBINE PLANT SYSTEM															
<u>Turbine System</u>															
Piping	3.6E-1	4.9E+1	8.0E-2	1.8E-2	1.3E-1	9.2E-3	4.5E+2	4.3E+1	1.1E-1	5.4E-1	1.1E-2	2.4E-2	2.0E-2	2.1E-2	5.4E+2

Table E-5 (Continued)
Radionuclide Inventory Estimates for Humboldt Bay Reactor Systems, July 1984 (mCi)
(Based on Data from PNL, 1983)

	^{54}Mn	^{60}Co	^{125}Sb	^{134}Cs	^{137}Cs	^{155}Eu	^{55}Fe mCi	^{63}Ni	^{90}Sr	^{99}Tc	239, ^{240}Pu	^{238}Pu	^{241}Am	^{244}Cm	TOTAL
CONDENSATE SYSTEM															
Piping	2.4E-3	1.2E+1	2.2E-2	1.6E-3	4.5E-2	3.1E-2	1.2E+1	1.0E+1	1.3E-2	1.0E-2	4.3E-2	2.4E-2	1.1E-1	1.4E-2	3.4E+1
Main Condenser	2.1E+1	2.7E+3	5.0E+0	1.0E+0	7.0E+0	6.0E-1	2.6E+4	2.7E+2	6.1E+0	3.1E+1	6.8E-1	1.4E+0	1.2E+0	1.2E+0	2.9E+4
Condensate Demin.	7.7E-3	3.0E+0	3.8E-2	1.5E-3	2.6E-1	3.0E-3	1.2E+3	3.1E+3	7.0E-3	8.0E-2	1.6E-3	1.9E-3	3.8E-3	4.4E-3	1.2E+3
Feedwater System															
Piping	8.5E-4	5.1E+0	5.8E-2	3.6E-3	6.3E-2	6.5E-3	6.5E-1	6.4E-1	3.1E-2	8.0E-3	2.5E-2	1.7E-2	4.2E-2	3.6E-3	6.6E+0
WASTE DISPOSAL SYSTEM															
Liquid Waste Treatment															
Piping	2.4E-2	9.1E+0	1.1E-1	4.4E-3	8.0E-1	7.8E-3	3.7E+3	9.4E-1	2.5E-2	2.4E-1	4.7E-3	5.7E-3	1.1E-2	1.3E-2	3.7E+3
Waste Receiver Tank	3.3E-2	1.3E+1	1.6E-1	7.0E-3	1.1E+0	1.0E-2	5.0E+3	1.3E+0	4.0E-2	3.3E-1	6.6E-3	7.8E-3	1.6E-2	1.9E-2	5.0E+3
Conc. Waste Tank	2.1E-2	8.2E+0	1.1E-1	4.0E-3	7.3E-1	7.1E-3	3.3E+3	8.6E-1	2.1E-2	2.2E-1	4.3E-3	5.1E-3	1.0E-2	1.2E-2	3.3E+3
Waste Hold Tank	2.2E-2	8.4E+0	1.1E-1	4.0E-3	7.4E-1	7.8E-3	3.3E+3	8.7E-1	2.2E-2	2.2E-1	4.4E-3	5.2E-3	1.1E-2	1.2E-2	3.3E+3
SERVICE SYSTEM															
Spent Fuel Service System															
Fuel Basin-Walls	2.1E-1	4.7E+1	5.0E-1	3.1E+1	7.8E+2	2.9E-1	3.3E+2	1.0E+1	3.0E-1	2.7E+0	3.8E-2	4.2E-2	2.4E-1	1.4E-1	1.2E+3
- Racks	3.0E-1	6.6E+1	6.5E-1	4.3E+1	1.1E+3	4.0E-1	4.5E+2	1.4E+1	4.1E-1	3.7E+0	5.1E-2	5.8E-2	3.3E-1	1.9E-1	1.7E+3
Fuel Pool Cooler	4.6E-2	1.1E+1	9.0E-2	7.0E+0	1.7E+2	6.0E-2	7.0E+1	2.2E+0	6.0E-2	5.9E-1	8.0E-3	9.0E-3	5.2E-2	3.0E-2	2.6E+2

Table E-6
Radionuclide Concentration in Concrete Cores[a]
July 1984 (pCi/cm^2)[b]

Sample[f]	Depth (cm)	^{54}Mn	^{55}Fe	^{60}Co	^{125}Sb	^{134}Cs	^{137}Cs
1	0-1	1.3E-2	c	1.8E+0	< 9E-2	< 3E-2	5.2E+0
	1-2	< 4E-3	c	1.2E-1	< 6E-2	< 3E-2	2.9E-1
2	0-1	5.4E-1	c	1.7E+2	1.8E+0	4.0E+1	2.1E+3
	1-2	< 2E-3	c	3.4E-1	4.7E-2	< 1E-2	1.8E-1
3	0-1	1.2E-2	c	5.0E+0	< 4E-2	4.4E-2	2.4E+0
	1-2	< 4E-3	c	< 5E-2	< 6E-2	< 3E-2	< 6E-2
4	0-1	1.5E-1	2.8E+6[d]	7.7E+1	< 1E+0	4.0E+1	3.6E+3
	1-2	1.2E-2	c	6.8E+0	< 5E-1	1.0E+0	6.3E+1
5	0-1	< 1E-2	c	2.4E+0	< 3E+0	9.1E+0	4.5E+3
	1-2	< 3E-2	c	7.1E-2	< 9E-1	< 2E-2	4.7E-1
6	0-1	1.5E-1	c	7.9E+1	< 3E+0	2.2E+2	1.0E+4
	1-2	<4E-3	c	8.1E-1	< 3E-1	1.9E+0	1.5E+2
7	0-1	<2E-2	c	1.9E+1	< 2E+0	1.1E+2	6.3E+3
	1-2	<6E-3	c	2.0E+0	< 1E+0	4.4E+0	5.5E+2
8	0-1	2.8E+0	c	1.5E+3	6.5E+1	1.2E+2	4.0E+3
	1-2	< 4E-3	c	6.3E-1	< 9E-2	< 3E-2	2.3E-1
9	0-1	6.3E+0	c	1.1E+4	9.8E+1	1.0E+1	1.1E+3
	1-2	<7E-3	c	1.3E+0	<9E-2	<7E-2	2.2E-1
10	0-1	1.5E+1	c	1.4E+4	1.9E+2	1.0E+1	7.4E+2
	1-2	< 5E-3	c	1.1E+0	<9E-2	<3E-2	1.9E-1
11	0-1	< 5E-3	c	1.2E+1	< 1E-1	1.1E-1	1.5E+1
	1-2	< 7E-3	c	1.8E-1	< 9E-2	< 3E-2	9.3E-2
12	0-1	<8E-3	c	8.2E+0	<1E-1	4.0E-1	2.1E+1
	1-2	<8E-3	c	6.1E-1	<9E-2	<3E-2	1.4E-1
13	0-1	1.2E-2	c	1.1E+1	<3E-1	4.4E+0	1.9E+2
	1-2	<4E-3	c	1.1E-1	<9E-2	<3E-2	1.8E-1
14	Whole	<6E-2	c	2.4E+1	<4E+0	4.2E+2	2.2E+4
15	0-1	<9E-2	c	2.6E+1	<5E-1	2.7E-1	2.1E+1
	1-2	<4E-3	c	1.7E-1	<9E-2	<7E-2	1.1E-1

Table E-6 (Continued)
Radionuclide Concentration in Concrete Cores[a]
July 1984 (pCi/cm^2)[b]

Sample[f]	Depth (cm)	^{54}Mn	^{55}Fe	^{60}Co	^{125}Sb	^{134}Cs	^{137}Cs
16[e]	0-1	3.0E+1	2.1E+8[d]	1.0E+4	1.7E+1	9.1E+1	4.0E+3
	1-2	4.0E+0	c	1.8E+3	4.2E+0	3.4E+1	1.4E+3
17	0-1	2.2E+0	c	7.1E+2	1.4E+0	9.5E+0	4.8E+2
	1-2	<5E-3	c	1.8E+0	<9E-2	<7E-2	8.9E-1
18	0-1	<2E-2	c	8.1E+1	<9E-2	<3E-2	4.1E+1
	1-2	<8E-3	c	2.0E-1	<SE-2	<4E-2	<6E-2
19	0-1	2.6E-2	c	1.2E+1	<5E-1	9.8E+0	5.0E+2
	1-2	<4E-3	c	1.0E+0	<5E-1	<7E-2	2.0E+0
20	0-1	<4E-2	c	6.7E+0	< 1E-1	5.8E-1	3.5E+1
	1-2	<4E-3	c	1.6E-1	<5E-2	<3E-2	<7E+0
21	Whole	7.9E-2	c	1.9E+1	<2E-1	1.1E+0	6.8E+1
22	0-1	<6E-2	c	2.8E+1	<5E-1	4.0E+0	2.6E+2
	1-2	<9E-3	c	1.1E-1	<9E-2	<3E-2	3.3E-1
23	0-1	7.0E-1	c	1.6E+2	<3E-1	4.4E-1	3.9E+1
	1-2	<4E-3	c	2.2E-1	<9E-2	<7E-2	1.7E-1
24	0-1	c	c	c	c	c	c
	1-2	<4E-3	c	1.6E-1	<9E-2	<3E-2	<9E-2
25	0-1	2.8E+0	c	9.9E+2	1.1E+0	2.2E+0	1.2E+2
	0-2	<6E-3	c	6.6E-2	<4E-2	<7E-3	1.7E-1
26	0-1	<4E-3	c	1.9E-1	<9E-2	<4E-2	1.5E+0
	1-2	<4E-3	c	4.7E-2	<5E-2	<3E-2	<6E-2
27	0-1	<4E-2	c	5.3E+0	<3E-1	<3E-2	8.0E+0
	1-2	<4E-3	c	9.4E-1	<9E-2	<3E-2	<5E-2

a Excerpted from <u>Residual Radionuclide Distribution and Inventory at the Humboldt Bay Nuclear Power Plant</u>. PNL-4628, May 1983, decay corrected to 7-1-84.

b To convert to pCi/g multiply by 0.406.

Table E-6 (Continued)
Radionuclide Concentration in Concrete Cores[a]
July 1984 (pCi/cm^2)[b]

c Not analyzed.

d pCi/kg.

e Core taken over crack in concrete floor.

f Sample Locations

1. Sand Blast Pad
2. Hot Shop Floor Drain
3. Hot Shop Background
4. Radwaste Tank Area
5. Radwaste Tank Area
6. Radwaste Building
7. Radwaste Building
8. Reactor Building-66 ft
9. Reactor Building-66 ft
10. Reactor Building-66 ft
11. Reactor Building-34 ft
12. Reactor Building-24 ft
13. Concrete Roof Over Conc. Waste Tanks
14. Asphalt
15. Condensate Demin Room
16. Condensate Demin Room
17. Condensate Demin Room
18. Condensate Pump Room
19. Turbine Building
20. Turbine Building
21. Turbine Building
22. In Yard Near Former Stack
23. Condensate Storage Tank
24. Air Ejector Room
25. Reactor Building Refuel Level
26. Intake Pumping Platform
27. Access Control Area

Table E-7
Radiation Survey-Refueling Building[a]

Location		Dose Rate[b] mR/h Gamma[d]	Beta	Contamination Levels (μci/100cm^2) Contact[c] Alpha	Beta-Gamma	Smearable Alpha	Beta-Gamma[e]
+ 12 ft Elevation	floor	10	<1	f	3.6E-2	3.9E-6	1.1E-3
	wall			f	9.8E-3	2.2E-6	3.3E-4
Access Shaft -2 ft El	floor	7[g]	h	f	1.6E-2	7.1E-6	1.5E-3
	wall			f	2.1E-3	f	2.7E-5
-14 ft El	floor	28	0	f	4.2E-3	4.7E-6	2.3E-3
	wall			f	2.4E-3	2.3E-6	7.6E-4
-24 ft El	floor	1[g]	h	f	3.1E-3	1.4E-5	2.4E-3
	wall			f	1.0E-3	f	f
-34 ft El	floor	1[g]	h	f	2.1E-3	1.2E-5	3.0E-3
	wall			f	f	f	f
-44 ft El	floor	7[g]	1.5	f	8.3E-2	4.5E-6	1.3E-3
	wall			f	1.0E-2	f	2.7E-5
-54 ft El	floor	18	1.1	f	1.2E-1	4.5E-6	1.2E-3
	wall			f	2.1E-2	f	f
-66 ft El	floor	12	0	f	1.4E-1	2.3E-6	6.1E-4
	wall			f	6.4E-2	f	f
Cleanup HX Room -2 ft El	floor	65	0	f	1.0E-1	2.1E-5	9.4E-3
	wall			f	4.2E-2	f	1.9E-5
Cleanup Demin Room -2 ft El	floor	6	1.5	f	2.1E-1	1.0E-4	4.2E-2
	wall			f	2.1E-3	2.0E-6	3.5E-4
Shutdown HX Room -14 ft El	floor	55	1.1	f	f	3.7E-6	2.8E-3
	wall			f	2.1E-2	2.8E-7	2.0E-5
West Wing -66 ft El	floor	110	7.5	f	f	1.2E-5	2.7E-3
	wall			f	9.6E- 2	5.6E-7	f
Under Reactor -66 ft El	floor	23	21	1.7E-3	2.0E+0	9.0E-4	3.3E-1
	wall			f	3.2E-2	6.5E-5	4.4E-3

Table E-7 (Continued)
Radiation Survey-Refueling Building[a]

		Dose Rate[b]		Contamination Levels (μci/100cm^2)			
		mR/h		Contact[c]		Smearable	
Location		Gamma[d]	Beta	Alpha	Beta-Gamma	Alpha	Beta-Gamma[e]
New Fuel Vault +0 ft El	floor	5	47	3.4E-4	2.3E+0	1.9E-5	5.4E-3
	wall			f	f	1.1E-6	6.3E-4
TBDT Area -14 ft El	floor	23	35	f	1.6E-1	4.2E-6	9.6E-4
	wall			f	3.4E+0	1.1E-6	9.1E-3

a Average values of PG&E Survey conducted May 1984 unless otherwise specified.

b Ion Chamber.

c Minimum Sensitivity
Alpha: Approximately 1E-4 μCi/100cm^2
Beta: Approximately 5E-3 μCi/100cm^2 for Cutie Pie
Approximately 2E-6 μCi/100cm^2 for HP-210

d Based on ^{137}Cs.

e Based on ^{90}Sr (10%), ^{60}Co (45%) and ^{137}CS (45%).

f Not detected.

g Previous survey.

h Data not recorded.

Table E-8
Radiation Survey-Power Building[a]

Location		Dose Rate[b] mR/h Gamma[d]	Dose Rate[b] mR/h Beta	Contamination Levels (µci/100cm²) Contact[c] Alpha	Contact[c] Beta-Gamma	Smearable Alpha	Smearable Beta-Gamma[e]
Cond. Demin.	floor	11	0	f	3.2E-2	8.5E-6	1.4E-3
Cubicle	wall			f	3.2E-2	f	9.7E-5
Cond. Demin.	floor	14	1.5	2.6E-4	3.5E-2	1.1E-5	2.7E-3
Regen. Room	wall			1.0E-3	7.1E-2	1.1E-5	1.5E-3
Cond. Demin.	floor	14[g]	h	f	3.5E-3	1.4E-6	1.5E-4
Op. Area	wall			f	8.8E-3	f	6.1E-5
Cond. Pump	floor	13[g]	h	f	f	2.0E-6	5.0E-4
Room	wall			f	f	f	2.8E-5
Air Ejector	floor	55	56	f	5.6E+0	1.7E-6	7.8E-2
Room	wall			f	f	h	1.5E-3
Condenser	floor	19	< 1	f	6.0E-3	5.7E-7	5.7E-4
Area	wall			f	f	h	h
Pipe Tunnel	floor	15	1.5	f	4.7E-3	1.lE-6	2.9E-4
	wall			f	f	1.4E-7	2.1E-5
Feed Pump	floor	< 1[g]	h	f	5.2E-4	f	8.4E-5
Room	wall			h	h	h	h
Seal Oil	floor	0.005 [g]	h	f	f	f	2.1E-5
Room	wall			h	h	h	h
Turbine Enc	floor	<1.[g]	h	f	3.1E-3	8.5E-7	1.2E-4
+27 ft El	wall			f	4.2E-3	2.8E-7	f
Turbine Washdown Area +27 ft El	floor	< 1[g]	h	f	1.0E-3	1.7E-6	6.1E-5
Hot Lab	floor	< 1[g]	h	f	1.2E-2	f	7.3E-5
Laundry/ Demin Area +27 ft El	floor	< 1[g]	h	f	2.6E-3	4.3E-7	7.7E-5
Laundry/ Hot Lab +34 ft El	floor	h	h	f	1.0E-3	f	2.0E-4

Table E-8 (Continued)
Radiation Survey-Power Building[a]

Notes:

a Average values of PG&E Survey conducted May 1984 unless otherwise specified.

b Ion Chamber.

c Minimum Sensitivity
Alpha: Approximately 1E-4 $\mu Ci/100cm^2$
Beta: Approximately 5E-3 $\mu Ci/100cm^2$ for Cutie Pie
Approximately 2E-3 $\mu Ci/100cm^2$ for HP-210

d Based on ^{137}Cs.

e Based on ^{90}Sr (10%), ^{60}Co (45%) and ^{137}Cs (45%).

f Not detected.

g Previous survey.

h Data not recorded.

Table E-9
Radiation Survey- Yard Structures

Location		Dose Rate[b] mR/h Gamma[d]	Dose Rate[b] mR/h Beta	Contamination Levels (μci/100cm^2) Contact[c] Alpha	Contact[c] Beta-Gamma	Smearable Alpha	Smearable Beta-Gamma
Hot Shop	floor	< 1[g]	h	f	1.3E-2	4.5E-5	6.0E-3
	wall			f	f	f	f
Calibration facility	floor	< 1[g]	h	f	2.5E-3	f	f
Former Stack							
-0 ft El	floor			f	4.3E-2	f	1.1E-4
+12 ft El	floor	1.8[g]	h	f	9.3E-3	5.6E-7	1.9E-5
+26 ft El	floor			f	f	5.6E-7	1.9E-4
Radwaste Treatment	floor	15	6.8	f	4.9E-1	1.0E-6	4.2E-3
	wall			f	6.5E-3	7.0E-7	1.9E-4
Low Level Waste Building	floor	190	7.5	f	3.8E-1	h	h
Radwaste Handling Building	floor	5[g]	h	f	f	2.8E-7	1.5E-4

a Average values of PG&E Survey conducted May 1984 unless otherwise specified.

b Ion Chamber.

c Minimum Sensitivity
Alpha: Approximately 1E-4 μCi/100Cm^2
Beta: Approximately 5E-3 μCi/100cm^2 for Cutie Pie
Approximately 2E-3 μCi/100cm^2 for HP-210

d Based on ^{137}Cs

e Based on ^{90}Sr (10%), ^{60}Co (45%) and ^{137}Cs (45%)

f Not detected

g Previous survey

h Data not recorded

Table E-10
Cross-connections Radionuclide Analysis[a]

	^{60}Co	% MPC[b]	^{134}Cs	% MPC[b]	^{137}Cs	% MPC[b]
IN-PLANT LIQUIDS (μCi/ml) 1982 - 1983						
Boiler No. 1	c		c		c	
Boiler No. 2	c		c		c	
Turbine Lube Oil	c		c		c	

	^{60}Co	% MPC[d]	^{134}Cs	% MPC[b]	^{137}Cs	% MPC[b]
INSTRUMENT AND SERVICE AIR (μCi/ml) 1982 - 1983						
Cond. Demin. Room	c		c		c	
+12 ft Elevation	c		c		c	
Instrument Vault	c		c		c	

	^{60}Co	% MPC[b,e]	^{134}Cs	% MPC[b,e]	^{137}Cs	% MPC[b,e]
YARD DRAINS (μCi/ml) 1981 - 1983						
Unrestricted No. Loop	2.8 E-8	9.3 E-2	c		7.4 E-8	3.7 E-1

Table E-10 (Continued)

Cross-connections Radionuclide Analysis

Unrestricted So. Loop	2.2 E-7	7.3 E-1	3.3 E-9	3.7 E-2	1.1 E-7	5.5 E-1
Oily Water[f] Drain System	2.9 E-5	g	6.0 E-8	g	3.6 E-6	g
Restricted No. Loop	1.1 E-6	1.1 E-1	4.1 E-6	1.4 E0	1.5 E-5	3.8 E0
Restricted So. Loop	1.6E-6	1.6E-1	c		8.4 E-7	2.1 E-1

a Average above background.
b 10 CFR 20 Appendix B, Table II, Column 2 (most restrictive value).
c Background.
d 10 CFR 20 Appendix B, Table II, Column 1 (most restrictive value).
e 10 CFR 20 Appendix B, Table I, Column 2, (most restrictive value).
f Sludge sample from oily water separator (µCi/g) 1982.
g Not Applicable

Appendix F

CERTIFIED FUEL HANDLER TRAINING AND CERTIFICATION PROGRAM

F.1 INTRODUCTION

This program describes the training and certification for some Supervisors and Operators associated with the maintenance at Humboldt Bay Power Plant Unit 3 in the SAFSTOR mode consistent with its possession-only license.

F.2 APPLICABILITY

The Unit 3 Technical Specifications require that certain operations associated with the maintenance and handling of reactor spent fuel be directly supervised by a qualified individual. The following members of the plant staff (as a minimum) shall be certified in accordance with this program:

- Engineering Manager
- Supervisor of Operations
- Shift Foremen
- Selected operators who shall be performing duties requiring certified operators
- Training Coordinator

F.3 INITIAL CERTIFICATION

A training program shall be administered to certify members of the Plant staff as certified fuel handlers. The training program shall include the following:

- Reactor Theory (as applicable to the storage and handling of spent reactor fuel)
- Spent Fuel Handling and Storage Equipment - Design and Operating Characteristics
- Monitoring and Control Systems
- Radiation Protection
- Normal and Emergency Procedures
- Administrative Controls applicable during the SAFSTOR period

Reactor Theory training will include characteristics of the stored spent fuel, subcritical multiplication, factors affecting reactivity and criticality, and the basis for fuel handling restrictions and procedures.

The design and operating characteristics will include training in the functions and use of fuel handling tools, cranes, the spent fuel storage pool, and pool service systems and equipment. Prior to

shipments of spent fuel this training will include shipping casks, cask handling equipment, and procedures.

Monitoring and Control Systems will include training on the spent fuel pool level monitoring systems, criticality monitors, and Unit 3 Area Radiation Monitors.

Radiation protection training will include theory of radioactive emissions, control of radiation exposure, use of radiation detection and monitoring equipment, protective clothing and respiratory protection, and contamination control procedures. Training will emphasize the principles and practices associated with maintaining exposures as low as reasonably achievable (ALARA).

Normal and Emergency Procedure Training will include the Emergency Plan and any operations and emergency procedures associated with the operation of Unit 3 systems and equipment during SAFSTOR. This area shall also include training in the handling and processing of radioactive wastes.

Administrative Control Training will include the Unit 3 Technical Specifications, Security Plan, Quality Assurance Plan and plant administrative procedures associated with the operation, surveillance, and maintenance of Unit 3.

Examinations shall be conducted to evaluate an individual's overall depth of knowledge and understanding of the plant and of the possession-only license requirements.

F.4 PROFICIENCY TRAINING AND TESTING

The frequency of proficiency training shall be such that all six of the topics discussed in Section F.3, above, are covered in a 2-year period. If, in the course of the annual training, an infrequently performed activity or procedure is planned for which training should be conducted, it may be moved from its regularly scheduled time to accommodate the present circumstances. This training will be in addition to any previously scheduled topic. Additionally, any significant areas of weakness noted on the annual exams shall be given priority in the training schedule.

Annual examinations shall be used to demonstrate the proficiency of certified personnel. Examinations will be similar to but not as comprehensive as the initial certification examinations. Minimum passing grade for proficiency examinations shall be 70 percent in each section and 80 percent overall. Oral examinations shall be on a pass/fail basis.

F.5 CERTIFICATION

Upon successful completion of the initial certification training program, the Plant Manager or his delegate shall certify the individual as a Certified Fuel Handler. Normally an employee will complete the initial certification within one year after entering the program. After initial certification, personnel will be recertified every 2 years based on the successful completion of the Proficiency Training and Testing Program.

F.6 PHYSICAL REQUIREMENTS

As a prerequisite to acceptance into the training program and for recertification, a candidate must successfully pass a medical examination designed to ensure that the candidate is in generally good health and is otherwise physically qualified to safely perform the assigned work. Minor correctable health deficiencies, such as eyesight or hearing, will not per se prevent certification.

The medical examination will meet or exceed the requirements of ANSI Standard N546-1976, "American National Standard - Medical Certification and Monitoring of Personnel Requiring Operator License for Nuclear Power Plants."

F.7 DOCUMENTATION

Initial Certification and Proficiency Training shall be documented and maintained for certified personnel for a minimum of 5 years. The records shall include the dates and periods of training, results of all quizzes and examinations, copies of written examinations, oral examination records, and information on results of physical examinations.

Appendix G

DESCRIPTION OF ABANDONED OR REMOVED SYSTEMS

During SAFSTOR decommissioning, systems no longer required were secured and isolated. In addition, systems that were required to support SAFSTOR decommissioning activities but which will not be required during SAFSTOR were secured upon the completion of those activities. The systems that are abandoned in place, or removed altogether, are described in this appendix.

Spent Fuel Storage Pool and Associated Systems

Fuel Pool Coolers. The fuel pool coolers are located adjacent to the fuel pool circulating water pumps in the refueling building. Their function was to remove decay heat added to the pool water by the spent fuel. Due to the age of the spent fuel on site, decay heat is low enough that the coolers are no longer required. The fuel pool coolers have been drained, flushed, and removed from service.

Waste Disposal Systems

Condenser Off-Gas System. The condenser off-gas system functioned to receive the non-condensable gases from the main condenser air-ejectors and to delay the release of the gases to permit decay of the short-lived radionuclides. The system consists of a buried holdup pipe and a HEPA filter that are contained in a below-grade vault near the former ventilation exhaust stack. This system has been abandoned and the off-gas filter removed. A modification to upgrade the condenser off-gas system was under construction when Unit 3 was shut down in 1976. The new system was in the condenser off-gas treatment vault located north of the former ventilation exhaust stack. The system was never used since Unit 3 did not return to operation. This equipment was removed from the vault in 2002. The sump pump in the vault will be maintained in an operational status.

Laundry Waste Hold Tank. The laundry waste hold tank is the current designation for the 685-gallon regenerated resin storage tank, which was part of the condensate demineralizer regeneration equipment. This tank was used to hold laundry waste tank volumes while awaiting sample analysis prior to discharge, but is no longer used for this purpose and has been isolated.

Radwaste Concentrator and Condenser. The radwaste concentrator was designed to concentrate 7,500 gallon per week. The concentrator consists of a vessel about 14 feet high and 24 inches in diameter with a 40 square foot, callandria-type evaporating section near the bottom. Steam from the Unit 1 or Unit 2 auxiliary steam system is fed to the callandria outside of the tubes. Evaporation takes place within the tubes. The concentrator is located in a shielded cubicle in the radwaste building.

Concentrator vapor goes to a condenser, which is cooled with water from an independent cooling loop, and the condensate goes to the drip receiver tank for collection for further

treatment or disposal. The concentrated radwaste is discharged to one of the two concentrated waste storage tanks.

An independent cooling water pump circulates water between the condenser and a second heat exchanger, which receives its cooling water from the Unit 2 bearing cooling water system. This independent cooling water loop provides a radiological barrier between the concentrator and the Unit 2 system supplying cooling water.

The radwaste concentrator and condenser are no longer needed and are no longer in service.

Concentrator Feed Pumps. A concentrator feed pump is located in the radwaste building, draws suction on the receiver or hold tanks, and discharges to the concentrator. As an alternate, the radwaste pump can discharge to the concentrator. The concentrator feed pumps are no longer needed and are no longer in service.

Concentrated Waste Tanks. Two 5,000-gallon storage tanks are located in a shielded vault in the radwaste building. These tanks received concentrated wastes from the concentrator before it was removed from service. These tanks have no inherent means for draining and must be pumped down through access ports in the top of the tank.

Concentrator Drip Receiver Tank and Pump. A concentrator drip receiver tank is provided to collect the condensed vapors from the concentrator. The concentrator drip receiver pump either recirculates water in the tank for sample mixing purposes, or it discharges to the treated waste pump discharge header for final disposition. The concentrator drip receiver tank and pump are no longer needed and are no longer in service.

Spent Fuel Storage Pool Filter. An additional cartridge-type filter was located in the radwaste building, which was dedicated to processing water from the spent fuel storage pool. The filter is similar in design but smaller than the radwaste filter. It was removed and shipped for disposal.

Service Systems

Reactor Shield Cooling System. The function of the shield cooling system was to prevent excessive thermal stress concentrations in the concrete biological shielding surrounding the reactor. Four independent cooling coils provided shield cooling. The coils are located within the inner surface of the biological shield wall and are fabricated from carbon steel pipe. This system has been drained and isolated.

Nuclear Steam Supply System

Reactor Vessel and Internals. An isometric view of the reactor vessel is shown in Appendix H, Figure 2-19. The reactor vessel is a vertical cylindrical shape, 40.5 feet in length with a 10 foot inside diameter. The vessel is high strength carbon steel alloy, ASTM A-302, Grade B, fire box quality, with an interior surface clad with 304 stainless steel deposited by the weld deposit overlay method. The vessel wall thickness (including cladding) ranges

between 4 and 5 inches. A bolted closure head is located on top of the vessel and provides access for refueling. Table G-1 lists miscellaneous reactor data.

The vessel contains and supports the reactor internal components, coolant, and fuel assemblies. The internal components are:

- Core support assembly
- Core spray ring
- Feedwater sparger
- Steam dryer
- Control rods
- Control rod guide tubes
- Lower core shroud
- Upper core shroud
- Upper guide

The steam dryers, which are a screen type design, occupy the cross-section of the reactor vessel closure head. They are inclined to promote drainage and are removed with the reactor head.

Following the unloading of all fuel assemblies from the reactor vessel, the irradiated incore fission chambers were removed and stored in the spent fuel storage pool. Control rods and all core internals were left in place. The reactor vessel bottom was vacuumed to remove any existing residue and the reactor vessel head was installed with the stud nuts hand-tight to facilitate their removal at the time of final decommissioning. The main steam line spool piece, safety and relief valves, and reactor associated piping were reinstalled. A blank was installed in the flange on the steam line end of the main steam spool piece and a gap was left in the flange at the reactor vessel to permit venting of the vessel during the vessel drain.

Following reinstallation of the upper drywell head and reactor shield plug and following closure of the lower drywell head, the reactor vessel was drained. A test was conducted with the water level in the vessel lowered to the bottom of the vessel to identify conditions, which will exist during dry layup. Table G-2 tabulates the results of this test. The reactor vessel will remain drained during the SAFSTOR period.

Control Rod Hydraulic System. The control rod hydraulic system provided the hydraulic motive power for control rod positioning. The system consists of control rod drive mechanisms (located exterior to and below the reactor vessel), two supply pumps, accumulators, and a scram dump tank, along with necessary controls, valves, filters, and interconnecting piping. This system is deactivated and no future use is anticipated.

To reduce radiation dose rates in areas requiring routine access during SAFSTOR, several components of this system have been removed. These components include hydraulic filters at the -14 foot level, the scram dump tank, the level pots on the scram dump tank, and

associated piping at the -66 foot level of the access shaft. The system has been flushed and is left in a drained condition.

Liquid Poison System. The liquid poison system served as an emergency backup to maintain the reactor subcritical if the control rod drive system failed to insert a sufficient number of rods into the core.

The system includes a poison tank that contained approximately 385 gallons of sodium pentaborate solution. The tank is constructed of carbon steel and has a corrosion-resistant interior lining of a baked phenolic resin. The tank is located beside the north wall of the refueling building at the +12 foot elevation.

This liquid poison system has been drained and the nitrogen bottles have been removed and decontaminated. Caps have been installed on all lines entering the refueling building from this system.

Reactor Cleanup System. The reactor cleanup system was provided to remove corrosion products from the reactor vessel water. The system consists of a cleanup pump, two regenerative heat exchangers, a nonregenerative heat exchanger, a demineralizer, and a resin storage tank.

This equipment has been deactivated and isolated. Resins in the cleanup system demineralizer have been sluiced to the radwaste treatment system for disposal. The cleanup pump and system piping at the -66 feet level were a source of radiation to maintenance workers in the vicinity and therefore have been removed. The cleanup heat exchanger and demineralizer room will not be available for routine access during SAFSTOR. Routine access to the room will be prevented by a locked barrier.

Reactor Shutdown Cooling System. The reactor shutdown cooling system was provided to remove decay heat following a reactor shutdown when reactor pressure was less than 120 psig and during refueling operations. Cooling water to the system heat exchangers was supplied by the closed cooling water system. The reactor shutdown cooling system consists of two heat exchangers, two pumps and necessary connecting piping, valves, instrumentation, and controls.

The equipment in this system is located in a shielded room at the -14 foot elevation on the west side of the caisson. This system has been drained and the water processed as radioactive liquid waste. Several sections of pipe in the access shaft have been flushed and/or removed due to ALARA considerations. The system has been isolated mechanically (by appropriate valving) and electrically. Routine access to the shutdown room will not be required during SAFSTOR. Routine access to the room will be prevented by a locked barrier.

Emergency Condenser System. An emergency cooling system was provided to dissipate reactor decay heat following a reactor scram, if normal heat sinks were unavailable. This system was designed to maintain the reactor in a safe condition when all normal plant auxiliary power had been lost. The system could be placed in operation either automatically following a reactor scram or manually by remote opening of one motor-operated valve. The arrangement of the system was such that steam generated in the reactor would be

condensed in the emergency condenser and the condensate returned by gravity to the reactor. The system consists of an emergency condenser, an emergency make-up pump, and associated piping and valves.

The emergency condenser is located on the north side of the refueling building with a centerline at elevation +28 feet over the spent fuel storage pool. It is supported in this position by a steel foundation on each end mounted to the east and west sides of the pool. The emergency makeup pump was located in the yard between the condensate storage tank and the refueling building, but has been removed.

Supply and return lines from the reactor have been cut and capped, and the heat exchanger shell and tubes have been drained.

Core Spray and Suppression Pool Cooling. The core spray system was available to supply cooling water to the reactor core in the event of a loss of coolant accident. The system could take suction from the suppression pool and discharge the water to a core spray ring in the reactor vessel where it would be distributed over the core. A heat exchanger on the discharge side of the pumps was available to cool the suppression chamber water. During normal operation, the heat exchanger was used to maintain suppression chamber water temperature within technical specification limits.

The core spray and suppression pool cooling system consisted of two pumps located at the -66 foot level, a heat exchanger located at the -2 foot level, and necessary piping, valves, instruments and controls. The system, including the suppression chamber itself, has been drained and isolated. The two core spray pumps and associated suction and discharge piping at the -66 foot level have been removed. No further use of the system is planned. Core spray system piping sections in the access shaft at the -14 and -66 foot levels were identified as a source of high radiation. Since routine access through these areas will be required for decommissioning activities and during SAFSTOR, this piping has either been flushed or sections have been removed and the ends were capped to contain internal contamination. A liquid level detection system has been added to the suppression chamber.

Turbine Plant Systems

Turbine System. The turbine-generator unit is a tandem, compound, double flow, condensing turbine, direct connected to a 13,800 volt, 3-phase, 60-cycle, hydrogen-cooled synchronous generator. The turbine is a non-reheat, condensing machine with three extraction points for feedwater heating and designed for use with a boiling water reactor. It consists of a single flow high-pressure section and a double flow, low-pressure section with a crossover pipe connecting the two sections. The two units are mounted in tandem. A valved, turbine bypass line was provided which could dump steam straight to the condenser in lieu of passing first through the turbine. The turbine is in a layup state and a nitrogen blanket was used to fill the turbine internals. Dry air or nitrogen is purged through the generator.

Main Condenser. The condenser is a single-pass, horizontally divided waterbox, deaerating-type unit with an effective surface of 30,700 square feet and generally of conventional construction. The condenser is connected to the turbine by an elbow-shaped connecting piece. Standard construction materials were used, including fabricated carbon steel shell and tube support plates and unlined close-grained cast iron waterboxes. Tubes are 7/8-inch outside diameter 18 BWG aluminum brass. Tube sheets are silicon bronze.

The condenser is located alongside the turbine pedestal with tubes parallel to the turbine centerline. The low-pressure feedwater heater is installed within the condenser connecting piece.

A 6,500-gallon oversized storage-type hotwell was provided to allow decay of short-lived radioactivity. The hotwell is divided by a partition plate parallel to the tubes to facilitate location of tube leaks.

The condenser contained a significant level of internal surface contamination. Internal decontamination of the condenser has been performed to reduce these levels. In addition, sections of the 10-inch condensate piping between the condenser and the LP heater, which contained significant internal contamination, have undergone decontamination to reduce radiation levels. The condenser and condenser hotwell have been flushed, drained, and isolated from the rest of the condensate system and other connected systems. A nitrogen blanket has been provided in the condenser to prevent internal deterioration.

Condensate System. The condensate system is composed of two condensate pumps, a contaminated drain tank and pump, two low-pressure feedwater heaters, three half-capacity condensate demineralizers, and necessary piping, valves, instruments, and controls. The system has been drained and flushed. As part of a program to reduce radiation exposures for ALARA considerations, portions of the condensate system have been decontaminated. The system is presently isolated.

Resins in the condensate demineralizers have been transferred to the radwaste system for processing and disposal. One of the condensate demineralizers was converted to a spent fuel storage pool demineralizer for the SAFSTOR period (see Condensate Storage Tank). The remaining demineralizers and their associated equipment will remain drained and isolated. Access to the condensate demineralizer room will be restricted during SAFSTOR by a locked barrier.

Condenser Air Removal System. The condenser air removal system is comprised of a condenser vacuum pump, two sets of air-ejectors, and their associated condensers. This system has been drained and isolated. Due to ALARA considerations, selected sections of piping have been removed to reduce radiation exposures in the area. The resulting open piping has been sealed.

Condensate Demineralizer Resin Regeneration System. A system for regenerating condensate demineralizer resins was originally installed and used during the early years of Unit 3 operation. Use of the system was discontinued and the equipment was abandoned in place. It consists of three tanks and associated piping, valves, and controls located in a shielded area of the condensate demineralizer room. One of the tanks was previously

converted for use as a laundry waste hold tank, but is no longer used for this purpose and has been isolated.

The condensate demineralizer resin regeneration system has been flushed, drained, and isolated (with the exception of piping necessary for resin addition and removal associated with the demineralizer used for the spent fuel storage pool).

Gland Seal System. The gland seal system consists of a gland seal condenser and two gland seal exhausters. This equipment is located on the ground floor of the station building in the air-ejector room. Its function was to remove the mixture of air and steam from the turbine seals. Since no future use of this system is planned, the exhausters have been electrically disconnected, the system has been isolated and hot spots have been removed.

Feedwater System. The feedwater system consists of two feedwater pumps (or reactor feed pumps) and necessary piping, valves, instruments, and controls. The feedwater pumps are located in the feed pump room, which is on the ground floor (elevation +12 feet) of the power building.

The system has been drained and pumps electrically isolated. Due to ALARA considerations, sections of feedwater piping have been internally decontaminated and/or removed to reduce radiation levels in areas requiring periodic access. System openings have been sealed and the system isolated from other connected systems.

Closed Cooling Water System. The closed cooling water system consists of a cooling water return tank, two cooling water pumps, two cooling water heat exchangers, and interconnecting piping and valves that supply cooling water and seal water to a variety of coolers and pumps. Water contained in this system was treated with potassium chromate to inhibit corrosion.

The return tank is located below grade and immediately outside the east side of the station building. The pumps are located above grade on the outside of the building above the tank and on top of the vault. The heat exchangers are located outside the building along the southeast wall of the power building. The chromated water contents of the system have been drained to the suppression chamber and processed. The system has been drained, flushed, and isolated. No future use is planned.

Main Circulating Water System (or Salt Water System). The main circulating water system provided salt water cooling to the main condenser, the cooling water heat exchangers, and the suppression pool cooler.

The system contains an intake structure that housed a trash rake, two parallel traveling screens, two screen wash pumps, three sluice gates (including operators), two circulating water pumps, and associated piping, valves, and instrumentation. The system also includes intake piping to the main condenser, discharge piping and isolation valves from the main condenser, piping, valves, and instrumentation to the above noted heat exchangers, and a discharge structure located at the discharge canal. The traveling screens, screen wash pumps, and circulating water pumps have been removed to prevent deterioration. Openings into system internals have been sealed. The tube side of the main condenser was drained. The system has been isolated from connected systems.

Table G-1
Reactor Data

CATEGORY	SPECIFICATION
Reactor Type	Single cycle, natural circulation boiling water
Coolant/Moderator	Light water
License Number	DPR-7
Owner	Pacific Gas and Electric Company
Architect/Engineer	Bechtel
Supplier	General Electric Company
Weights:	
Reactor Vessel Assembled, Dry	288,146 lb
Closure Head (Including Steam Dryer and Cooling Spray Sparger)	28,068 lb
Vessel Shell (Including Feedwater Sparger)	231,372 lb
Feedwater Sparger	1,242 lb
Steam Dryer	1,440 lb
Studs, Nuts, and Washers	3,640 lb

Table G-2
Survey Results - Reactor Vessel Drain Test

SURVEY LOCATIONS		Before Vessel was drained mR/hr	After Vessel was drained mR/hr
(1)	Refueling Building		
	On top of roof, directly above reactor	< 1	< 1
	+ 12 ft elevation, at center of shield plug	0.4	1
	+12 ft elevation, around edges of shield plug	0.5 to 1.3	1
(2)	Access Shaft		
	Wall (4 ft above floor) behind the Suppression Chamber Cooler at -2 ft elevation	1.1	2
	Wall (4 ft above floor), south of the manlift at -14 ft elevation	0.5	1
	Southwest wall (4 ft above-floor) of void space at -14 ft elevation (south of Hydraulic System filters)	10	10
	Wall (4 ft above floor), south of the manlift at -24 ft elevation	0.5	1
	Wall (4 ft above floor), south of manlift at -34 ft elevation	0.8	1.5
	Center of Suppression Chamber manway (west side) at the -34 ft elevation	0.7	1
	Center of Suppression Chamber manway (east side) at the -34 ft elevation	2.0	1

Table G-2(Continued)
Survey Results - Reactor Vessel Drain Test

	SURVEY LOCATIONS	Before Vessel was drained mR/hr	After Vessel was drained mR/hr
(2)	Access Shaft (Continued)		
	Wall (4 ft above floor), south of manlift at -44 ft elevation	6	5
	Wall (4 ft above floor), south of manlift at -54 ft elevation	7	7
	Center of lower drywell head at - 66 ft elevation	28	21
	18 in. above floor at -66 ft elevation directly under center of lower drywell head	23	27
	6 ft above floor at -66 ft elevation directly under center of lower drywell head	26	26
	Caisson Sump at -66 ft elevation	28	25
	Drywell instrument penetration at -66 ft elevation (south side)	11	13
	Drywell instrument penetration at -66 ft elevation (north side)	28	25
	18 in. above floor at -66 ft elevation directly under edge of drywell (north wall)	50	60
	6 ft above floor at -66 ft elevation directly under edge of drywell (north wall)	60	65
(3)	Pipe Tunnel (Valve Gallery)		
	Instrument penetration at the +2 ft elevation	40	33
	Feedwater line drywell penetration	8	10

Table G-2 (Continued)
Survey Results - Reactor Vessel Drain Test

SURVEY LOCATIONS	Before Vessel was drained mR/hr	After Vessel was drained mR/hr
• Main steam line drywell penetration	16	15
• Wall (6 ft above floor at -14 ft elevation) on the south side of drywell	10	8
• Wall (2 ft above floor at -14 ft elevation) on the south side of drywell	7	6.5
• Wall (6 ft above floor at -14 ft elevation) on the east side of drywell	18	18
• Wall (2 ft above floor at -14 ft elevation) on the east side of drywell	8	7

APPENDIX H

FIGURES

The figures contained in this appendix are referred to in various sections and appendices of the DSAR. The figures were developed either at the time Unit 3 entered SAFSTOR or during SAFSTOR, but none have been updated to reflect current plant conditions. The figures are provided for general information purposes only.

Plant drawings reflecting current plant conditions are maintained by the Engineering Department. These figures should be used to obtain specific information regarding current plant conditions.

LIST OF FIGURES

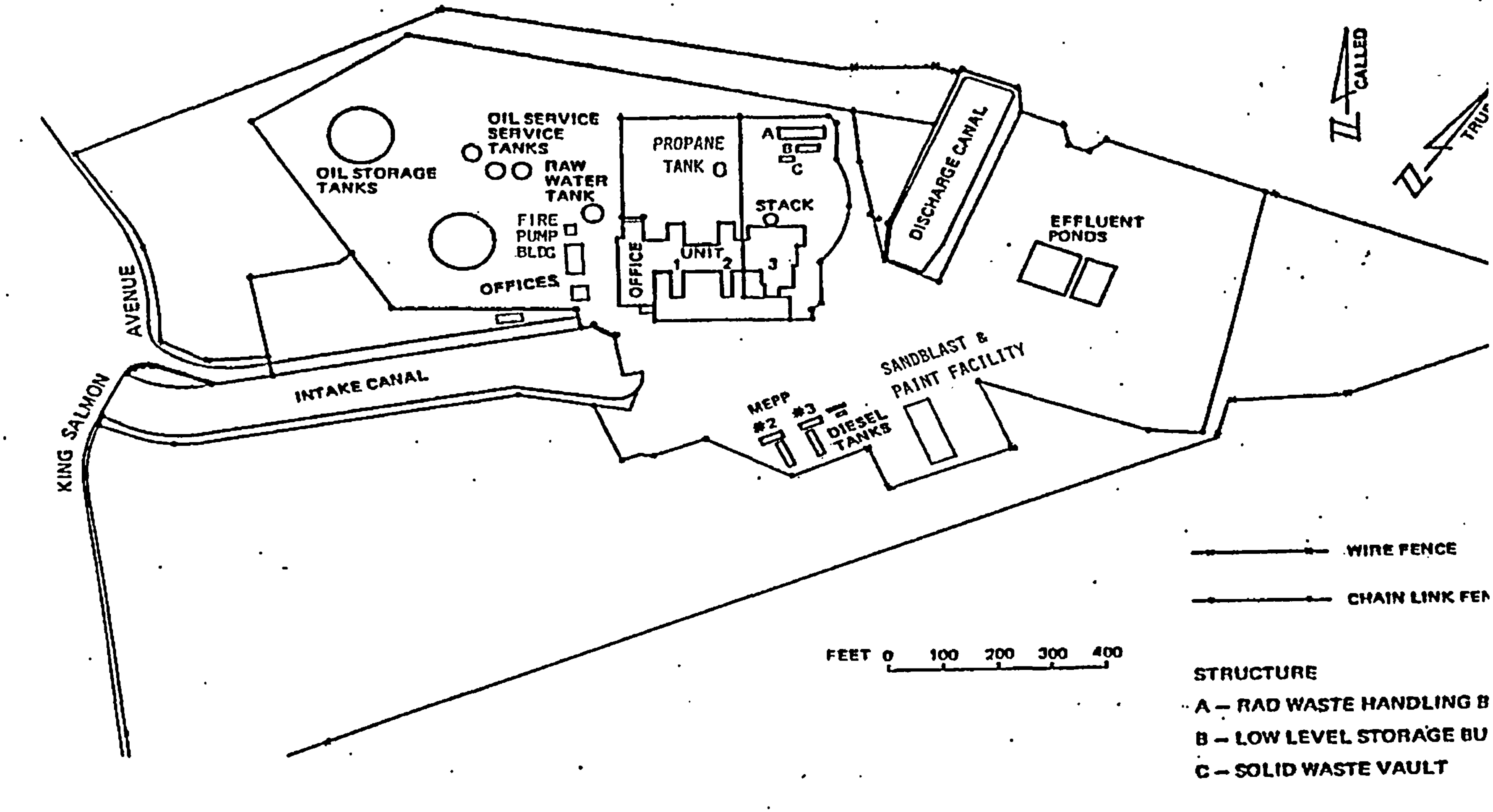


FIGURE 2-1
HUMBOLDT BAY POWER PLANT SITE PLAN

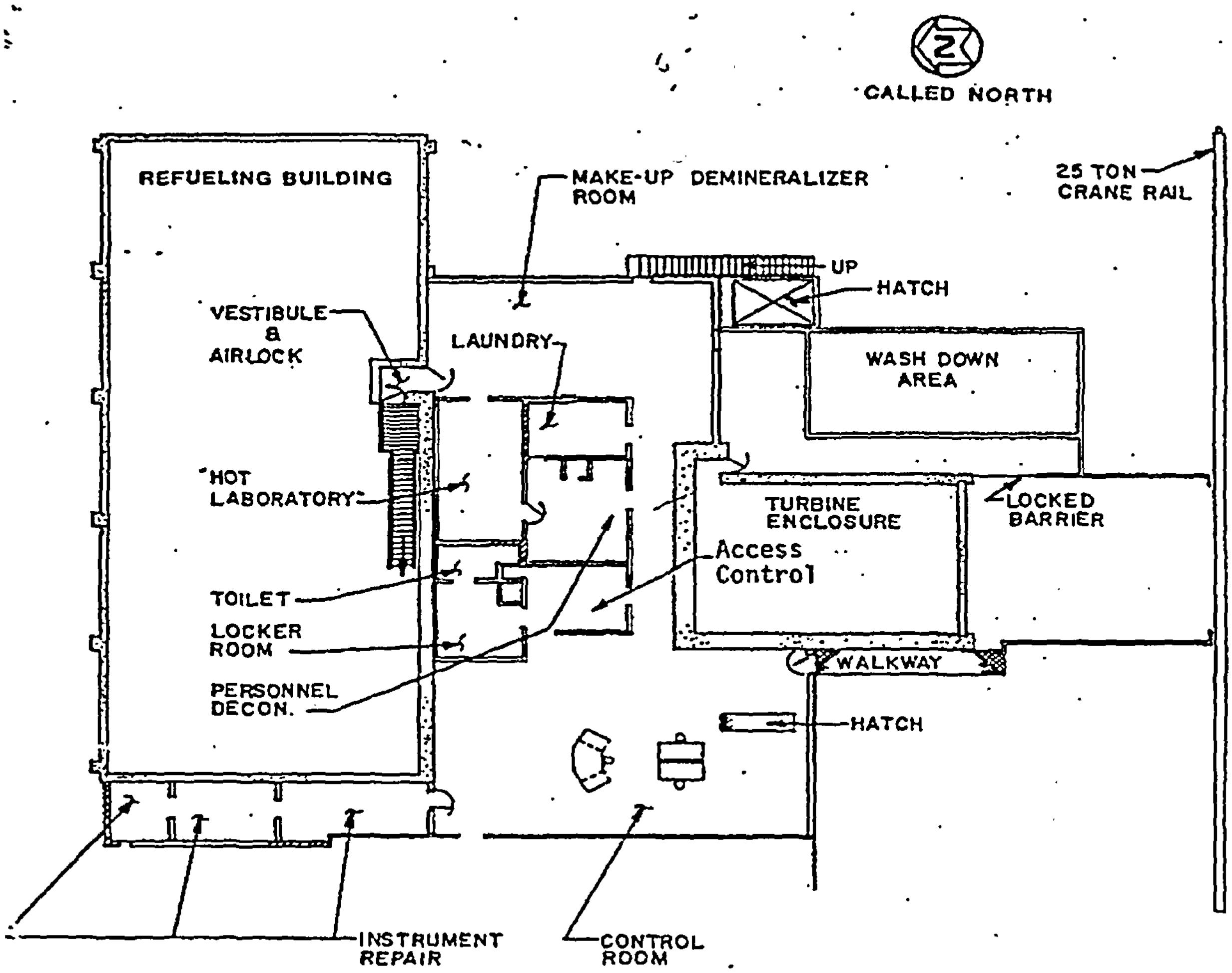


FIGURE 2-2
HUMBOLDT BAY UNIT 3
OPERATING FLOOR PLAN (EL. 27'-0")

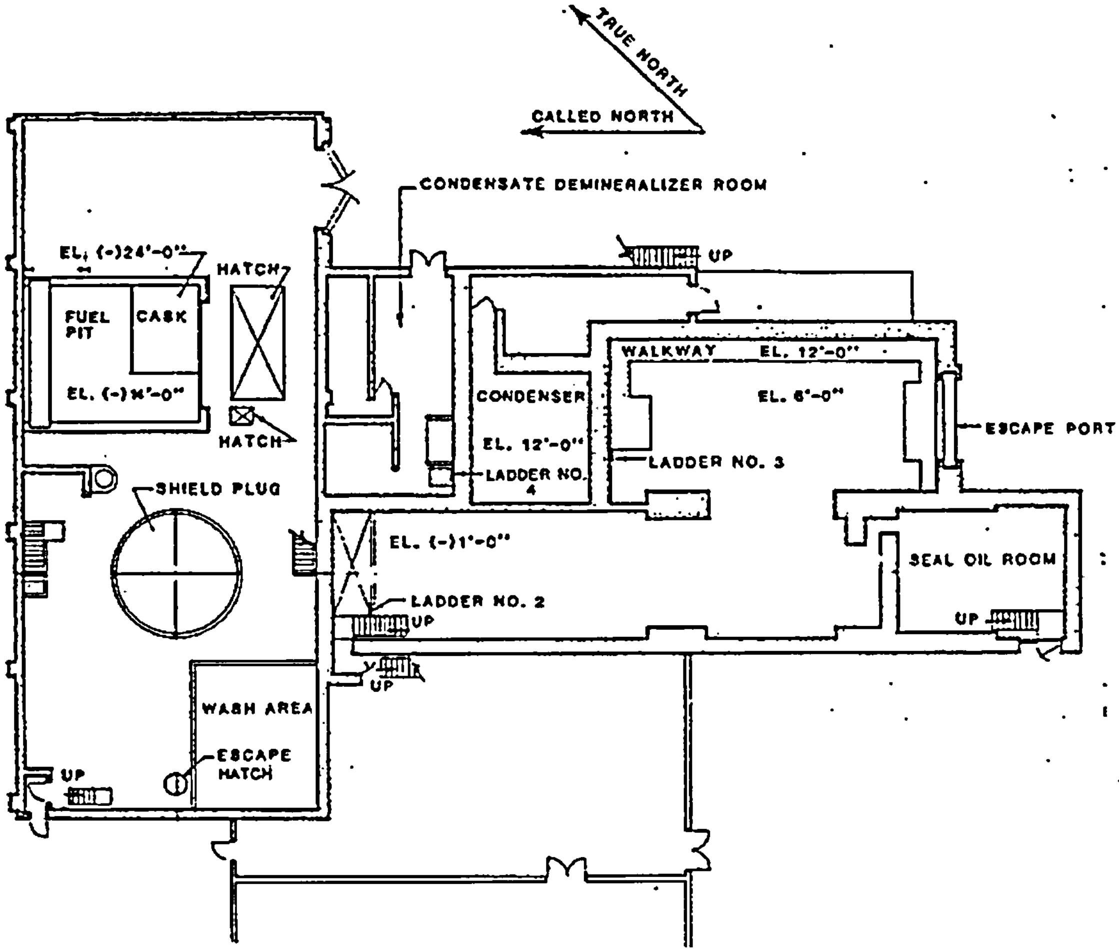


FIGURE 2-3
HBPP UNIT 3 GROUND FLOOR PLAN

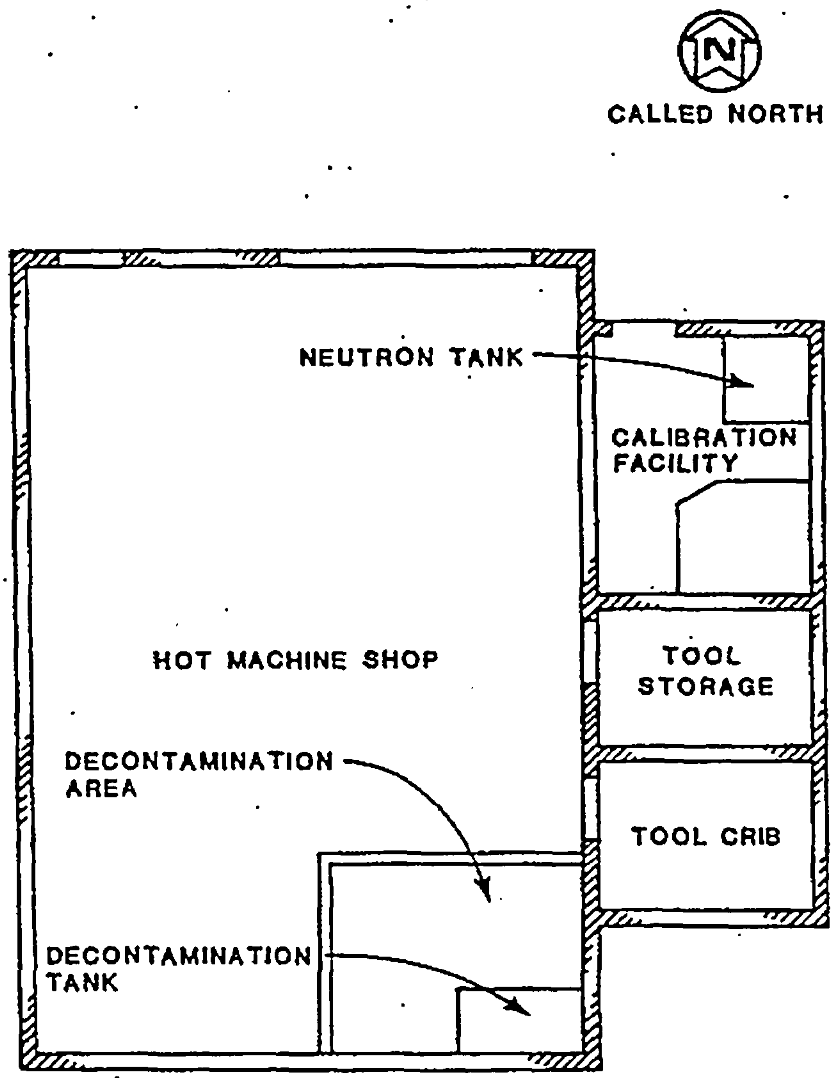


FIGURE 2-4
HOT MACHINE SHOP
AND CALIBRATION FACILITY

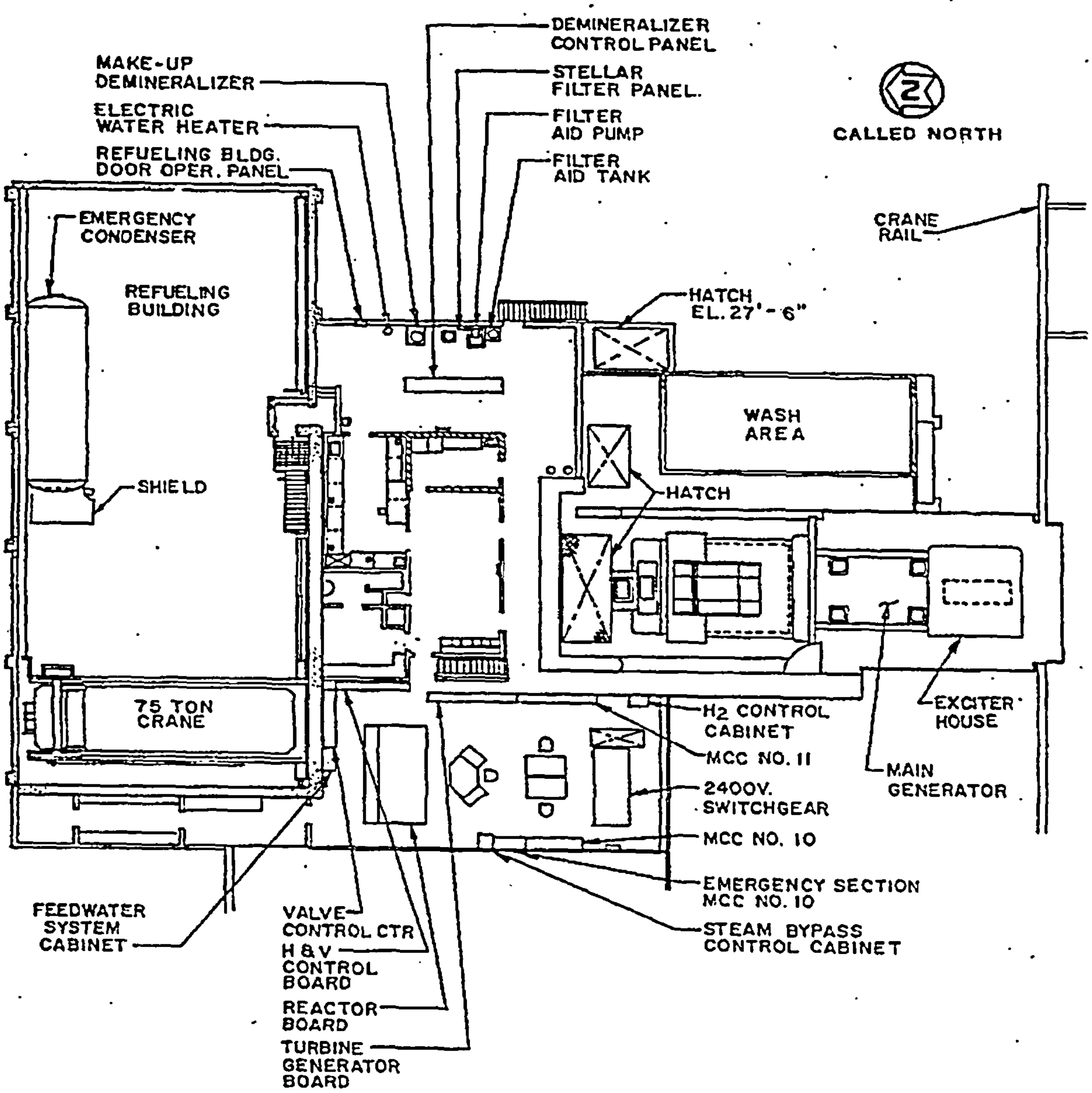


FIGURE 2-5
EQUIPMENT LOCATION – OPERATING FLOOR PLAN
(EL 27"-0")

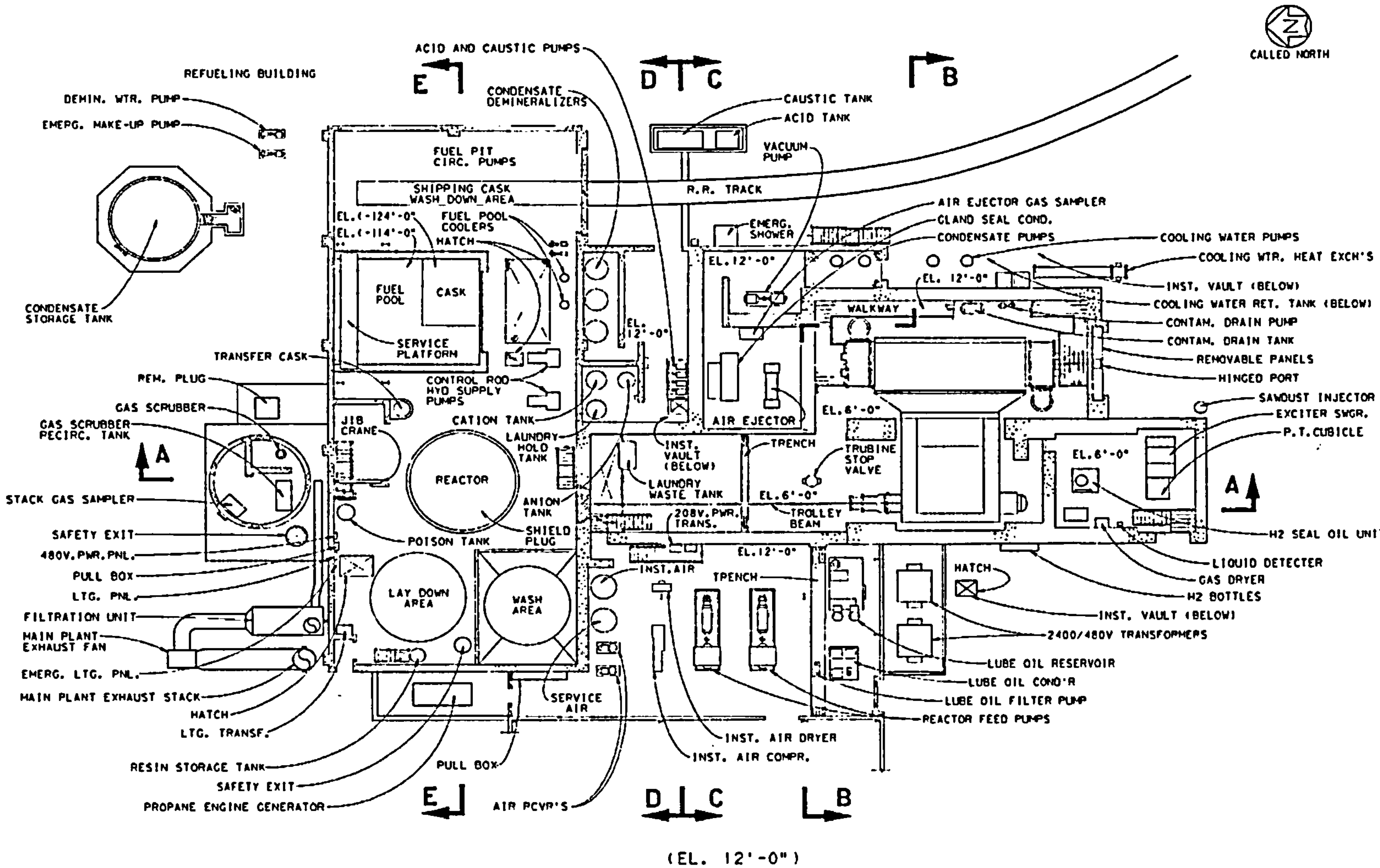


FIGURE 2-6
EQUIPMENT LOCATION - GROUND FLOOR PLAN

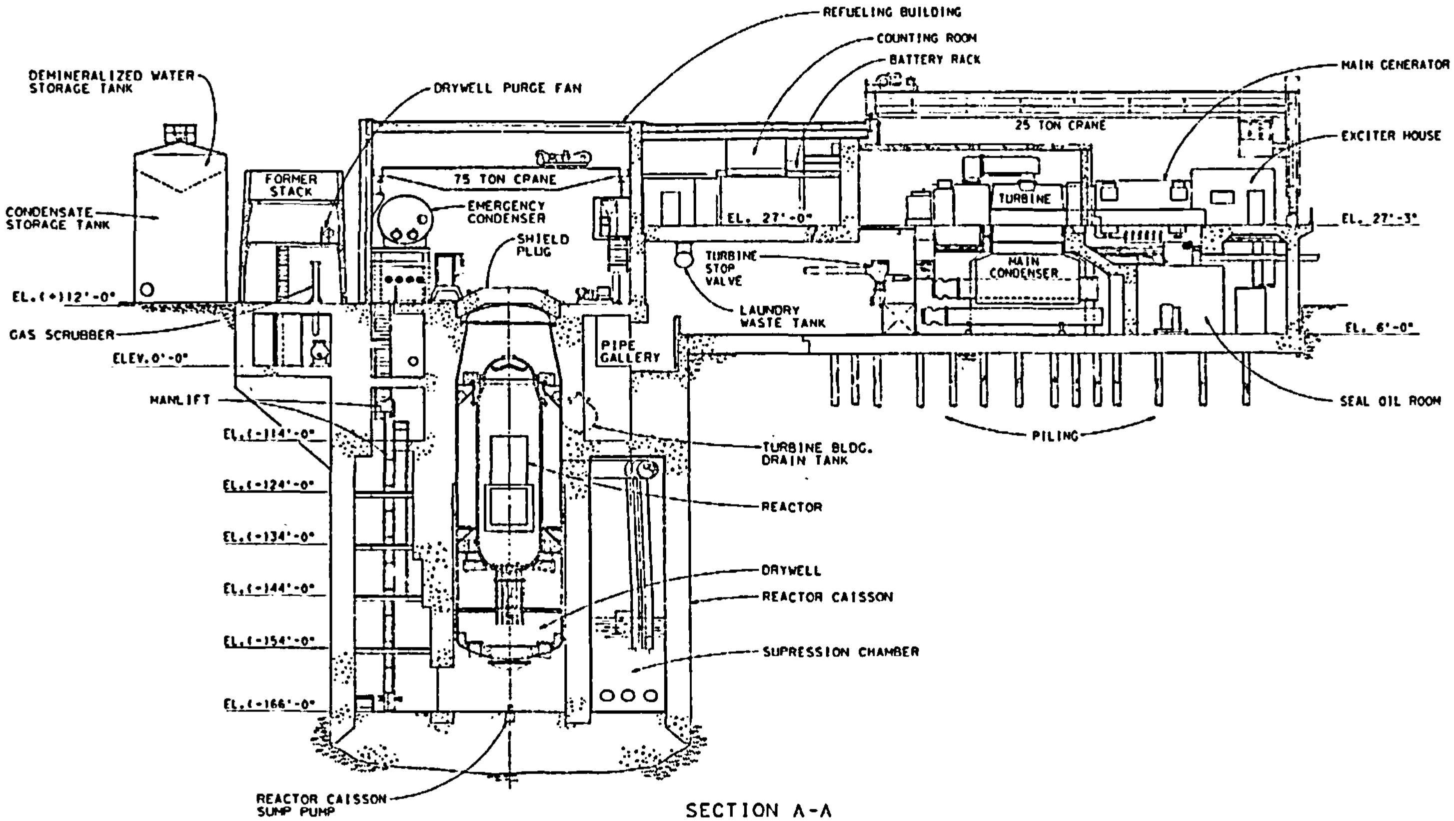
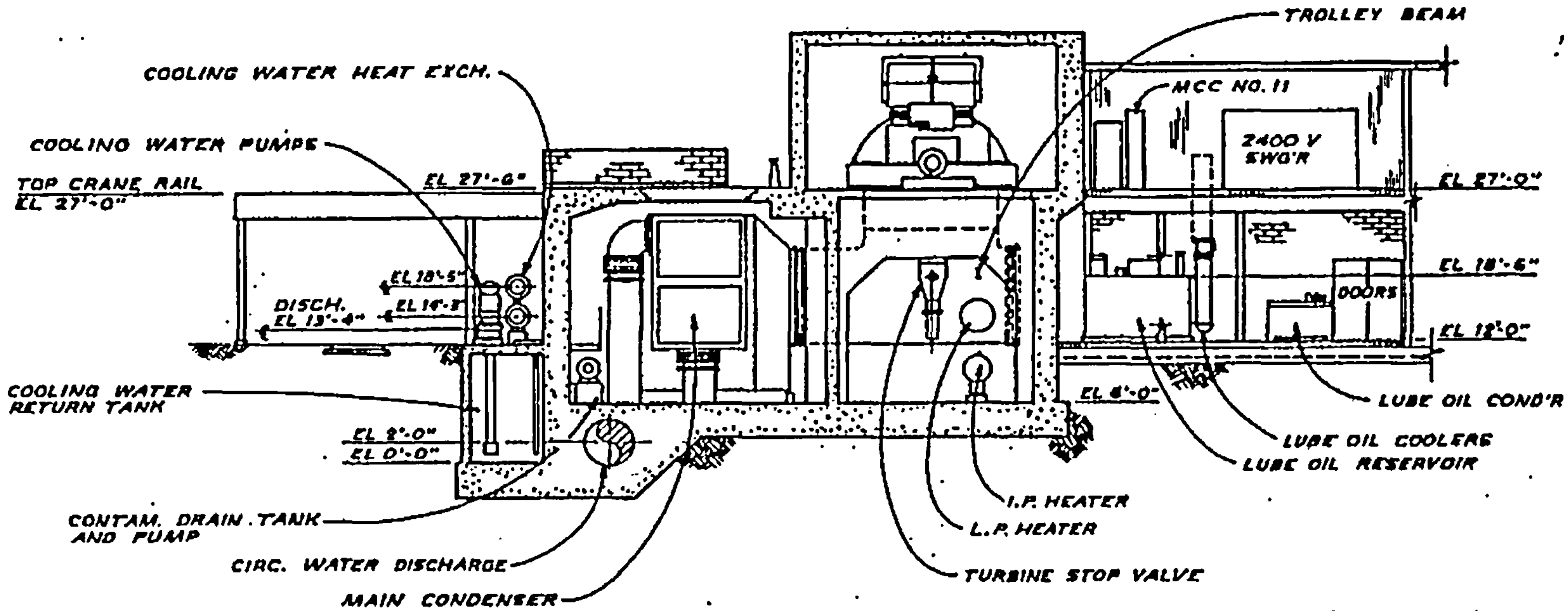
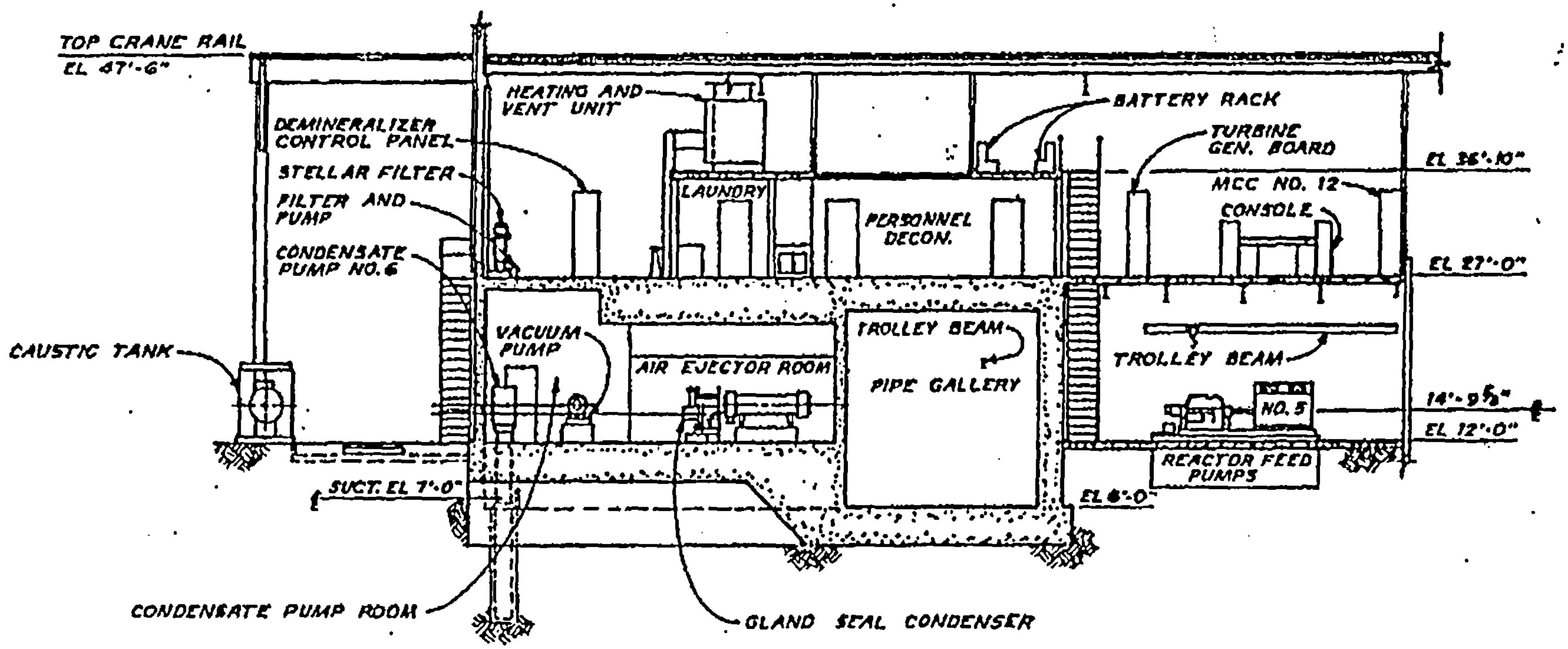


FIGURE 2-7
EQUIPMENT LOCATIONS



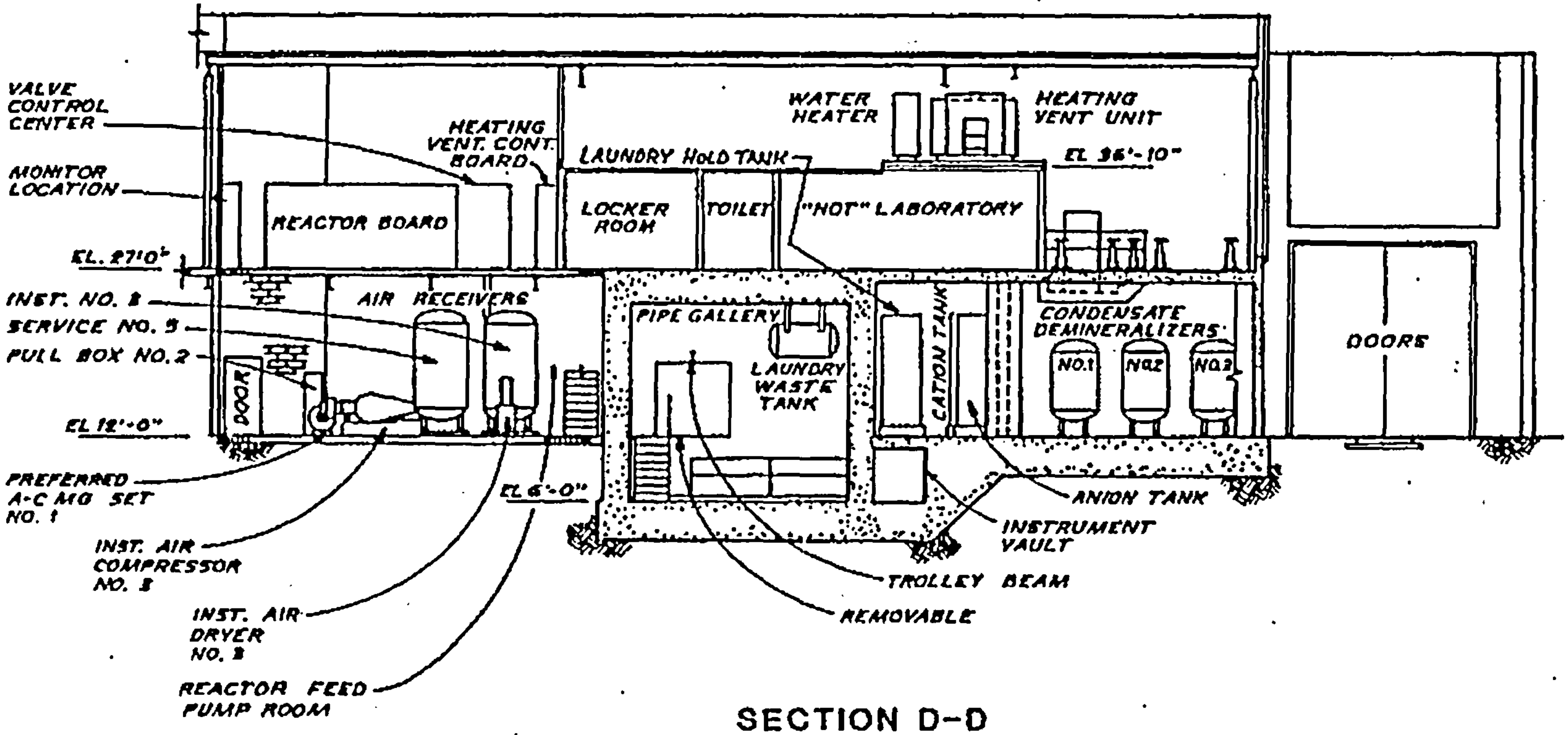
SECTION B–B

FIGURE 2-8
EQUIPMENT LOCATIONS



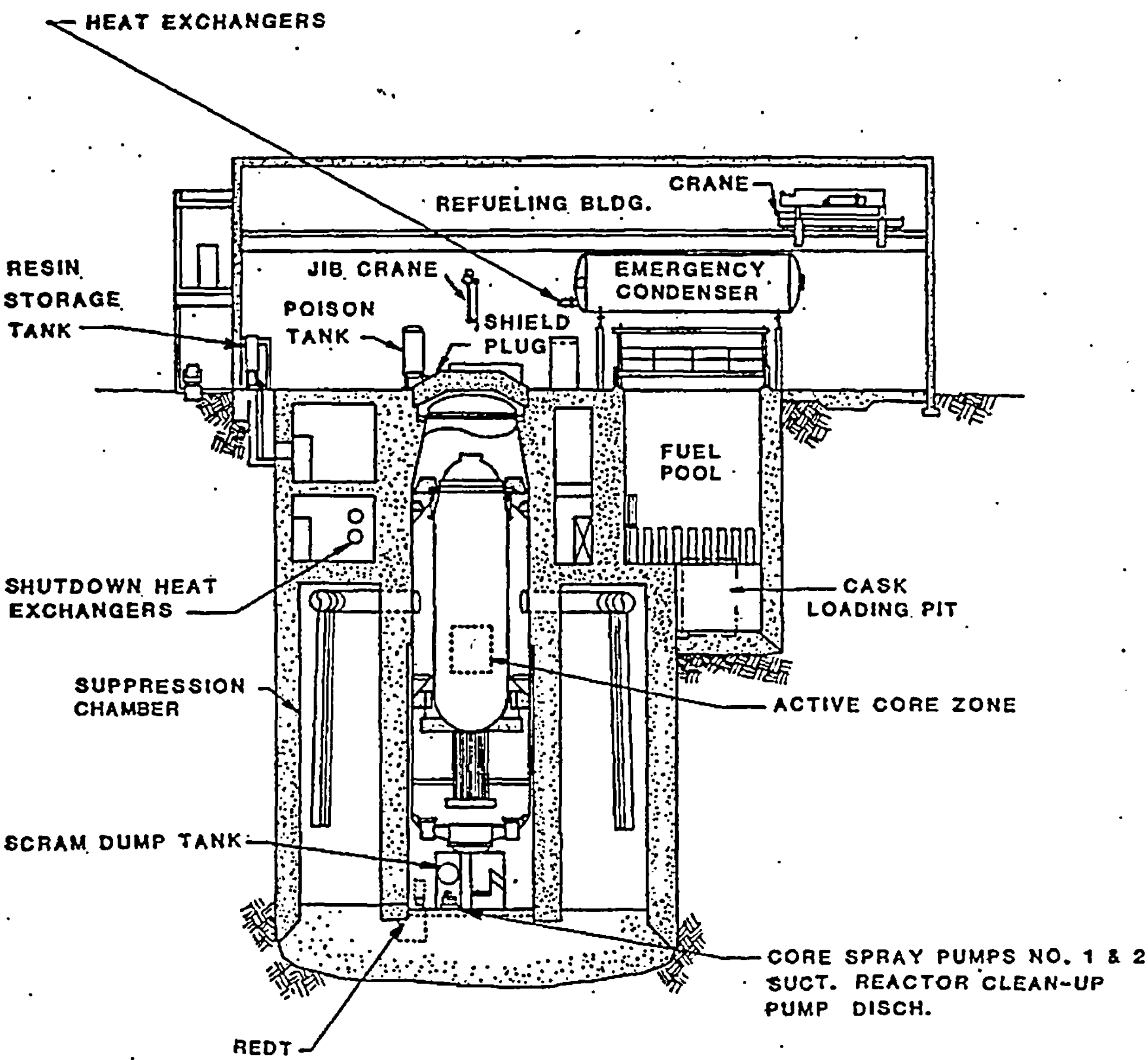
SECTION C–C

FIGURE 2-9
EQUIPMENT LOCATIONS



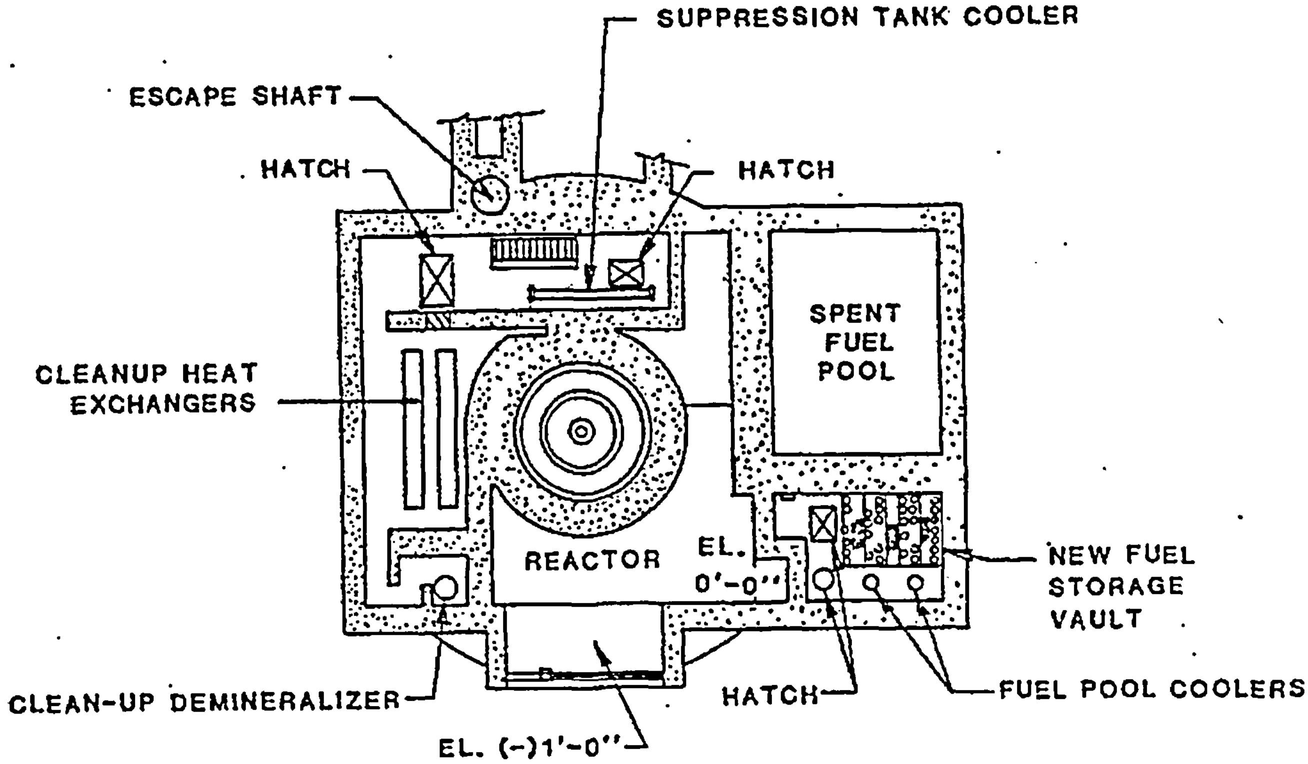
SECTION D-D

FIGURE 2-10
EQUIPMENT LOCATIONS



SECTION E-E

FIGURE 2-11
EQUIPMENT LOCATION



PLAN AT EL. (-) 2'-0"

FIGURE 2-12
EQUIPMENT LOCATION

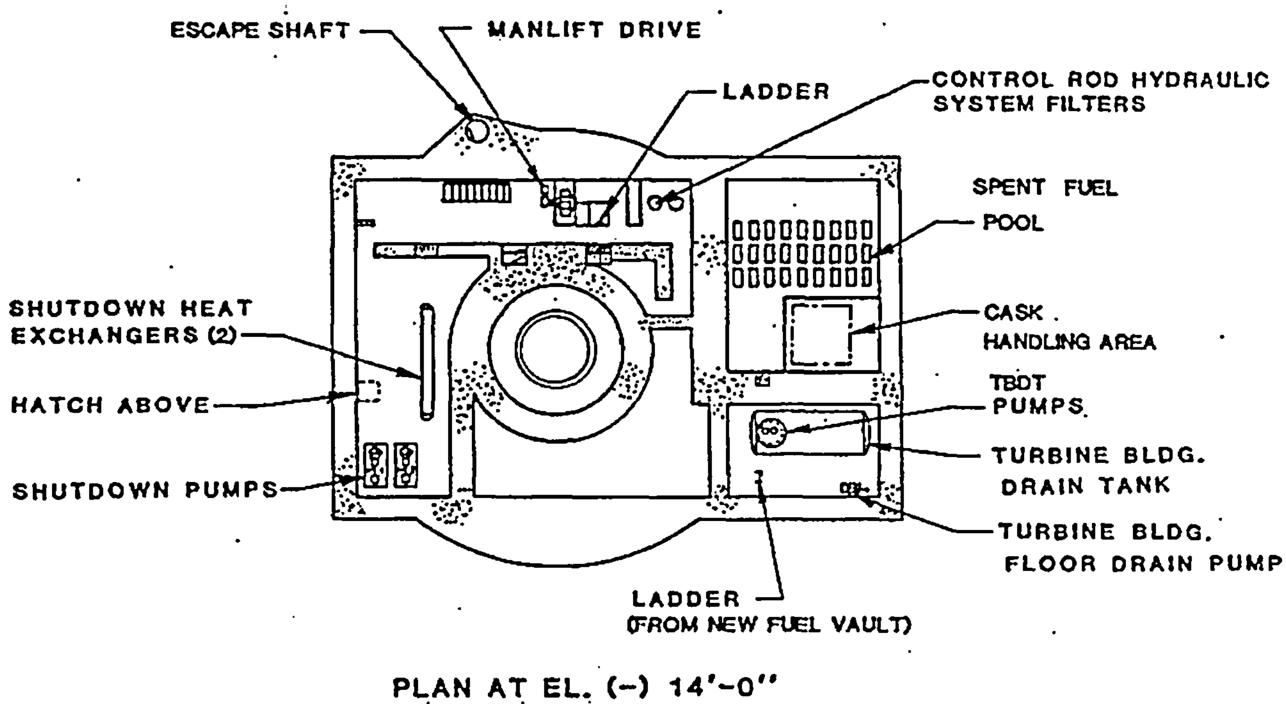
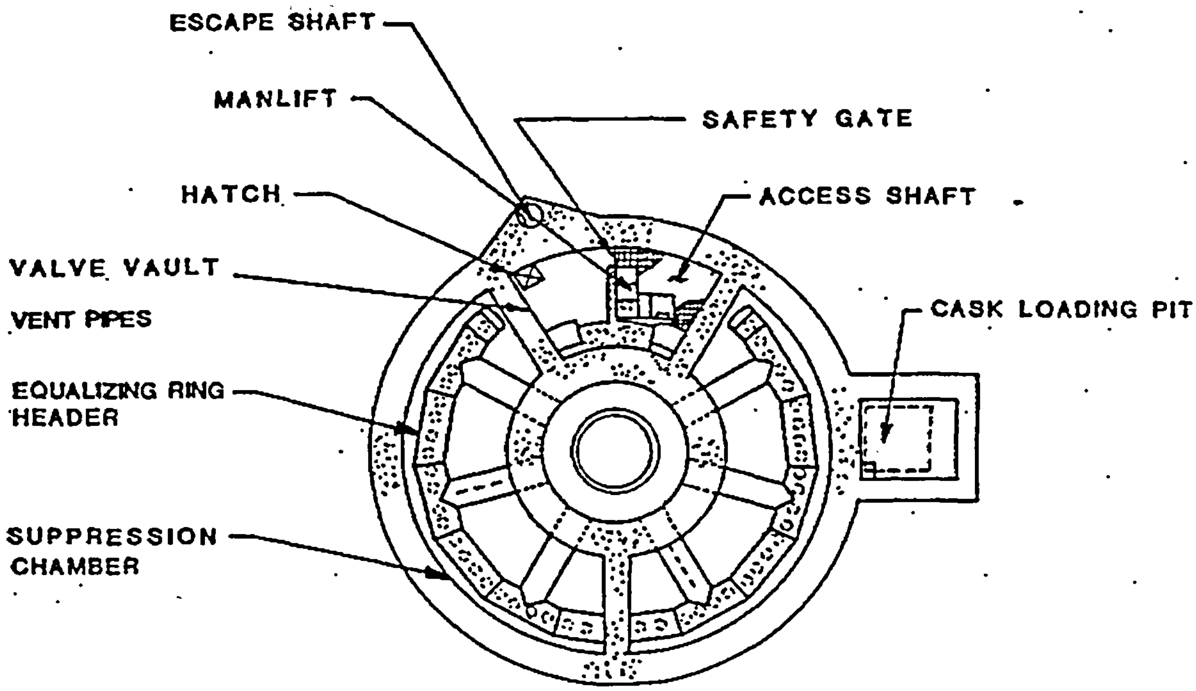
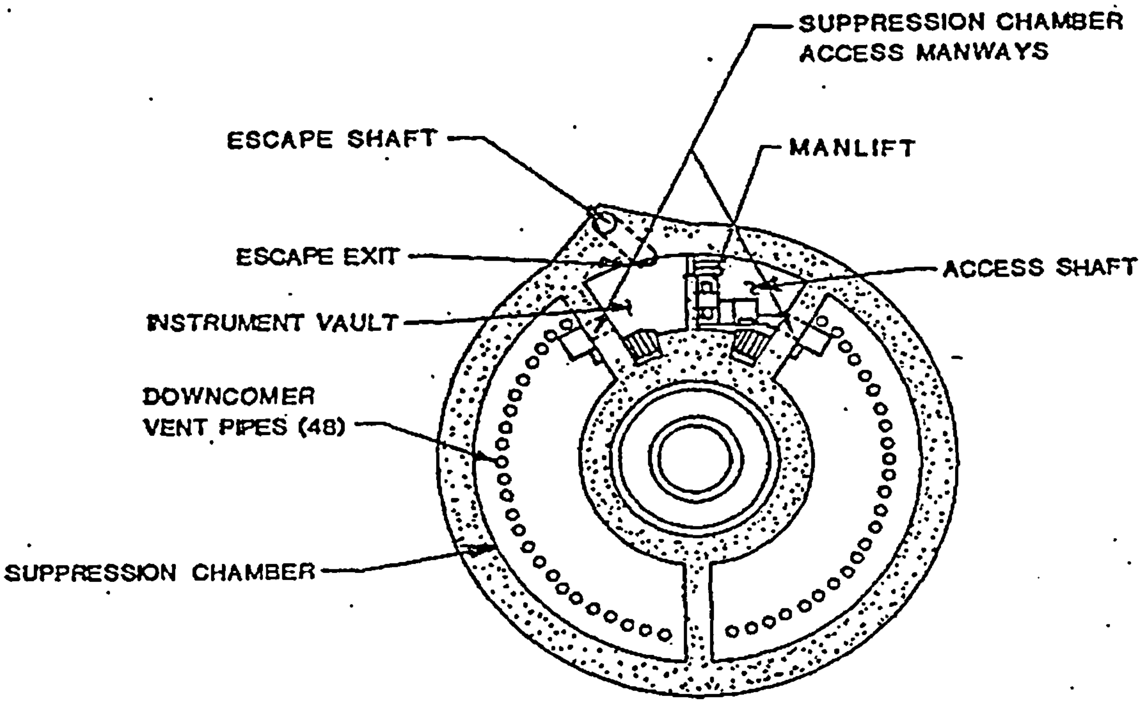


FIGURE 2-13
EQUIPMENT LOCATION



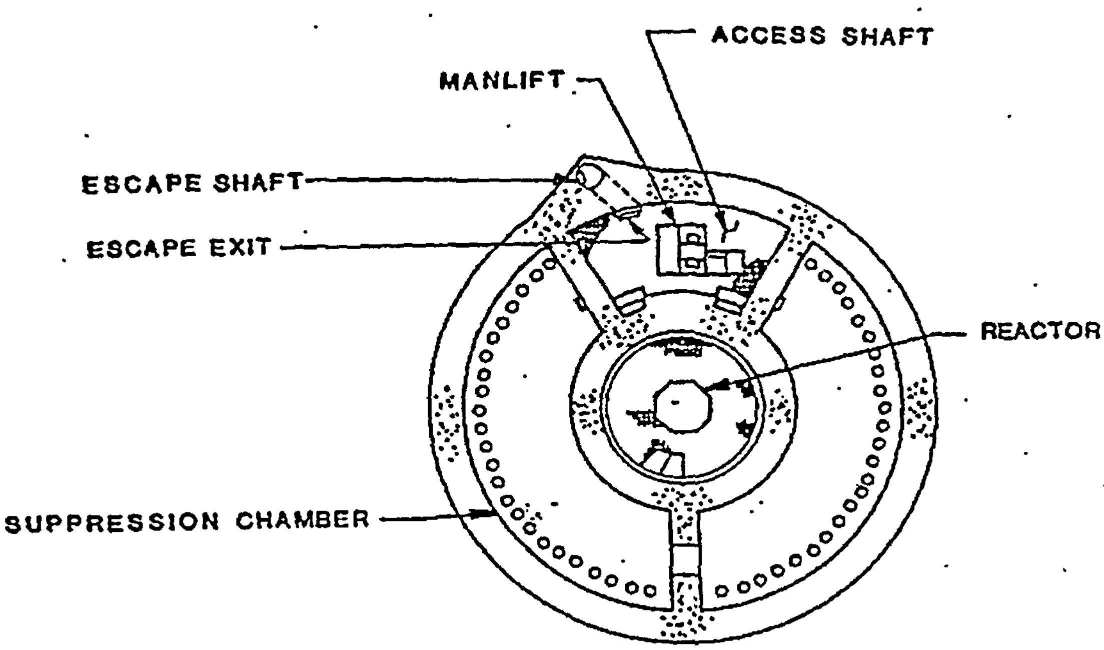
PLAN AT EL. (–)24'–0"

FIGURE 2-14
EQUIPMENT LOCATION



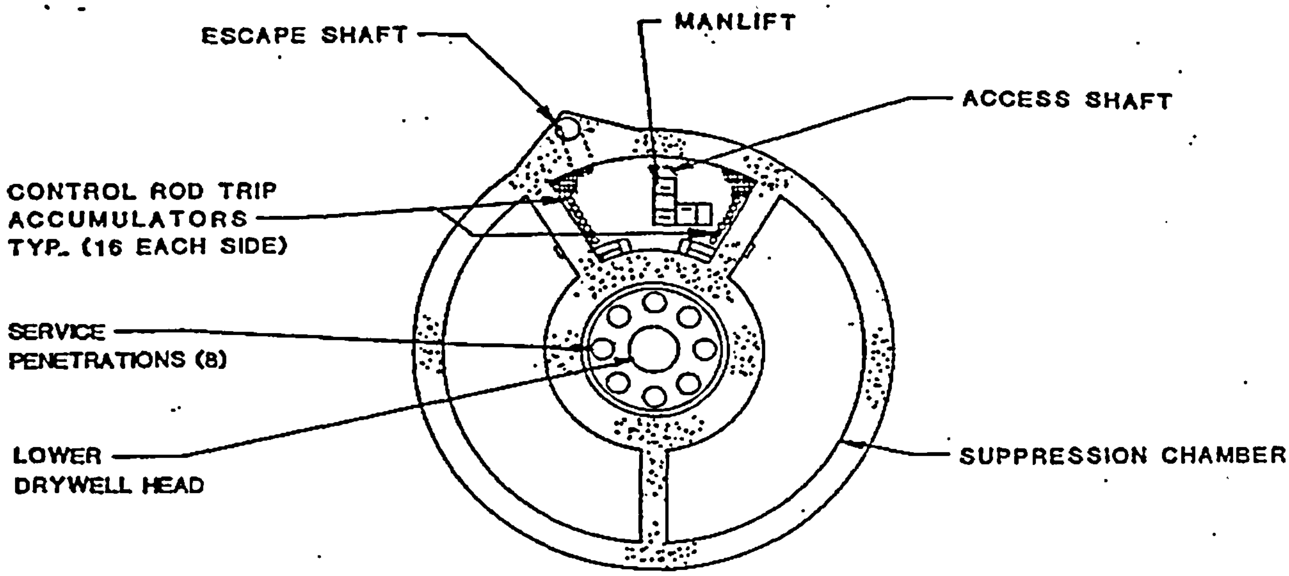
PLAN AT EL. (–)34'–0"

FIGURE 2-15
EQUIPMENT LOCATION



PLAN AT EL. (-)44'-0"

FIGURE 2-16
EQUIPMENT LOCATION



PLAN AT EL. (-)54'-0"

FIGURE 2-17
EQUIPMENT LOCATION

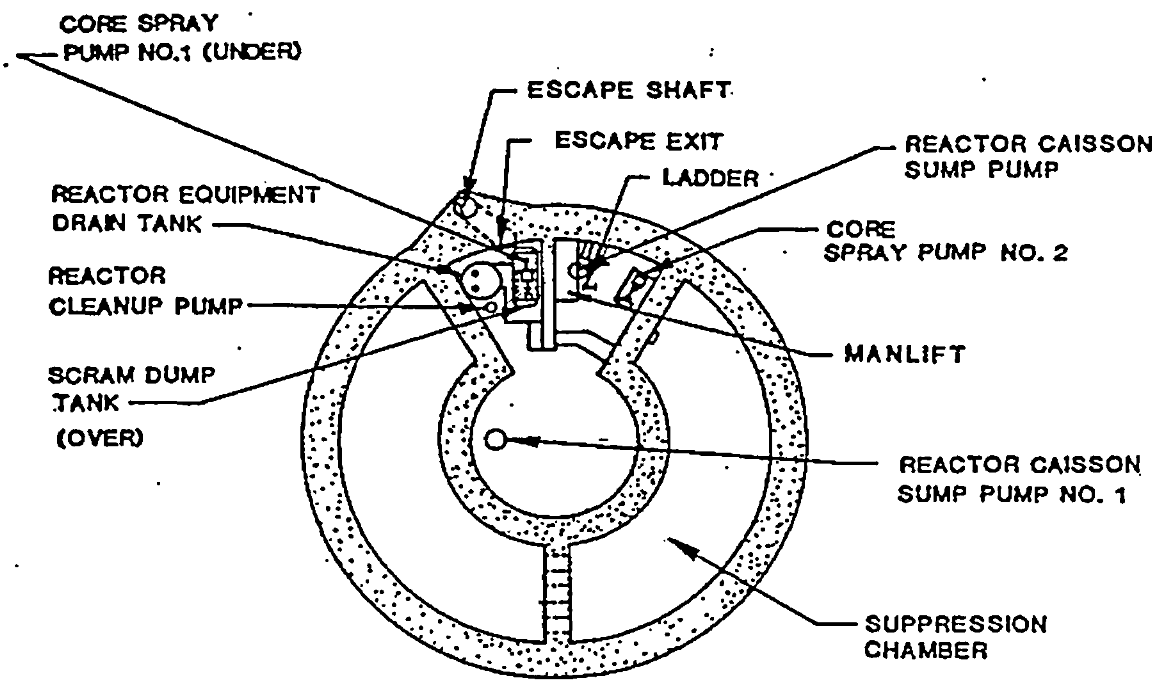


FIGURE 2-18
EQUIPMENT LOCATION
PLAN AT EL. (-) 66'-0"

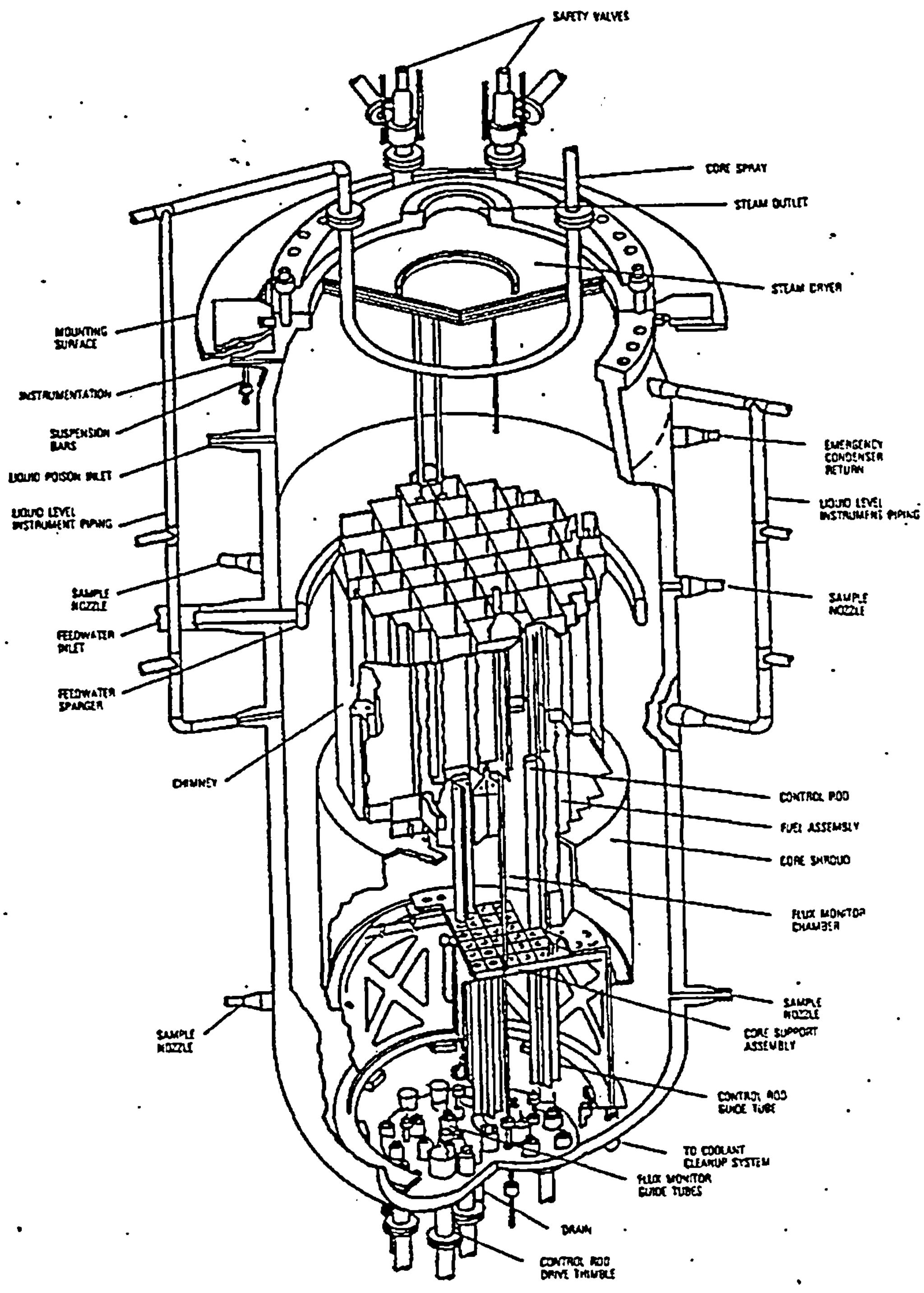


FIGURE 2-19
SCHEMATIC DIAGRAM OF REACTOR PRESSURE VESSEL AND INTERNALS FOR HBPP UNIT 3

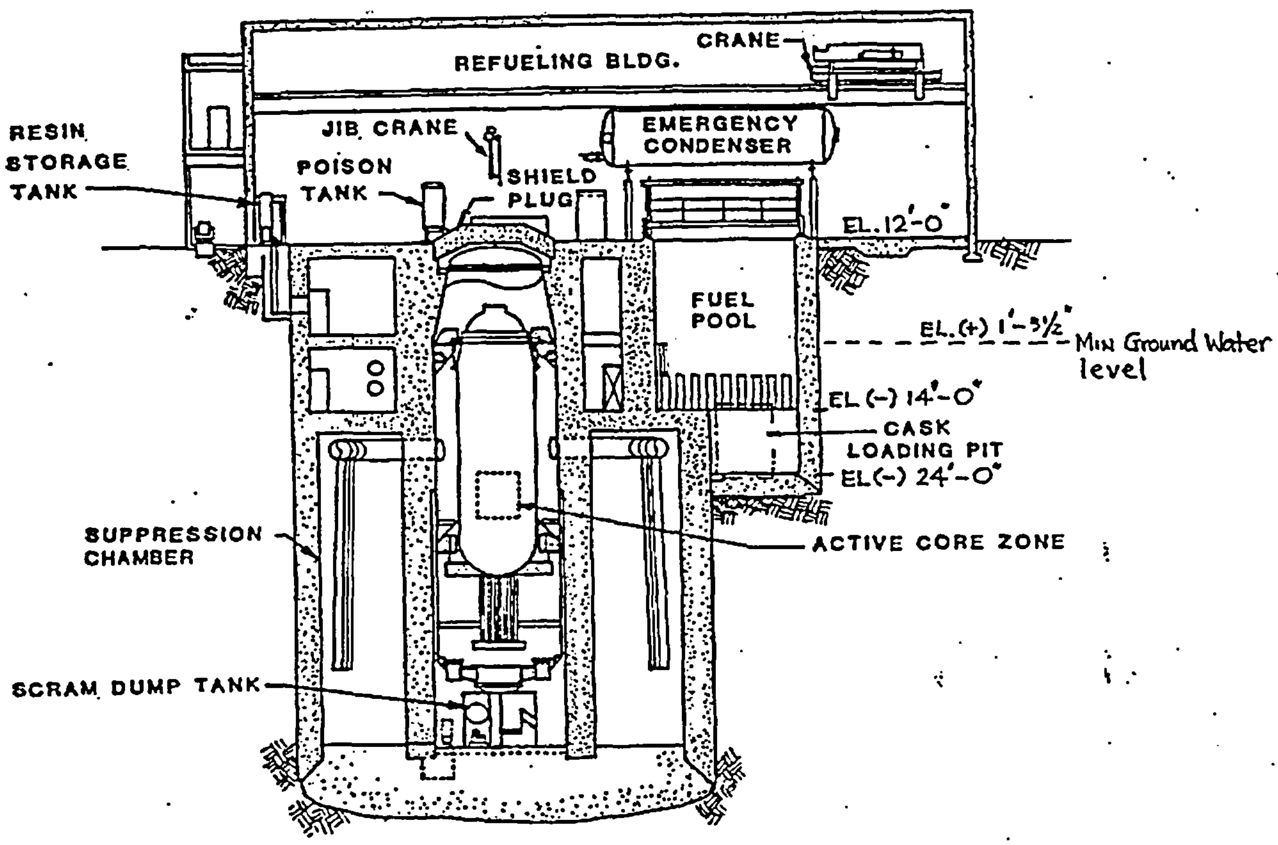


FIGURE 1
HUMBOLDT BAY PLANT SECTIONAL VIEW

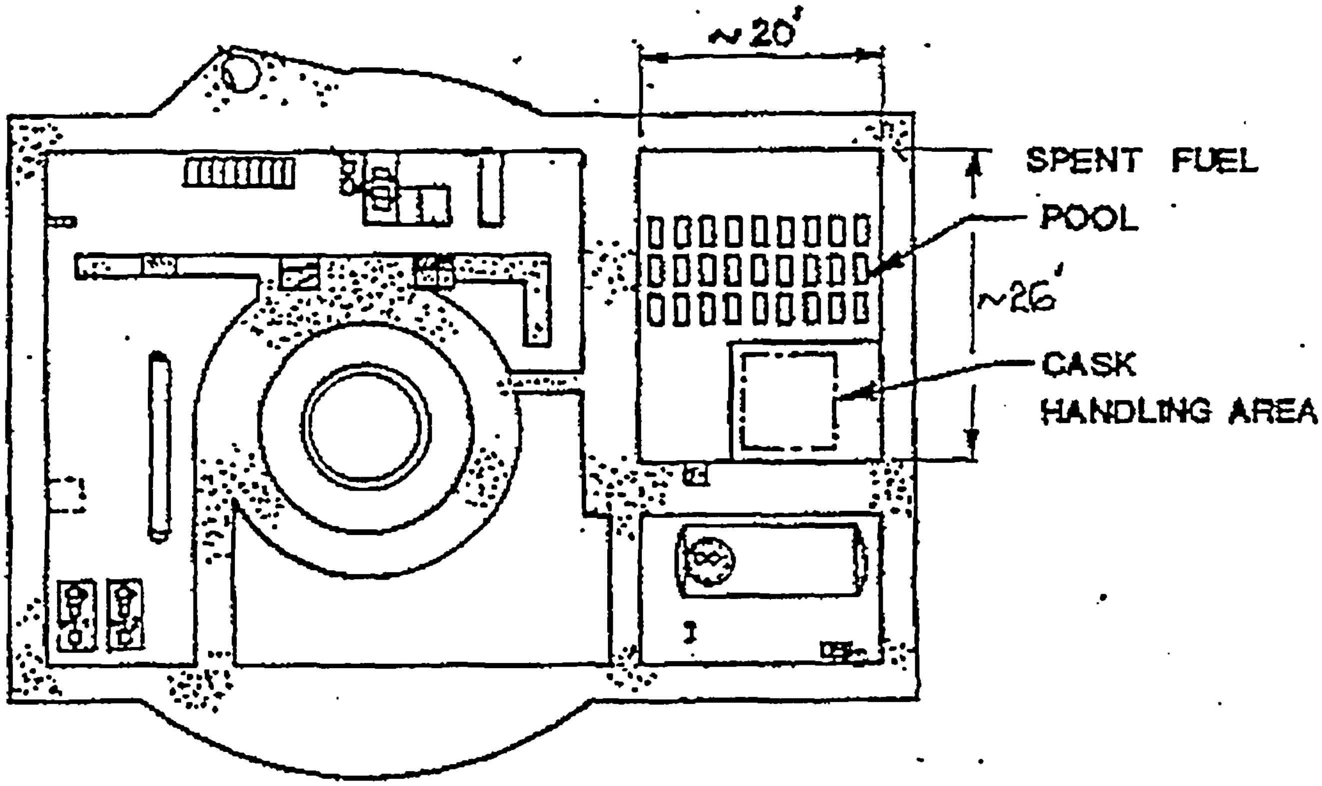


FIGURE 2
HUMBOLDT BAY PLANT PLAN VIEW

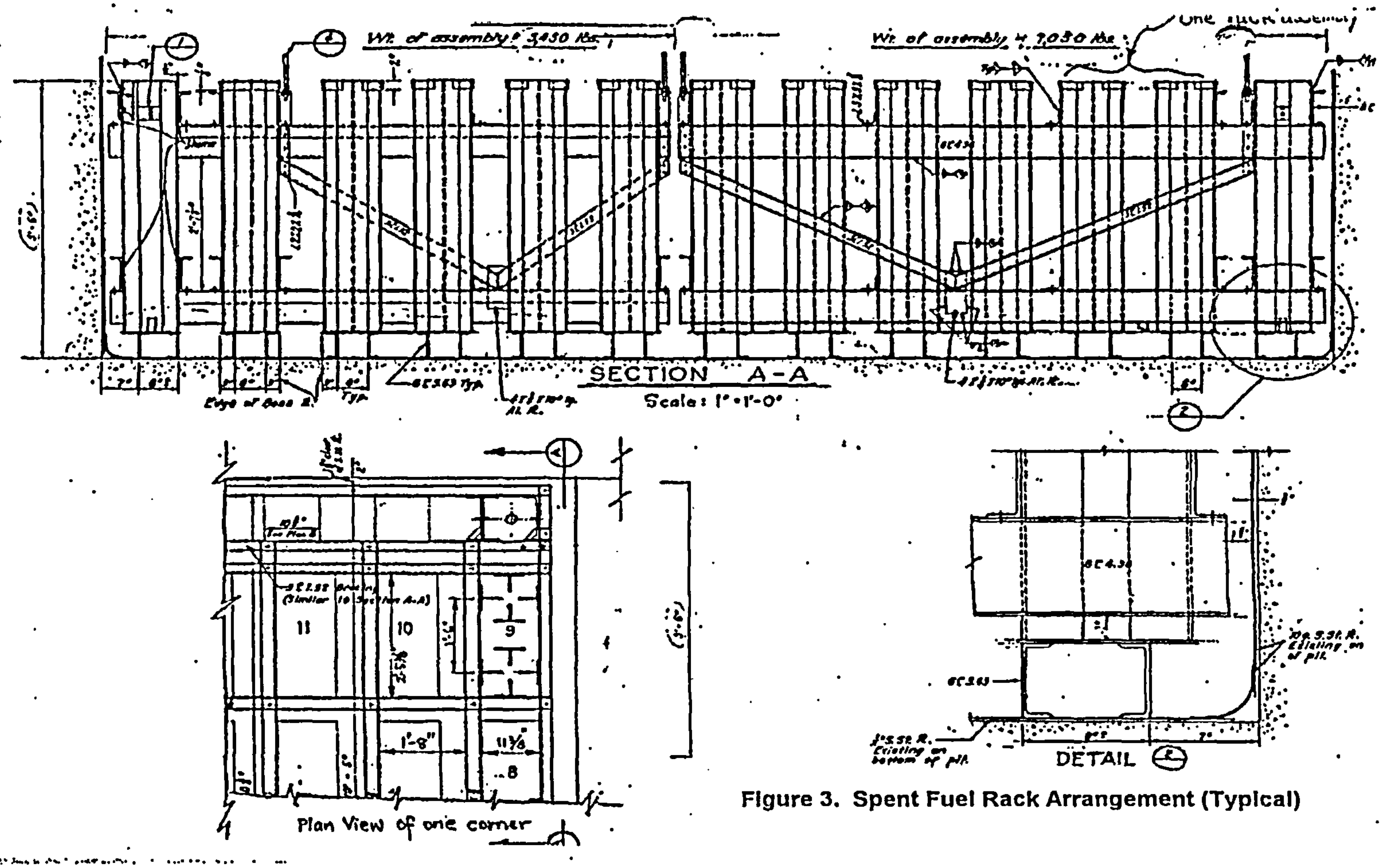


Figure 3. Spent Fuel Rack Arrangement (Typical)

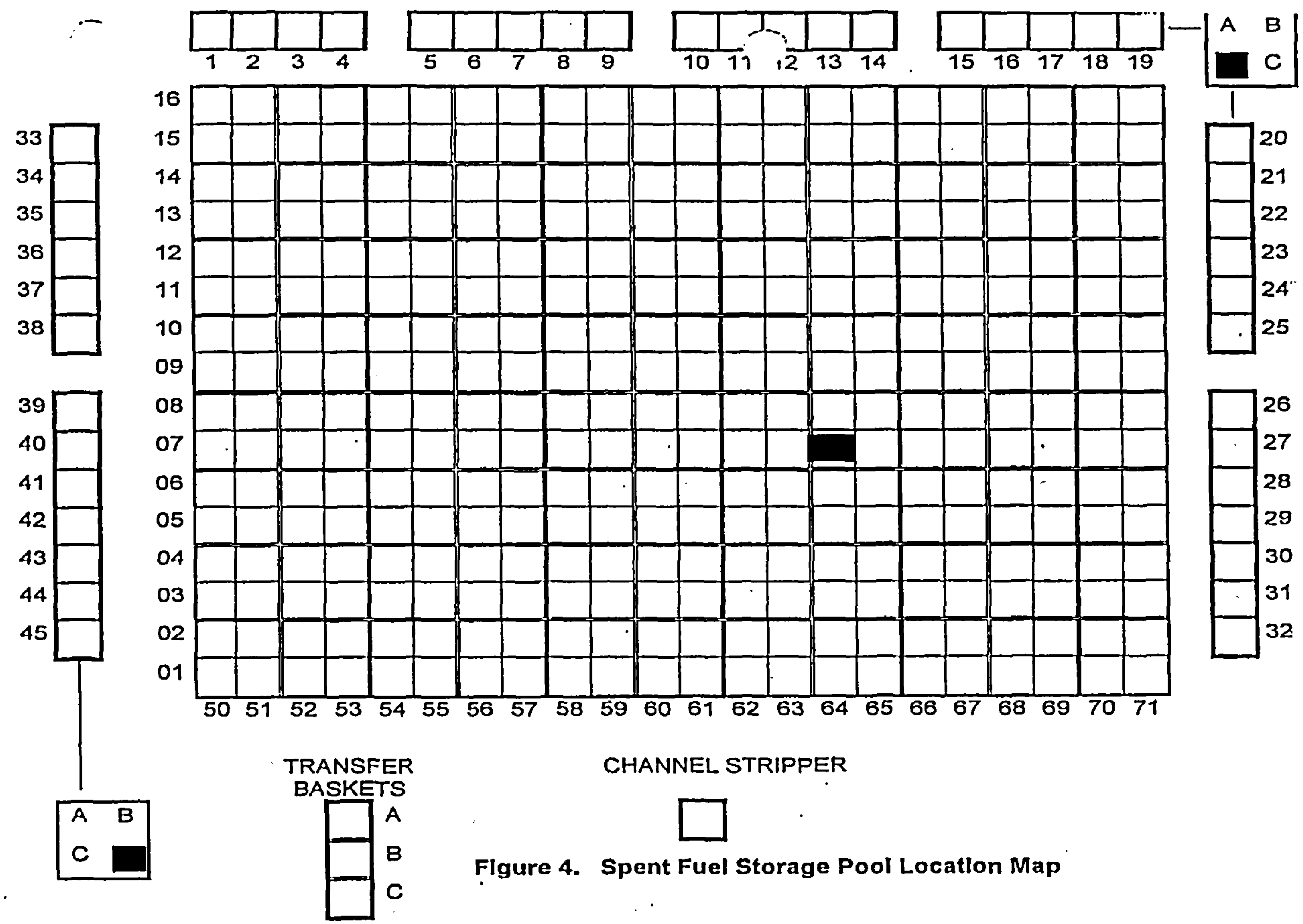


Figure 4. Spent Fuel Storage Pool Location Map

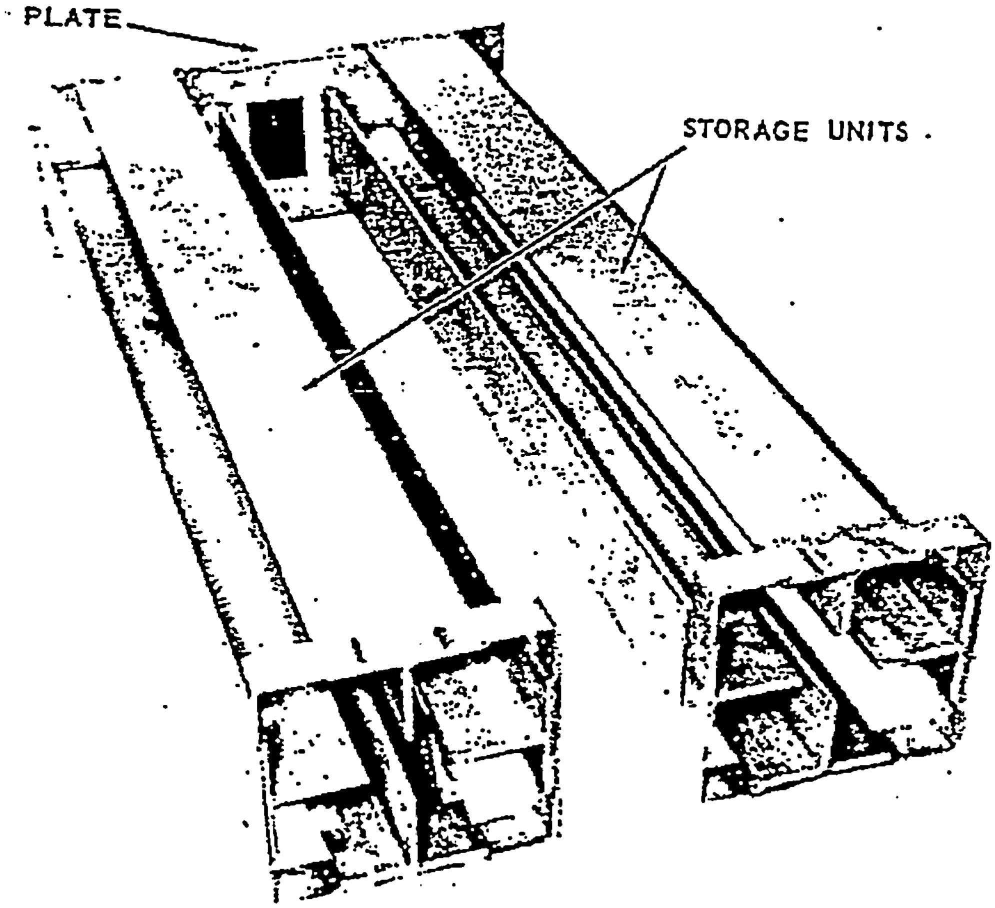
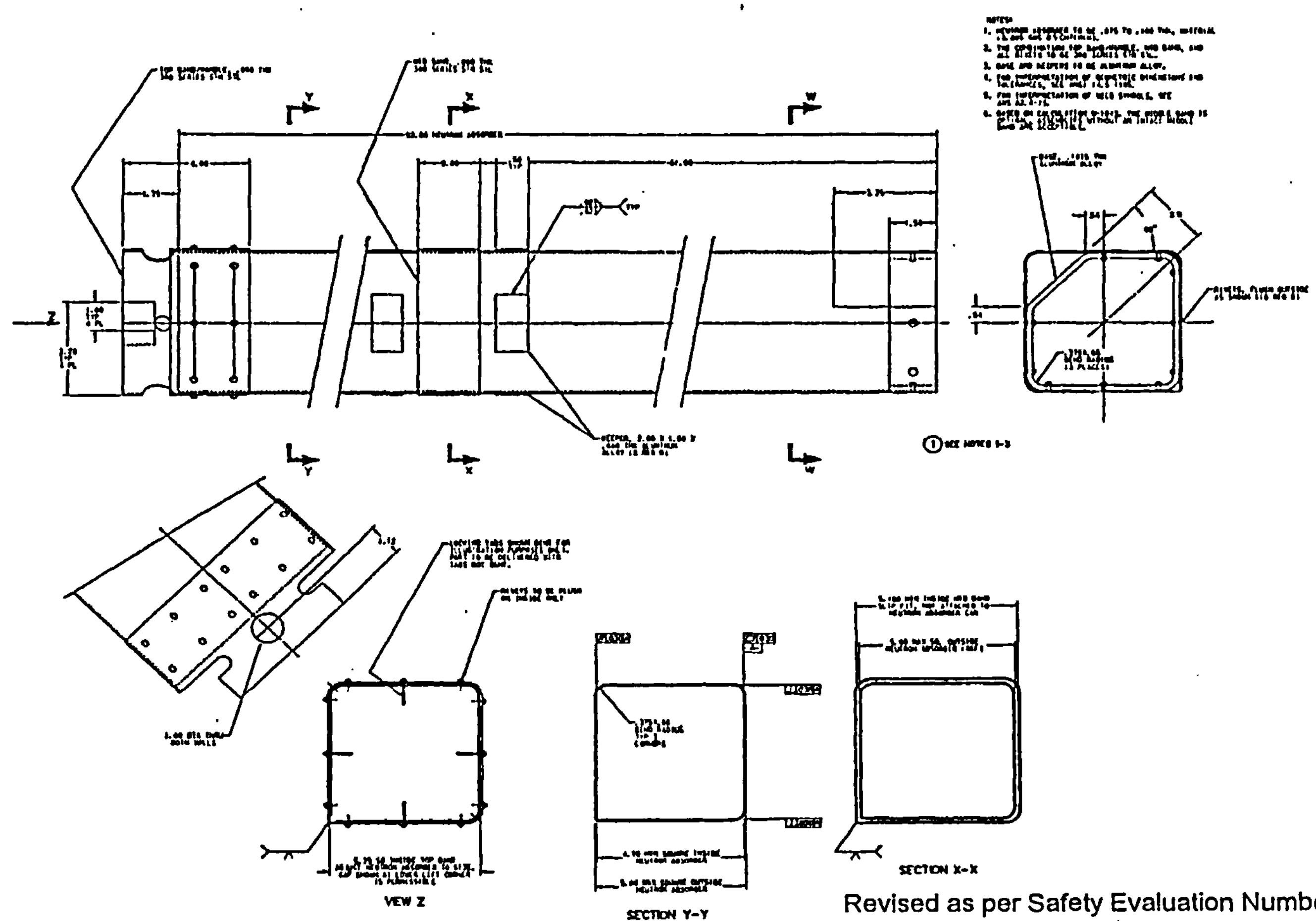


FIGURE 5
HUMBOLDT BAY POWER PLANT STORAGE RACKS



Revised as per Safety Evaluation Number 2000-10

FIGURE 6
FUEL ASSEMBLY PROTECTIVE CAN
(EXCERPTED FROM 60199924-17)