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PG&E Letter HBL-10-005

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Docket No. 50-133, OL-DPR-7
Humboldt Bay Power Plant, Unit 3
Revision 8 to the Defueled Safety Analysis Report, Revision 26 to the Humboldt Bay Quality Assurance Plan, and Revision 3 to the Technical Specifications Bases

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.71(e), Pacific Gas and Electric Company (PG&E) is submitting the Humboldt Bay Power Plant (HBPP), Unit 3 Defueled Safety Analysis Report (DSAR), Revision 8, in accordance with 10 CFR 50.4(b)(6). The changes contained in Revision 8 of the DSAR include plant modifications performed since DSAR, Revision 7, which was submitted to the NRC in PG&E Letter HBL-08-003, dated March 7, 2008, and reflects current plant configuration and organization.

Enclosure 1 contains Revision 8 of the DSAR in its entirety. The technical changes were made under the provisions of 10 CFR 50.59 and have not been previously submitted to the NRC. Revision bars are not shown in Revision 8 due to the extensive changes made since Revision 7 was submitted to the NRC.

PG&E is also submitting Revision 26 of the Humboldt Bay Quality Assurance Plan (QAP) in accordance with 10 CFR 50.54(a)(3). Enclosure 2 contains Revision 26 to the Humboldt Bay QAP in its entirety. Revision bars are not shown in Revision 26 due to the extensive changes made since Revision 21 was submitted to the NRC in PG&E Letter HBL-08-003, dated March 7, 2008.

In addition, PG&E is submitting Revision 3 of the HBPP Technical Specifications Bases in accordance with HBPP Technical Specification 5.6.2.d. Enclosure 3 contains Revision 3 to the HBPP Technical Specifications Bases in its entirety. Changes made since Revision 2 was submitted to the NRC in PG&E Letter HBL-06-004, dated March 13, 2006, are noted with revision bars. The changes were made under the provisions of 10 CFR 50.59, and have not been previously submitted to the NRC.

MMSS01



I state under penalty of perjury that the foregoing is true and correct.

Executed on March 3, 2010.

If you have any questions regarding this submittal, please contact Mr. David Sokolsky at (707) 444-0801.

Sincerely,

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dds2/0801

Enclosures

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PG Fossil Gen HBPP Humboldt Distribution

Enclosure 1
PG&E Letter HBL-10-005

**HBPP UNIT 3 DEFUELED SAFETY
ANAYLSIS REPORT, REVISION 8**

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 continues to submit DSAR revisions in accordance with 10 CFR 50.71.

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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 and decommissioning period.

Section 2.0 contains a description of the facility; including Unit 3 plant structures and systems.

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 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.

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

Appendix A contains accident analysis for Unit 3 during the SAFSTOR period and decommissioning activities.

Appendix B contains restriction to preclude inadvertent criticality in the unlikely event that spent fuel fragments are discovered during dismantlement of the reactor vessel.

Appendix C contains figures developed at the time Unit 3 entered SAFSTOR, as well as figures developed for cask loading activities. The figures are provided for general information purposes.

1.2 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.

1.2.1 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.

1.2.2 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.

1.2.2.1 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.

1.2.2.2 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.

1.2.2.3 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 decommissioning.

1.2.3 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.

1.2.4 LICENSING PREPARATIONS FOR DECOMMISSIONING

NRC issued License Amendment No. 41 on February 14, 2008, allowing for the deletion or relocation of numerous sections of the plant technical specifications to be effective upon completion of the transfer of spent fuel from the spent fuel pool into the Independent Spent Fuel Storage Installation (ISFSI), which is located within the Owner Controlled Area.

On June 16, 2008, the NRC issued License Amendment 43, allowing the deletion of the Physical Security Plan to be effective upon completion of the transfer of spent fuel from the spent fuel pool into the ISFSI.

On December 11, 2008, PG&E completed the transfer of spent fuel from the spent fuel pool into the ISFSI. On December 22, 2008, the NRC issued rescissions of NRC Order pertaining to interim safeguards and security compensatory measures, and additional security measures associated with access authorization and fitness for duty, effective December 22, 2008.

1.3 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.

1.3.1 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.

1.3.2 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 foot 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.

1.3.3 Hydrology

1.3.3.1 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.

1.3.3.2 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.

1.3.3.3 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.

1.3.4 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.

1.3.5 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.

1.4 SAFSTOR AND DECOMMISSIONING ACTIVITIES

During the period of SAFSTOR, PG&E will perform major decommissioning activities after the completion of spent fuel transfer into the ISFSI. Decommissioning activities and schedule are described in the PSDAR.

2.0 FACILITY DESCRIPTION

The HBPP site is comprised of two fossil-fueled units (Unit 1 - 52 MWe and Unit 2 - 53 MWe), a single nuclear unit (Unit 3 - 63 MWe), two gas turbine-powered mobile emergency power plants (MEPP No. 2 and 3 - 15 MWe each), and an Independent Spent Fuel Storage Installation (ISFSI). The 163 Mwe fossil-fueled Humboldt Bay Generating Station is under construction and will replace Units 1 and 2 and the MEPPS. Necessary support structures, equipment, and tanks are also located on the plant site. A site plan is shown in Appendix C, Figure C-1.

The principal activities of the Plant are related to the generation and transmission of electric power and the associated service activities. Activities associated with Unit 3 consist of monitoring and surveillance of the decommissioned facility and decontamination and dismantlement in preparation for the 10 CFR Part 50 license termination. The ISFSI stores spent fuel from Unit 3 under a 10 CFR Part 72 license.

Unit 3, which was permanently shut down and defueled in 1984, consisted of a General Electric natural circulation, single cycle boiling water reactor, the associated turbine-generator, and necessary support and auxiliary systems.

Liquid wastes are processed in the radwaste treatment building, located in an excavated portion of an earthen embankment north of the refueling building. A steel building encloses the entire liquid radwaste treatment area. During decommissioning activities, alternative and equivalent methods may be established to process liquid wastes.

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.

2.1 PLANT STRUCTURES

The plant structures are shown in Appendix C, Figures C-2 through C-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.

2.1.1 POWER BUILDING

The power building is not required to function except for ALARA considerations to minimize potential occupational personnel exposures and for safe storage of contaminated systems, structures and components. Negative pressure to the outside with flow to the refueling building is maintained by ventilation equipment to limit unmonitored releases to the environment.

2.1.2 REFUELING BUILDING

The refueling building is not required to function except for ALARA considerations to minimize potential occupational personnel exposures and for safe storage of irradiated and contaminated systems, structures and components until decommissioned.

The refueling building ventilation system will remain operational during decommissioning (see DSAR section 2.2.3). The refueling building ventilation system will be operational when decommissioning activities are being performed that have the potential to create airborne radioactivity.

During decommissioning activities, alternative and equivalent methods may be established for work area ventilation. The alternative methods will include filtration and monitoring of ventilation exhaust.

2.1.3 REACTOR CAISSON

The Reactor Caisson is not required to function except for ALARA considerations to minimize potential occupational personnel exposures and for safe storage of irradiated and contaminated systems, structures and components until decommissioned.

2.1.4 VENTILATION STACK

The 50-foot high, 48-inch diameter ventilation stack is made of carbon steel and is the single discharge point of plant airborne effluents.

2.1.5 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.

2.1.6 ONSITE COMBUSTIBLE FUEL STORAGE

The description of combustible fuel storage facilities at the HBPP is given in Table 2-1.

2.2 PLANT SYSTEMS DESCRIPTION

For this DSAR, the plant systems described are:

- Spent Fuel Pool and Liner Gap
- Waste Disposal
- Plant Fire Protection Features
- Ventilation
- Radiation Monitoring

The operational systems and major components comprising each of these major system groupings are described in the sections below.

2.2.1 SPENT FUEL STORAGE POOL

A spent fuel storage pool is integral to the reactor caisson. The water in the pool provides for shielding and contamination control. The spent fuel storage pool is approximately 20 feet wide by 26 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 completely covers 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.

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 Turbine Building Drain Tank (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 potential leakage of radioactive contaminants to the outside of the building.

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 pumps are used for water chemistry control.

2.2.2 WASTE DISPOSAL SYSTEMS

The waste disposal systems in Unit 3 include the gas treatment system, 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.

2.2.2.1 Gas Treatment System.

The gas treatment system (GTS) can be used to mitigate the release of airborne particulate radioactive material into the atmosphere of the refueling and turbine buildings during both normal D&D activities and accident situations. This system consists of two exhaust fans, a high efficiency particulate air (HEPA) filter, associated system piping, valves, instruments, and controls. The system components are located on three levels in the base of the former main ventilation exhaust stack.

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 with additional HEPA filtration prior to discharge through the main ventilation exhaust stack.

2.2.2.2 Liquid Waste Collection System.

The liquid waste collection system consists of the TBDT, reactor equipment drain tank (REDT), reactor caisson sump, two turbine building drain tank pumps, two reactor equipment drain tank pumps, the reactor caisson sump pumps, the laundry waste tank, and a yard drain system.

The TBDT 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 TBDT pump or can be valved to drain directly to the REDT via the caisson floor drain system. The TBDT will continue to be used during the SAFSTOR period along with the associated valves, pumps, and instrumentation and controls until the system is available for decommissioning.

The 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 REDT and its associated pumps will continue to be used throughout the SAFSTOR period until the system is available for decommissioning. 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 until the system is available for decommissioning. 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 will remain in operation throughout the SAFSTOR period until the system is available for decommissioning. The laundry system has been secured. The laundry waste tank remains in service in order to collect drains from respiratory cleaning 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. However, some cleaning of clothing, respirators and other material may be performed using the plant system described above.

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.

2.2.2.3 Liquid Waste Treatment System.

This system will remain operational throughout the SAFSTOR and decommissioning 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 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. Spent resins from 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. 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 and decommissioning 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.

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.

2.2.2.4 Solid Radwaste System

There are no specific solid radwaste processing systems. Solid radwaste characterization, size reduction, and packaging for transport will be performed by a variety of methods to support D&D activities.

2.2.3 SERVICE SYSTEMS

Plant Fire Protection Features. HBPP fire protection features are described in TBD-301, "Fire Hazards Analysis". The Fire Hazards Analysis provides the basis and HBPP position relative to NRC Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants during Decommissioning and Permanent Shutdown." The Fire Hazards Analysis references supporting procedures that further describe all elements of the HBAP A-13, "Fire Loss Prevention Program" including system checks, equipment description, systems description, administrative controls, personnel training, and fire response.

Main Unit 3 Ventilation System. The plant heating and ventilation system helps maintain the consequences of postulated decommissioning accidents acceptable and 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.

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 are provided to permit isolation of the refueling building from the remaining ventilated areas of Unit 3.

The structure that previously held the (never operational) condenser offgas treatment 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 to unoccupied areas of Unit 3. In addition, ventilation from the radwaste treatment facility has been tied into the ventilation system.

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, 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. The 75-ton hook will be required to support final plant dismantlement.

2.2.4 RADIATION MONITORING SYSTEMS

2.2.4.1 Process Radiation Monitoring System

The radwaste discharge line is monitored by a Radioactive Liquid Effluent Monitoring System which 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 signal conditioning and analysis equipment. The resulting count rate is displayed on a rate meter located in the Unit 3 control room. The rate meter displays the liquid count rate over a range enveloping 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. The Radioactive Liquid Effluent Monitor 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 and decommissioning. Grab samples are utilized as required to determine activity levels in other process streams.

2.2.4.2 Stack Radiation Monitoring System

The stack gas monitoring system consists of a sampling probe (located near the top of the 50 foot plant stack); and a monitoring skid containing a continuous monitor for particulate activity, a fixed particulate filter holder for effluent analysis, flow meters, and sample pump. The continuous monitor has the capability to monitor alpha emitting particulate radioactivity in the plant discharge. A nominal 2 cfm sample of stack air exhaust is continuously pulled from the sample probe. Approximately 1.2 cfm of the flow is pulled through the continuous monitor with the remainder being used for the periodic fixed filter change out for radioactive airborne effluent reporting. The sample pump discharges into plant ventilation ductwork leading back to the stack.

The particulate filter is replaced in accordance with the ODCM, and the old filter is analyzed in the plant laboratory to determine particulate activity in the stack effluent. Multiple filters may be composited and sent to off site analysis.

Table 2-1

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.	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.

3.0 RADIATION PROTECTION

3.1 RADIOLOGICAL CHARACTERIZATION

3.1.1 RADIONUCLIDE INVENTORY

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

These radionuclide sources are not readily dispersible in their present condition but will become more readily dispersible during decontamination and dismantlement activities. Although the remaining radioactive source term at the defueled Unit 3 reactor site has been greatly reduced by radioactive decay and spent fuel removal, there still exists significant quantities of transuranic contamination within plant systems. Due to the internal hazard risk to workers of this transuranic contamination, administrative and engineering controls will be utilized to limit airborne radioactivity exposure to the workers and general public during decontamination and dismantlement.

3.2 MONITORING AND SURVEILLANCE

3.2.1 IN-PLANT MONITORING

Routine and job specific surveys will be conducted using portable beta-gamma and alpha radiation detection instrumentation. Where significant airborne radioactivity may be generated during work evolutions airborne sampling including the use of continuous air monitors (CAMs) will be utilized.

Samples from radioactive systems, structures and components will be taken and analyzed to assist with developing personnel protective measures and radioactive waste shipping requirements.

3.2.2 ONSITE ENVIRONMENTAL MONITORING

The following monitoring will be maintained through the SAFSTOR decommissioning period:

- Stack continuous monitoring
- Stack particulate filters
- Continuous sampling in discharge canal
- Fenceline dosimetry station monitoring
- Groundwater monitoring

Additionally, in areas where radioactive waste is stored that affects doses in the controlled or unrestricted area, surveys will be performed to demonstrate compliance with member of the public dose limits.

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

3.2.3 OFFSITE ENVIRONMENTAL MONITORING

Environmental monitors will be maintained in accordance with the ODCM and the Quality Assurance Plan.

3.2.4 PERSONNEL MONITORING

While external radiation dose rates are for the most part fairly low as compared to previous light water decommissioning projects, external monitoring with TLDs for all occupationally exposed workers entering the Restricted Area will be required. Internal monitoring will be provided for those individuals deemed likely to exceed 10% of an ALI through a combination of normal lapel sampling and special bioassay based on lapel sample results.

3.3 RADIOACTIVE WASTE PROCESSING AND DISPOSAL

3.3.1 SOURCES OF RADIOACTIVE WASTES

During the SAFSTOR decommissioning period, more wastes will be generated. Spent fuel storage pool water, rain and groundwater in-leakage will be collected and processed as required. Specific dismantlement projects will result in the generation of waste.

3.3.2 LIQUID WASTE PROCESSING AND DISPOSAL

Liquid radioactive wastes may be processed by filtration, and/or demineralization, and/or other appropriate methods when treatment is required. Samples of liquid wastes to be released to the environment are analyzed before release to ensure that they are within the discharge limits specified in the ODCM and 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 in the plant discharge canal. Following Unit 1 and Unit 2 shutdown, liquid radioactive waste discharge will be directly to the discharge canal utilizing tidal dilution flow in accordance with the ODCM.

The expected sources of liquid radwaste from Unit 3 include: spent fuel pool water, 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 radiologically controlled areas. Treatment, sampling, and discharge control will insure that ODCM and 10CFR20 limits are met at the point of discharge to the environment.

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. 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 D&D activities may be solidified for disposal or treated with other liquid radioactive waste.

3.3.3 SOLID WASTE PROCESSING AND DISPOSAL

During D & D 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.

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.

Activated components and spent cartridge-type filters (and filtered crud) will be characterized, processed, and packaged in appropriate shipping containers for shipment to disposal sites, shipment to storage sites (for class B or C waste), or for greater than class C waste package for on site storage in the ISFSI.

Dry active wastes (DAW) includes contaminated protective clothing, plastic, rags, dismantled piping and equipment, contaminated soil, concrete rubble, etc. DAW is characterized, processed, and packaged in appropriate shipping containers for shipment to appropriate approved disposal site(s). On site storage and shipment of packaged waste will be controlled to maintain doses to workers and members of the public ALARA.

Characterization of waste will be accomplished 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 radionuclide or dose rate measurement. Waste classification will be in accordance with 10 CFR Part 61, the disposal site license, and any other regulatory requirements in effect at the time. Other regulatory guidance, such as NRC Branch Technical Positions, 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. Combinations of truck shipments with

transloading to long haul trucks or rail may also be considered for low level waste shipment. For certain larger components, alternative shipment methods may be considered. 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 HEALTH PHYSICS

During the SAFSTOR and decommissioning period, radiation protection and health physics programs will be provided to ensure the health and safety of workers on site. 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).

3.4.1 ORGANIZATION AND RESPONSIBILITIES

The organization described below is the organization, as it exists during the SAFSTOR and decommissioning period. The organization will be changed during decontamination and dismantlement as staffing levels or work requirements dictate.

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 the 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.

Chemistry and Radiation Protection Technicians (C&RP Techs) are the employees, augmented with contract radiation protection technicians as work dictates, 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.

Plant staff qualifications are discussed in section 4.1 of the DSAR.

3.4.2 ALARA PROGRAM

It is the policy of PG&E to design, operate, maintain, modify, and dismantle its nuclear power plants in such a manner as to maintain personnel's 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
- Decommissioning – procedures, license compliance, techniques, equipment usage, maintenance, and operating experience feedback from company and industry experience, and planning.
- 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.

3.4.3 AIRBORNE CONTROL PROGRAM

Due to the internal hazard risk to workers of transuranic contamination, administrative and engineering controls will be utilized to limit airborne radioactivity exposure to the workers and general public during decontamination and dismantlement. It should be noted that due to the lower external dose rates and the potential for substantial internal dose from transuranic contamination, much greater use of respiratory protection will be justified than is usually prescribed for typical operating light water reactor maintenance and refueling work. The following types of controls will be utilized to maintain internal doses to workers and members of the public ALARA.

- Radiation Work Permits/Special Work Permits (RWPs/SWPs) that prescribe specific controls to be utilized
- Surface contaminate fixatives
- Limitations on "hot" cutting of contaminated equipment and piping
- Use of glove bags
- Use of containments
- Local HEPA ventilation
- HEPA vacuums to control loose alpha contamination
- Foaming of contaminated piping prior to cutting
- Continuous air monitors with alpha detection and alarm capability
- Lapel air monitoring of potentially exposed workers
- Use of respiratory protection equipment
- Sealing of cut ends of contaminated piping
- Decontamination of tools and equipment
- HEPA filtration of plant discharge
- Continuous monitoring (with alarm capability) for alpha particulate activity in the plant airborne effluent
- Limitations on the amount of activity in locally used HEPA ventilation equipment and HEPA vacuums
- Periodic testing of HEPA filters and post maintenance testing of HEPA filters

3.4.4 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. Individuals who, in the course of their employment or visit, are not likely to receive in excess of 100 mrem TEDE in one year at HBPP are considered members of the public. Visitors may be tour participants (members of the public), unmonitored workers, or offsite emergency response personnel. Visitors will receive radiological information as necessary.

Members of the Radiation Protection Department 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 and decommissioning period is an extension of the program that was in effect during operation of Unit 3, augmented to support the substantial decontamination and dismantlement activities. 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 items are controlled by plant procedures:

- Radiological work control
- Personnel monitoring
- Monitoring and control of airborne radioactivity
- Respiratory protection program
- Control of access
- Facilities monitoring
- Radiation protection equipment and instrumentation
- Protective clothing requirements
- Radiation Protection records

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3.	Diesel day tanks	19,800	Tanks	401
4.	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.

4.0 ORGANIZATION AND ADMINISTRATIVE CONTROLS

4.1 PLANT STAFF ORGANIZATION AND RESPONSIBILITIES

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

The minimum qualifications for members of the plant staff are evaluated in accordance with plant procedures. An individual may be assigned to a position without meeting the requirements of that position 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.

The Decommissioning Manager is responsible for plant dismantlement activities and preparing the site for termination of the Part 50 License.

4.2 ADMINISTRATION AND CONTROL

4.2.1 TRAINING PROGRAM

4.2.1.1 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. Personnel classified as radiation workers will also receive radiation worker training. 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.

Visitors will receive radiological protection information as necessary.

Training programs include those required by the Emergency Plan, Administrative requirements, and applicable state and federal regulations.

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.

4.2.2 Quality Assurance Program

Decommissioning and SAFSTOR activities will be performed in accordance with the QA Program. The QA Program is designed to ensure that decommissioning activities and activities during the SAFSTOR and decommissioning 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.

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 intervals 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 or modification to these systems that could impair their operation.

5.1 FIRE LOSS PREVENTION PROGRAM

HBAP A-13, "Fire Loss Prevention Program" and the Technical Basis Document TBD-301, "Fire Hazards Analysis" together provide the information and references necessary including system checks, equipment description, systems description, administrative controls, personnel training, and fire response per NRC Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants during Decommissioning and Permanent Shutdown."

5.1.1 Plant Fire Protection Features

The fire protection features are described in TBD-301, "Fire Hazards Analysis." Specific references are provided that cover all protection features for HBPP.

5.1.2 Fire Loss Prevention Program Responsibilities

Responsibilities are clearly explained in HBAP A-13, "Fire Loss Prevention Program." These responsibilities clearly provide the organizational requirements described in NRC Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants during Decommissioning and Permanent Shutdown."

5.1.3 Fire Protection Training

HBAP B-13, "Qualification and Training Requirements of Fire Loss Prevention Personnel" describes the training requirements for the HBPP Fire Loss Prevention Program consistent with the requirements of NRC Regulatory Guide 1.191, "Fire Protection Program for Nuclear Power Plants during Decommissioning and Permanent Shutdown" and Cal-OSHA Title 8, Article 157, Section 6151(g).

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 monitored and analyzed per plant procedures.

5.3 SERVICE SYSTEMS

5.3.1 Refueling Building Ventilation System

In the interest of ALARA, the refueling building ventilation system will normally be operated to maintain a negative pressure with the exception of times when major openings are required for equipment ingress or egress.

The capability of the refueling building ventilation system to maintain a negative pressure in the refueling building shall be tested once each quarter.

5.3.2 Spent Fuel Storage Pool Service Systems

A minimum amount of water shall be maintained as specified in plant procedures.

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. If low voltage is detected, the supply is automatically transferred to a 480 volt ac source from Unit 1 or 2. The emergency section shall supply the following loads:

- Emergency lighting
- Main annunciator system
- The following radiation monitoring systems: stack gas sampling and liquid effluent monitoring

During the above transfers, and subsequently should those sources be unavailable, the main annunciator and the radiation monitoring (stack gas sampling and liquid effluent monitoring systems) are provided with battery backup from either Unit 1 or Unit 2 Station Batteries and emergency lighting is provided by battery operated lights.

The transfer of the emergency 480 V AC and battery backup system shall be tested for proper operation at least annually with loads connected to simulate emergency operation.

5.4 MONITORING SYSTEMS

5.4.1 Portable Monitoring Equipment

During planned evolutions which are expected to increase radiation levels, monitoring shall be accomplished with portable instruments whenever personnel are in the refueling building.

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

Level indication of the spent fuel storage pool water level shall be monitored per plant procedures.

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, in accordance with plant procedures.

APPENDIX A

Implications of Accidents during SAFSTOR Decommissioning

The Spent Fuel Pool (SFP) will remain in service until it has been determined it is no longer required during the SAFSTOR Decommissioning period. The large volume of water in the pool provides containment for the radioactive contamination on the various pool surfaces and can be used for shielding of radioactive waste generated during Unit 3 dismantlement. Releases of radioactive materials will be minimized by containment of the spent fuel pool water and removal of radioactive contaminants from the water itself. The purity of the water will be maintained to prevent pool corrosion and to limit radioactive material concentrations. pH and chemical contaminant levels will be maintained in ranges where corrosive attack is minimized to protect against release of radioactivity. Maintaining radioactive nuclide concentrations in the pool water ALARA will reduce radiation levels in the pool vicinity; in the unlikely event of a liner failure, the release of radioactivity to the surrounding groundwater will be minimized.

Early in the operation of Unit 3, SFP leakage was detected, and a stainless steel liner was installed to alleviate the problem. Approximately 50 liters (12 gallons) of water has historically been 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 SFP), is conducted on a periodic basis. ¹³⁷Cs 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.

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

1.1 IDENTIFICATION AND PROBABILITY OF ACCIDENTS DURING SAFSTOR DECOMMISSIONING

The following are considered credible and worthy of assessment for the SAFSTOR Decommissioning period:

- Explosions, delayed ignition of flammable vapor clouds, release of toxic chemicals, or fire

1.1.1 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 DECOMMISSIONING

While accidents have an extremely small probability of occurrence during SAFSTOR Decommissioning, the consequences of the accidents listed in Section 1.1 have been analyzed to determine the potential worst case doses.

1.2.1 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 Decommissioning 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.

APPENDIX B

Criticality Analysis SAFSTOR and DECOMMISSIONING

Post-Removal of Spent Fuel & Fuel Fragments from the Spent Fuel Pool

CONTENTS

<u>Section</u>	<u>Page</u>
A. Criticality Analysis	B-1
B. References	B-1

The transfer of all spent fuel and fuel fragments from the Spent Fuel Pool to the Independent Spent Fuel Storage Installation (ISFSI) was completed on December 11, 2008. Decommissioning efforts will eventually progress to opening of the reactor vessel. Because of fuel failures during reactor operation, the possibility of fuel fragments residing on or below the core support assembly can not be discounted. In the event fragments are present, significant margin to criticality would exist, even if water is present, since any distribution of fragments represents a significantly over-moderated condition. Staying within the allowed accumulation (Reference 1) of the equivalent of a single, intact fuel assembly represents a significantly under-moderated condition and, again, significant margin to criticality would exist.

B. REFERENCES

1. TBD-305, Rev. 0, Spent Fuel Pool Fuel Fragment/Debris Evaluation

APPENDIX C

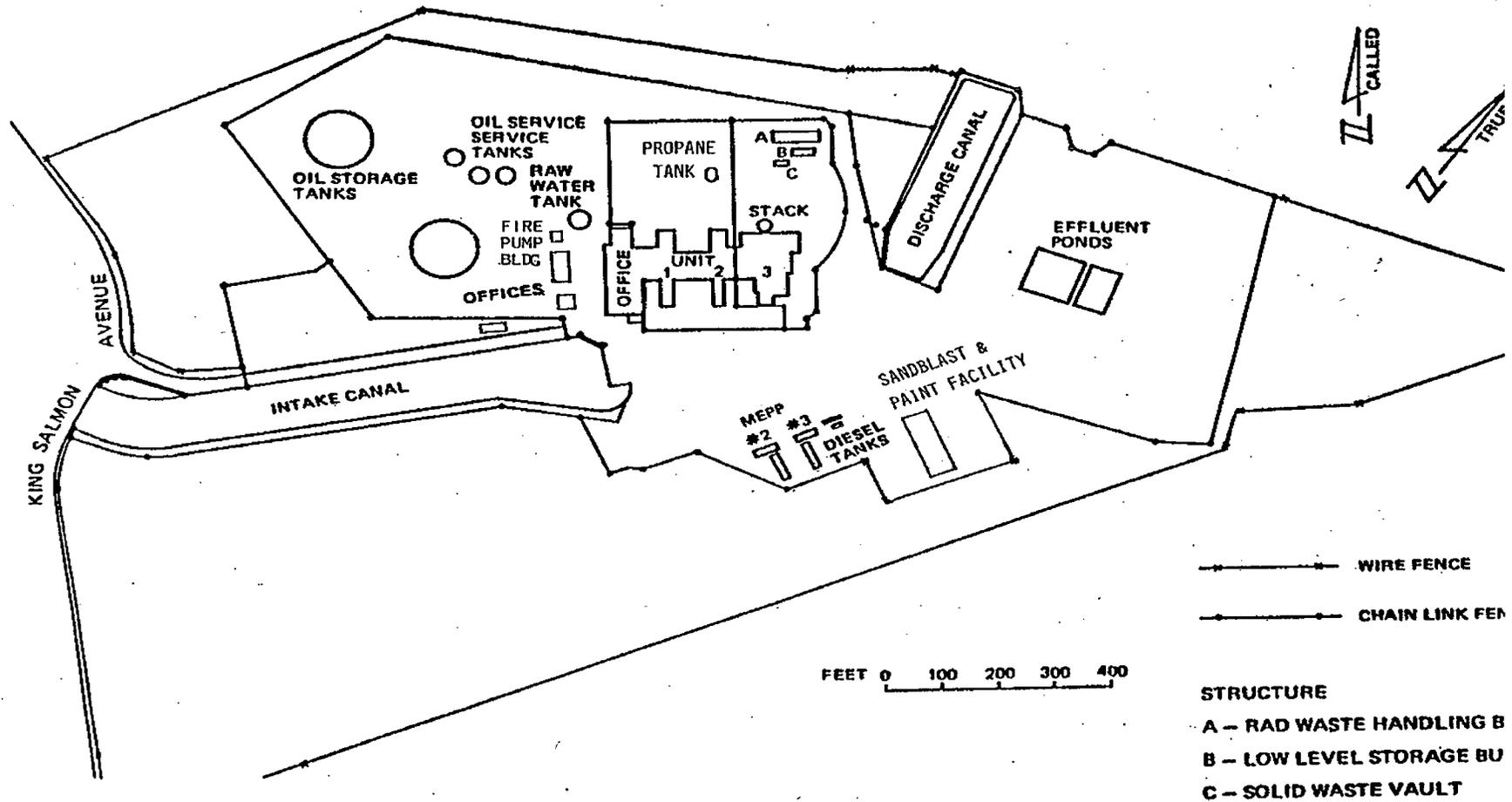
FIGURES

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.

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**FIGURE 2-1
HUMBOLDT BAY POWER PLANT SITE PLAN**



2
CALLED NORTH

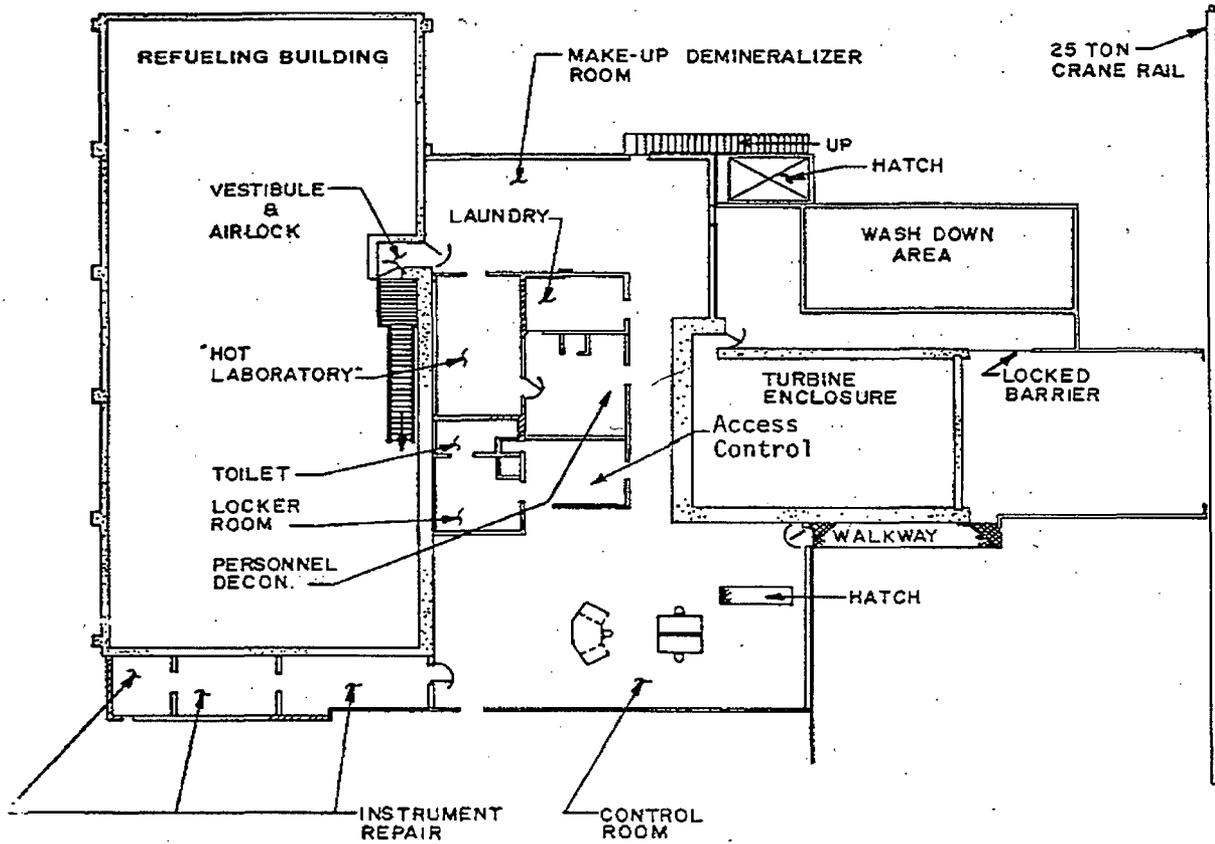


FIGURE 2-2
HUMBOLDT BAY UNIT 3
OPERATING FLOOR PLAN (EL. 27'-0")

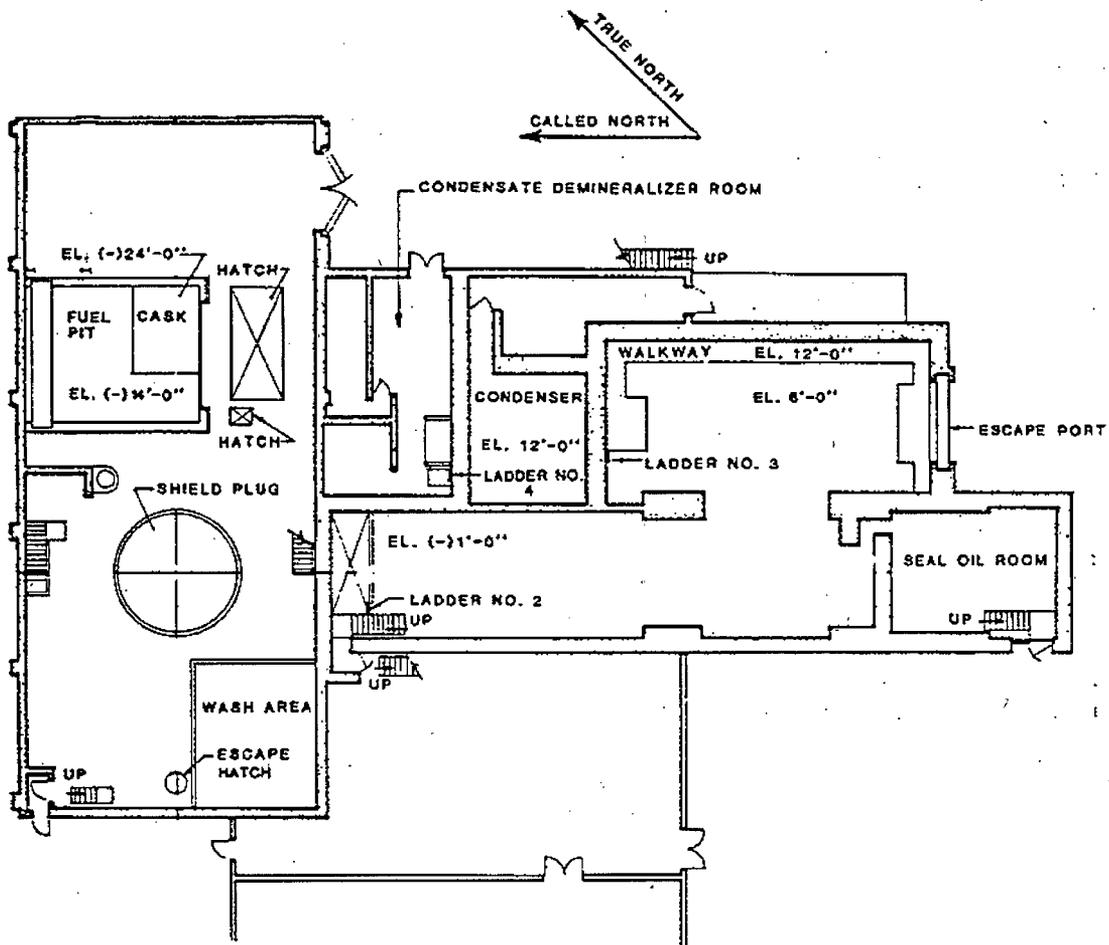


FIGURE 2-3
HBPP UNIT 3 GROUND FLOOR PLAN



CALLED NORTH

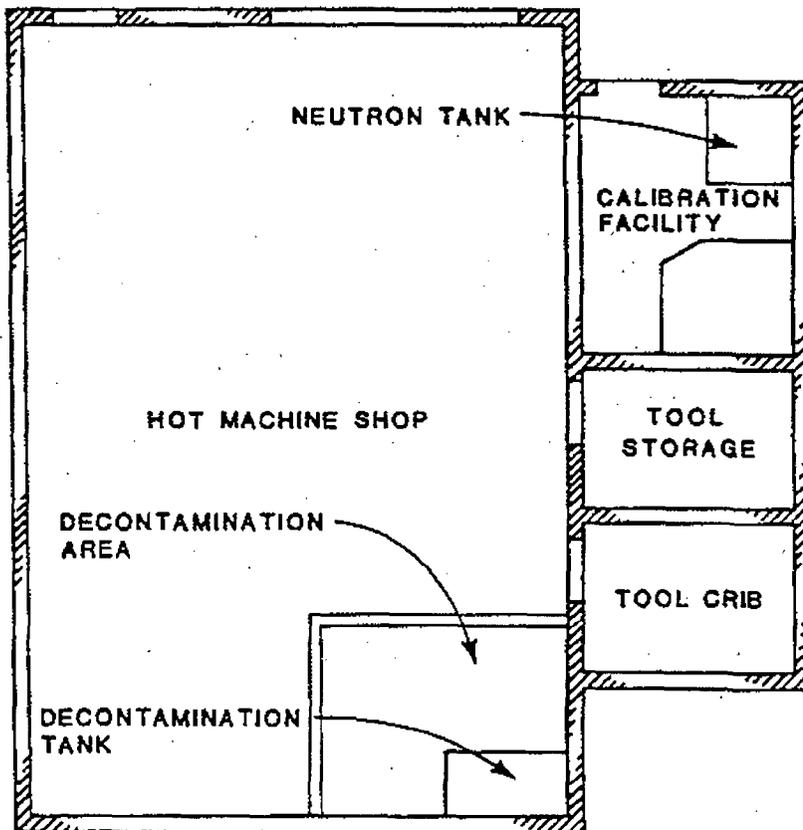
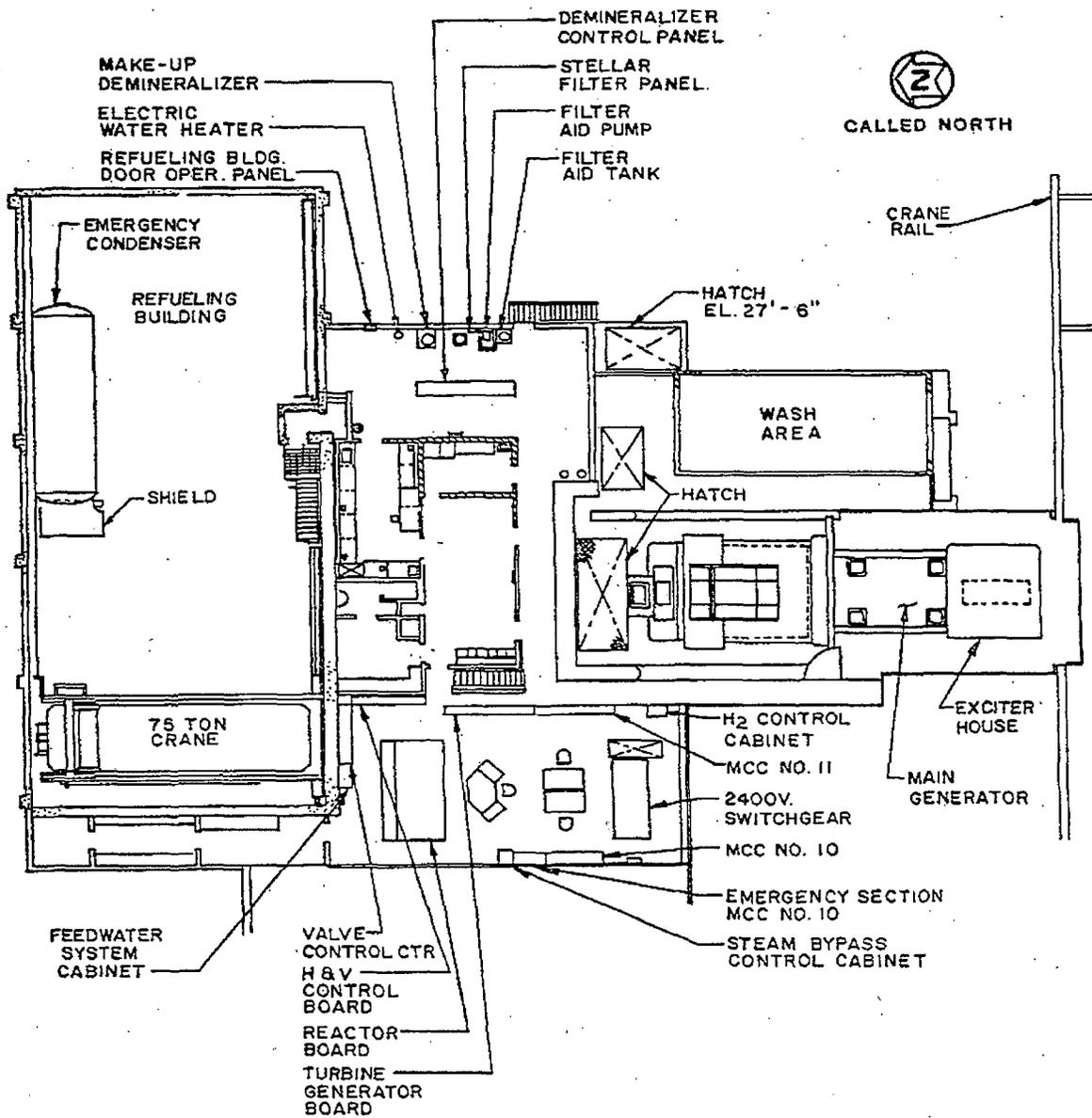
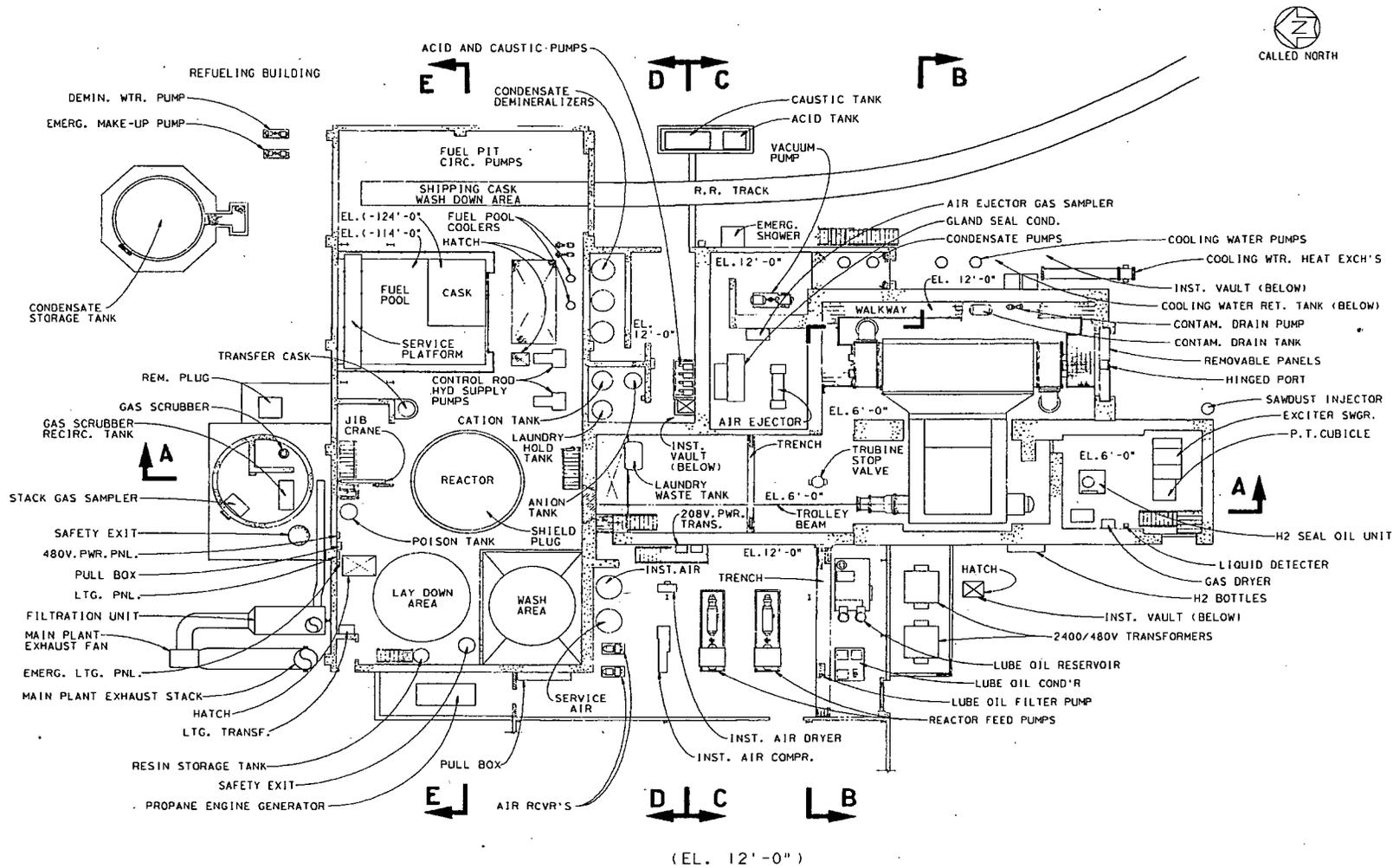


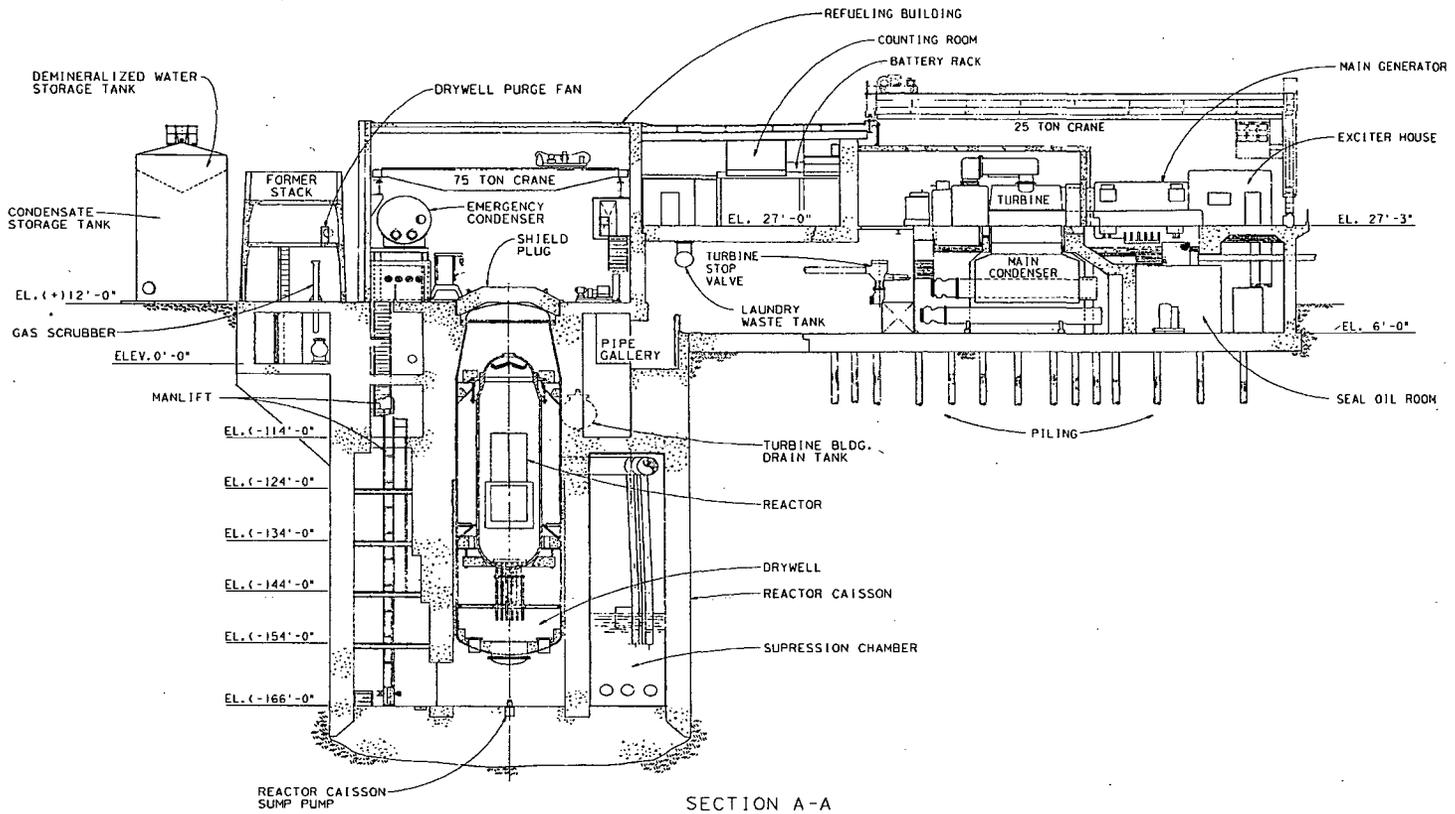
FIGURE 2-4
HOT MACHINE SHOP
AND CALIBRATION FACILITY



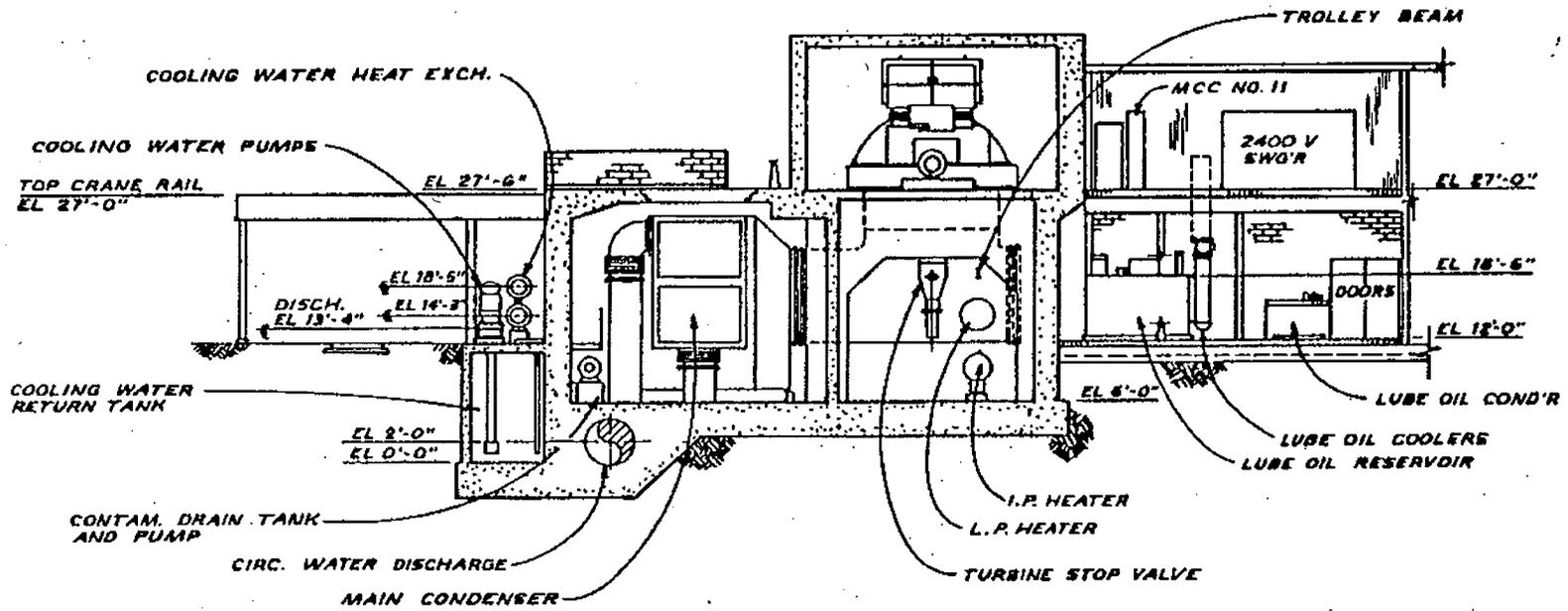
**FIGURE 2-5
EQUIPMENT LOCATION – OPERATING FLOOR PLAN
(EL 27'-0")**



**FIGURE 2-6
EQUIPMENT LOCATION - GROUND FLOOR PLAN**

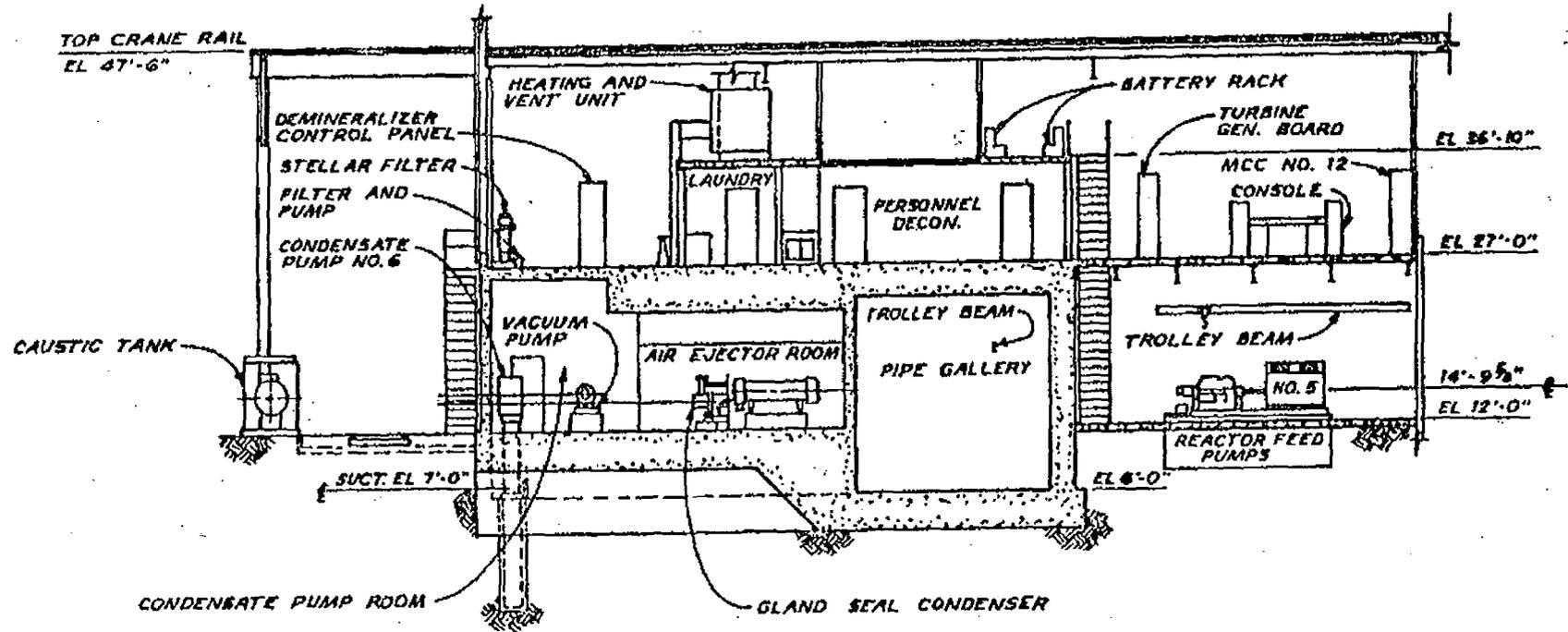


**FIGURE 2-7
EQUIPMENT LOCATIONS**



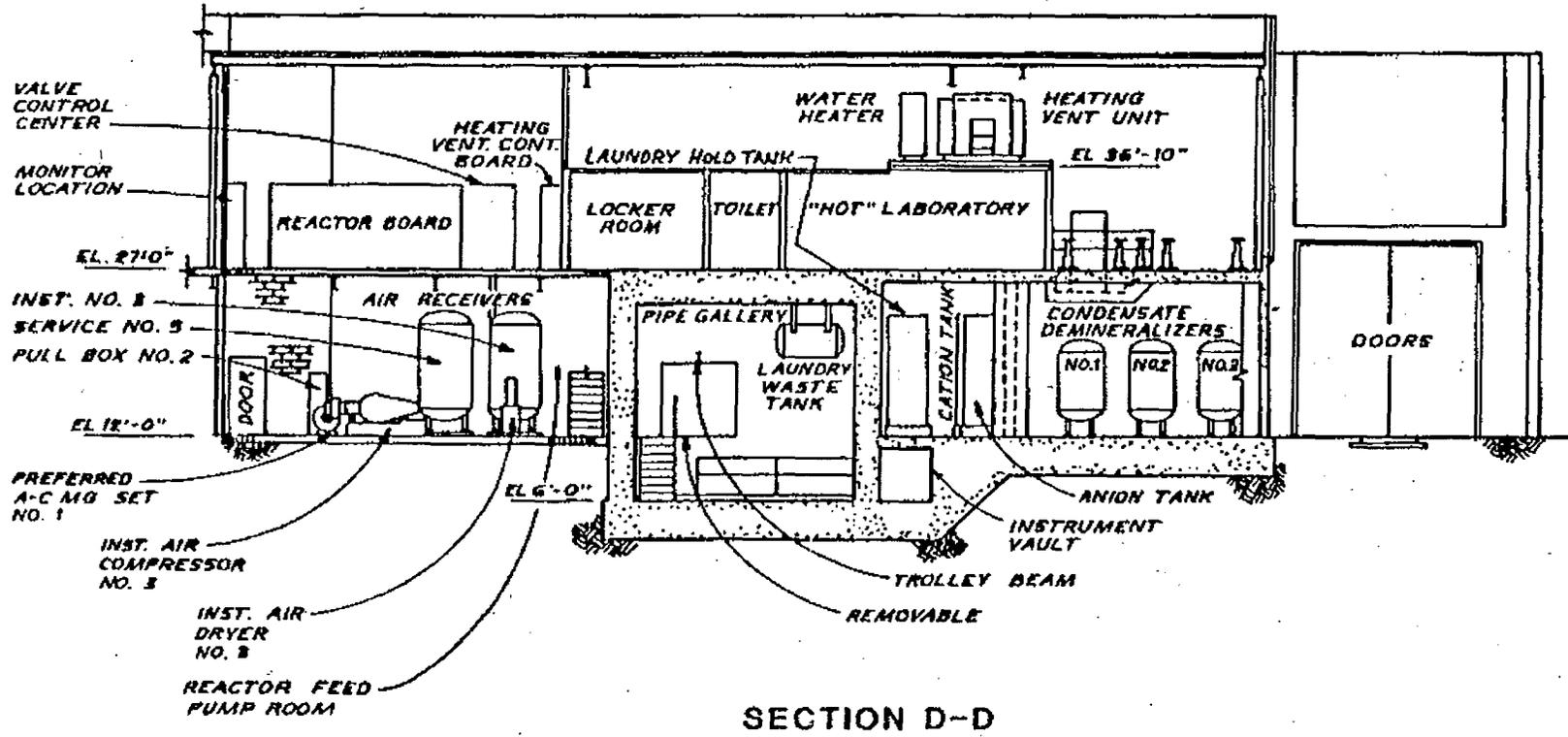
SECTION B-B

FIGURE 2-8
EQUIPMENT LOCATIONS



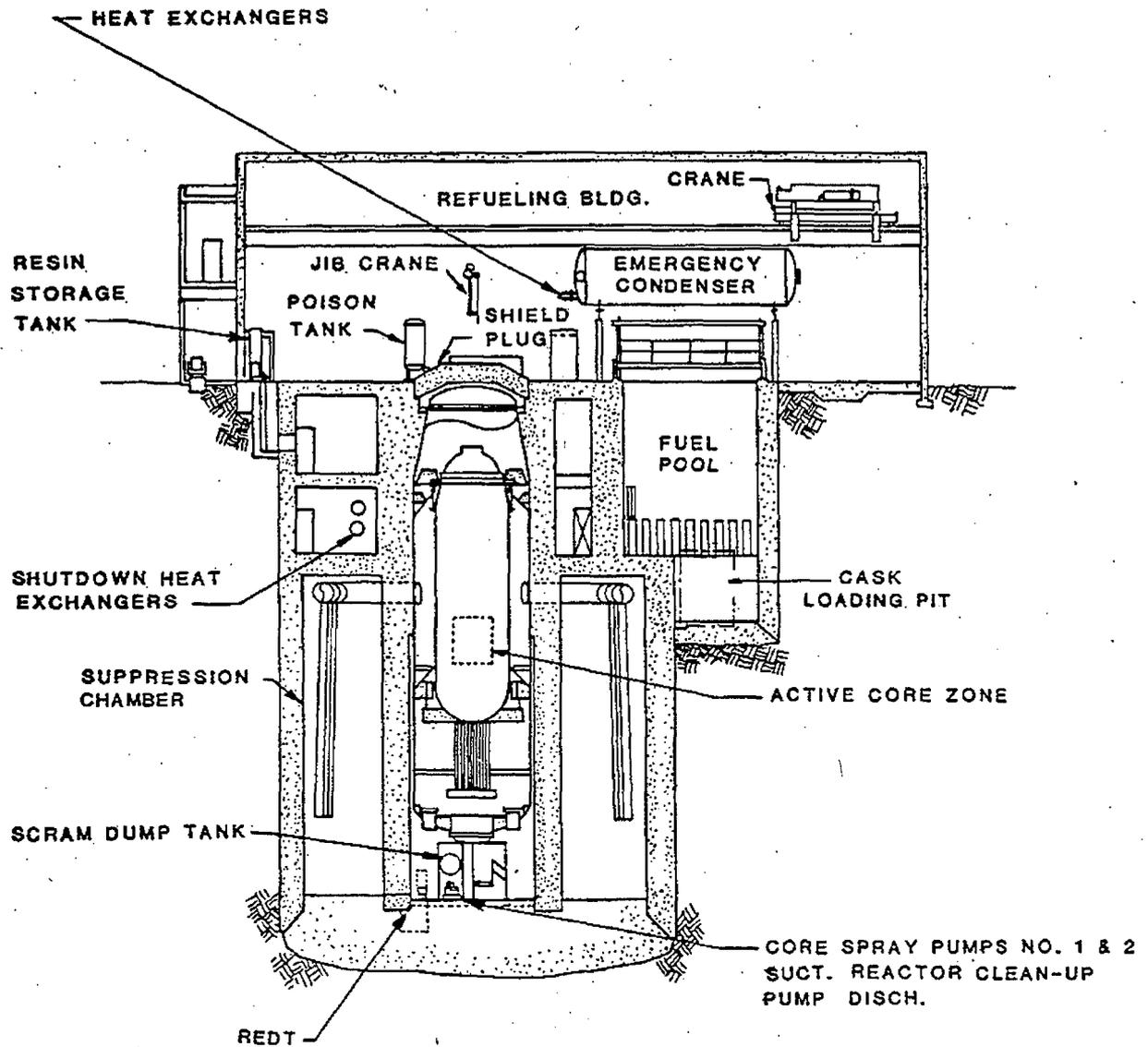
SECTION C-C

FIGURE 2-9
EQUIPMENT LOCATIONS



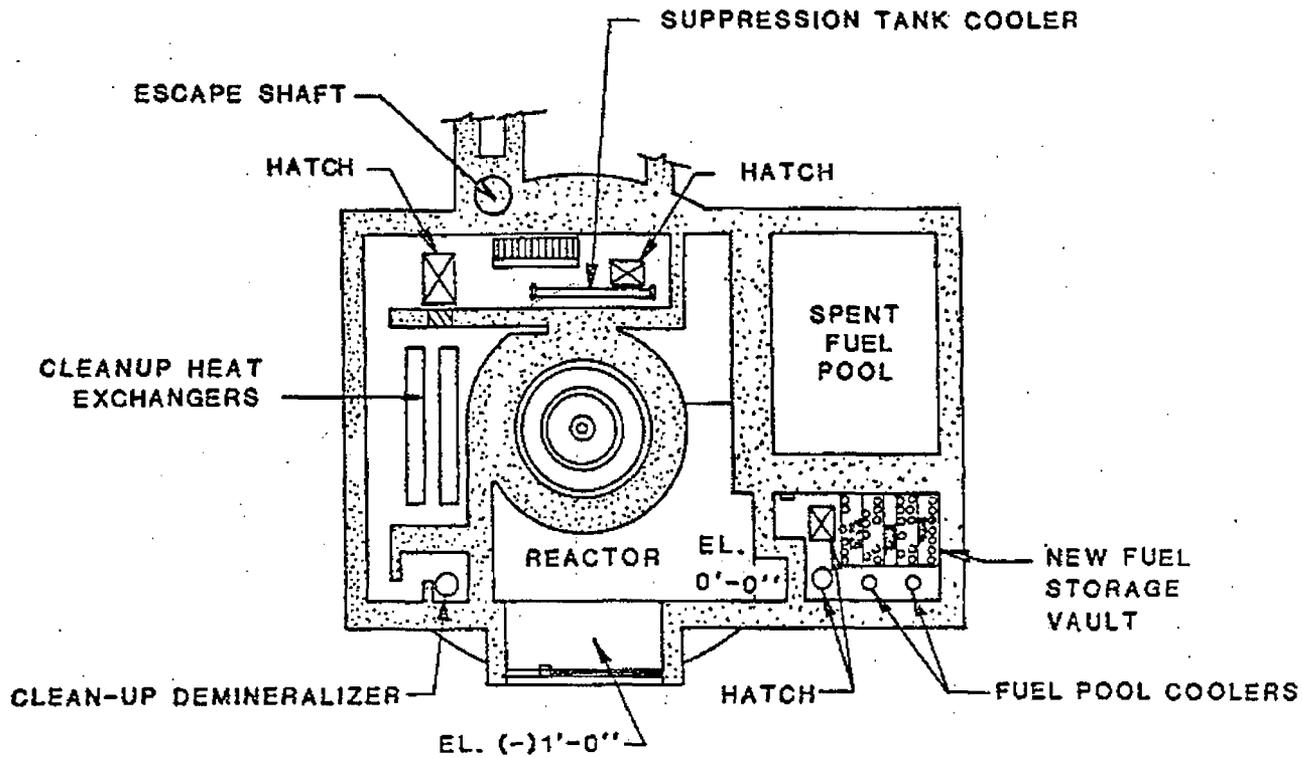
SECTION D-D

FIGURE 2-10
EQUIPMENT LOCATIONS



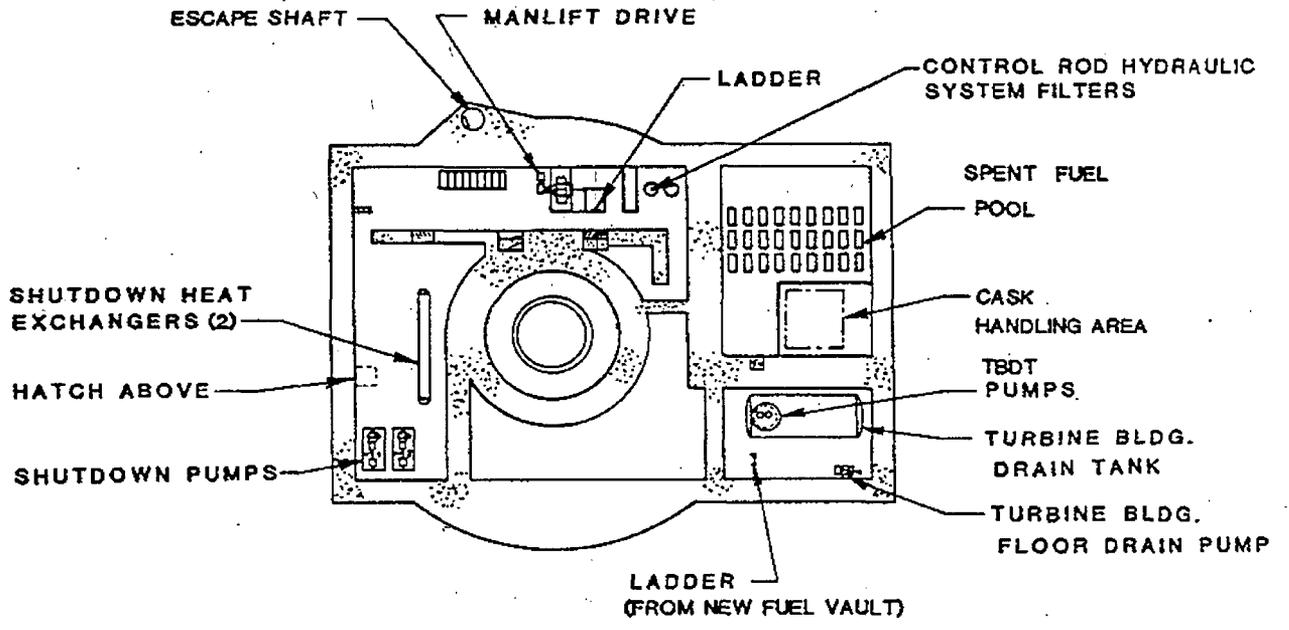
SECTION E-E

FIGURE 2-11
EQUIPMENT LOCATION



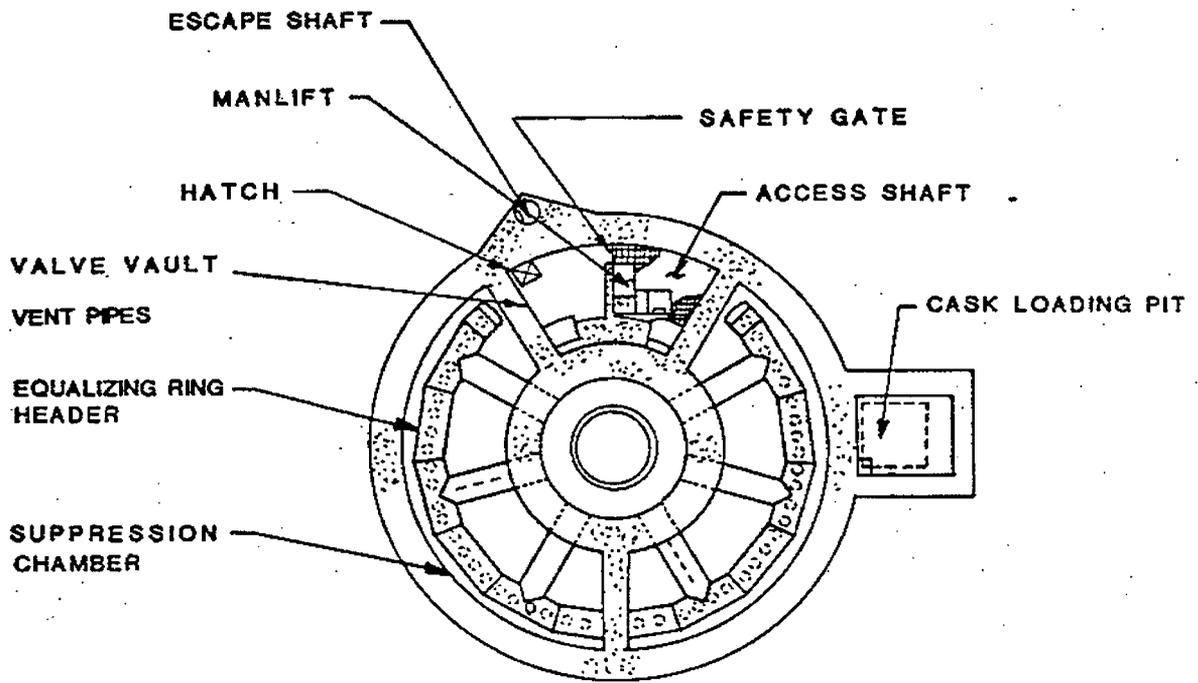
PLAN AT EL. (-) 2'-0"

FIGURE 2-12
EQUIPMENT LOCATION



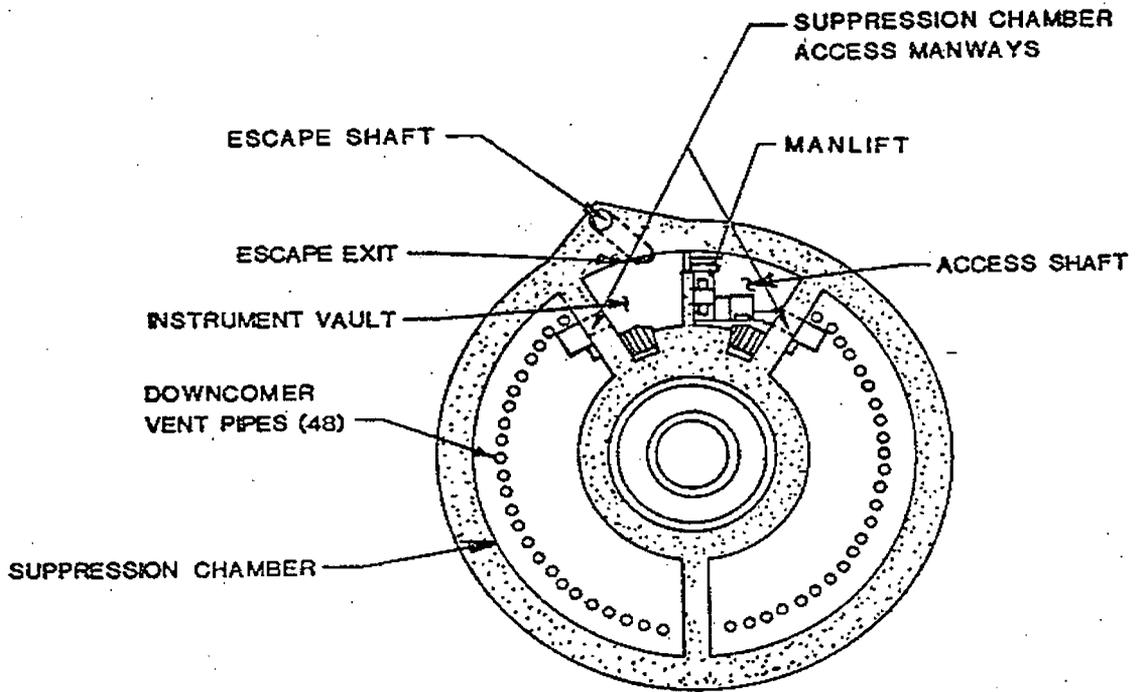
PLAN AT EL. (-) 14'-0"

FIGURE 2-13
EQUIPMENT LOCATION



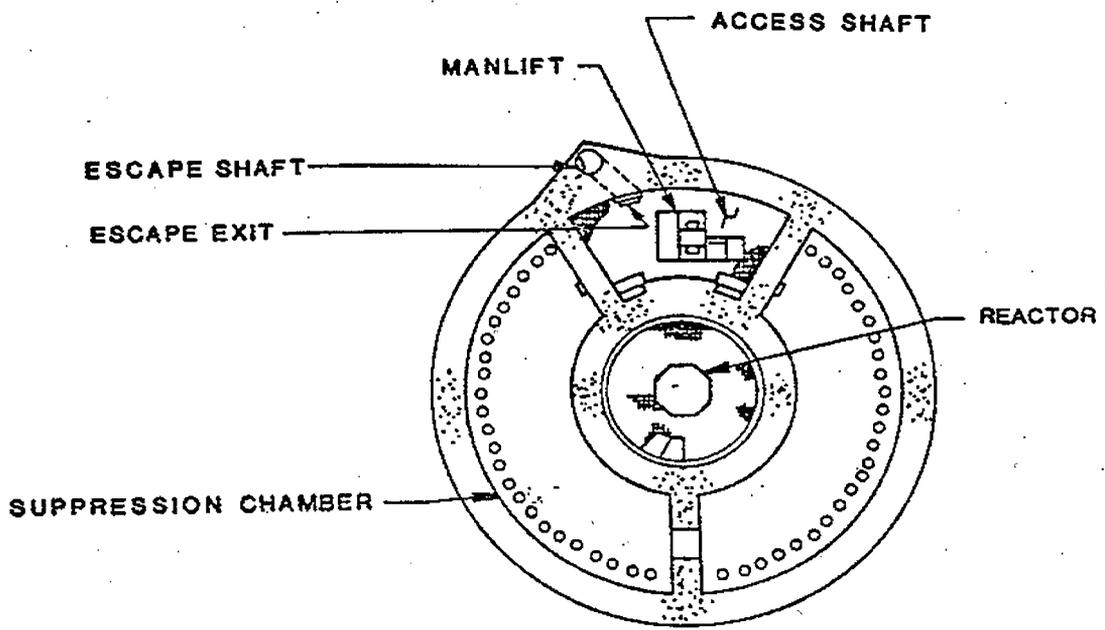
PLAN AT EL. (-)24'-0"

FIGURE 2-14
EQUIPMENT LOCATION



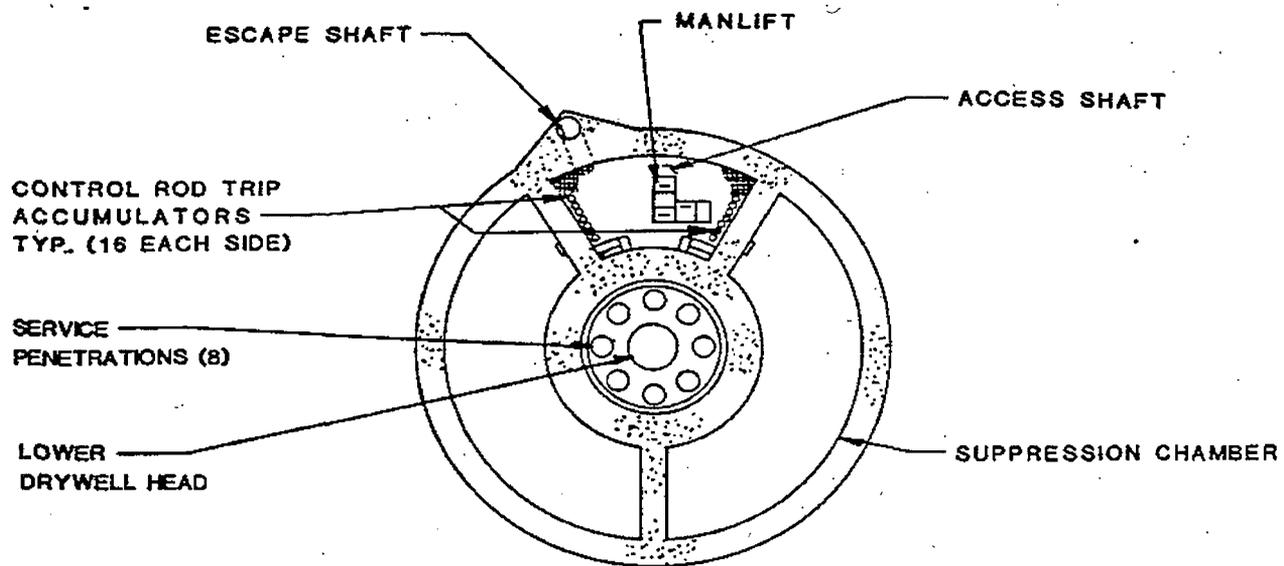
PLAN AT EL. (-)34'-0"

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

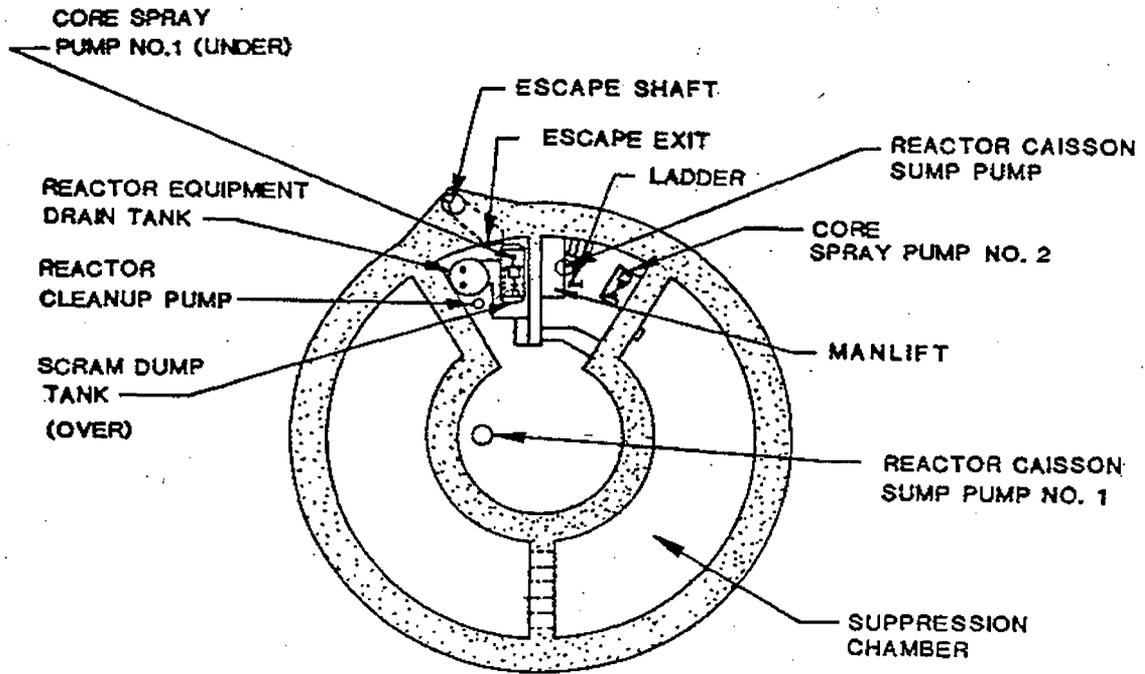
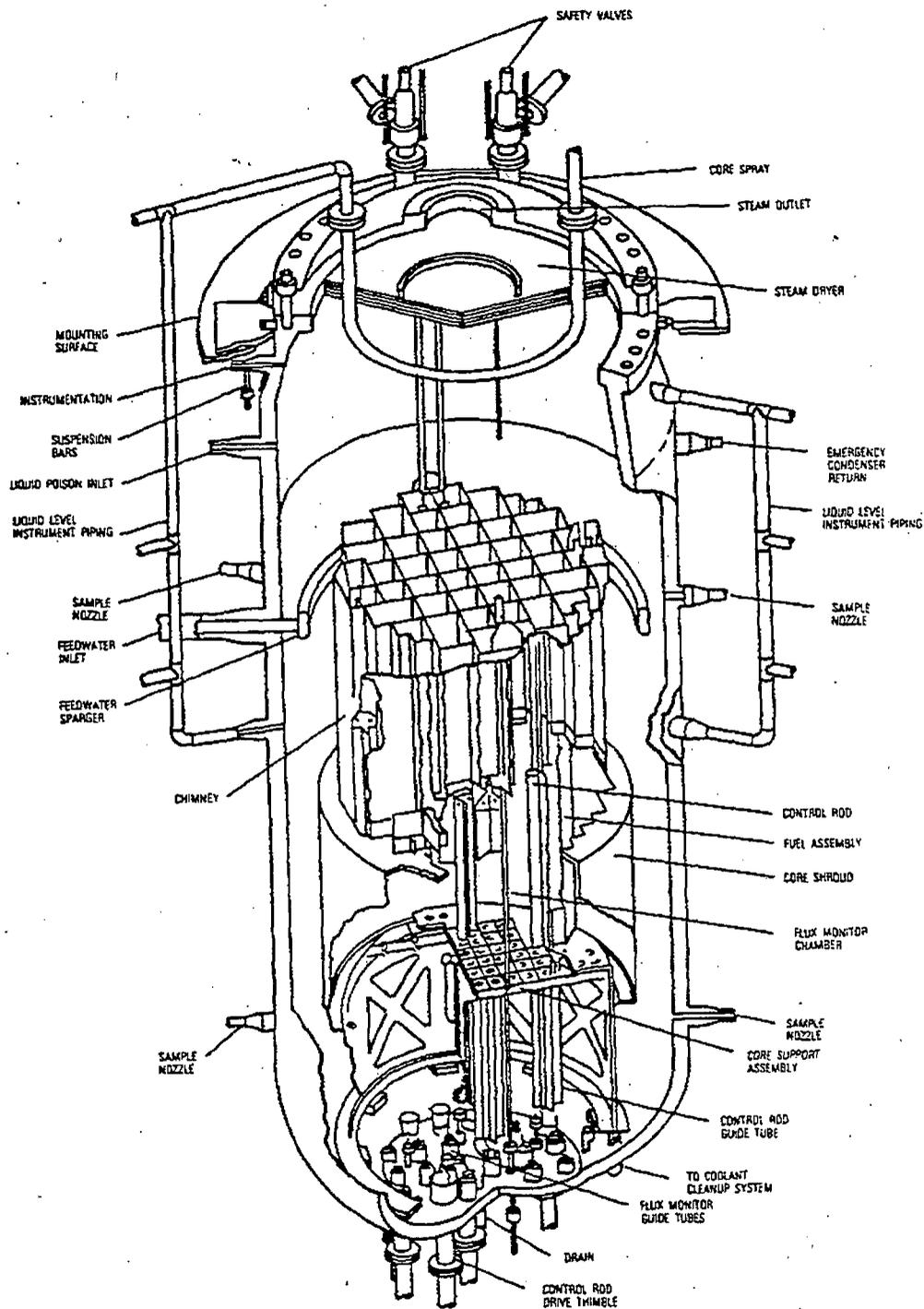


FIGURE 2-18
 EQUIPMENT LOCATION
 PLAN AT EL. (-) 66'-0"



**FIGURE 2-19
SCHEMATIC DIAGRAM OF REACTOR PRESSURE VESSEL
AND INTERNALS FOR HBPP UNIT 3**

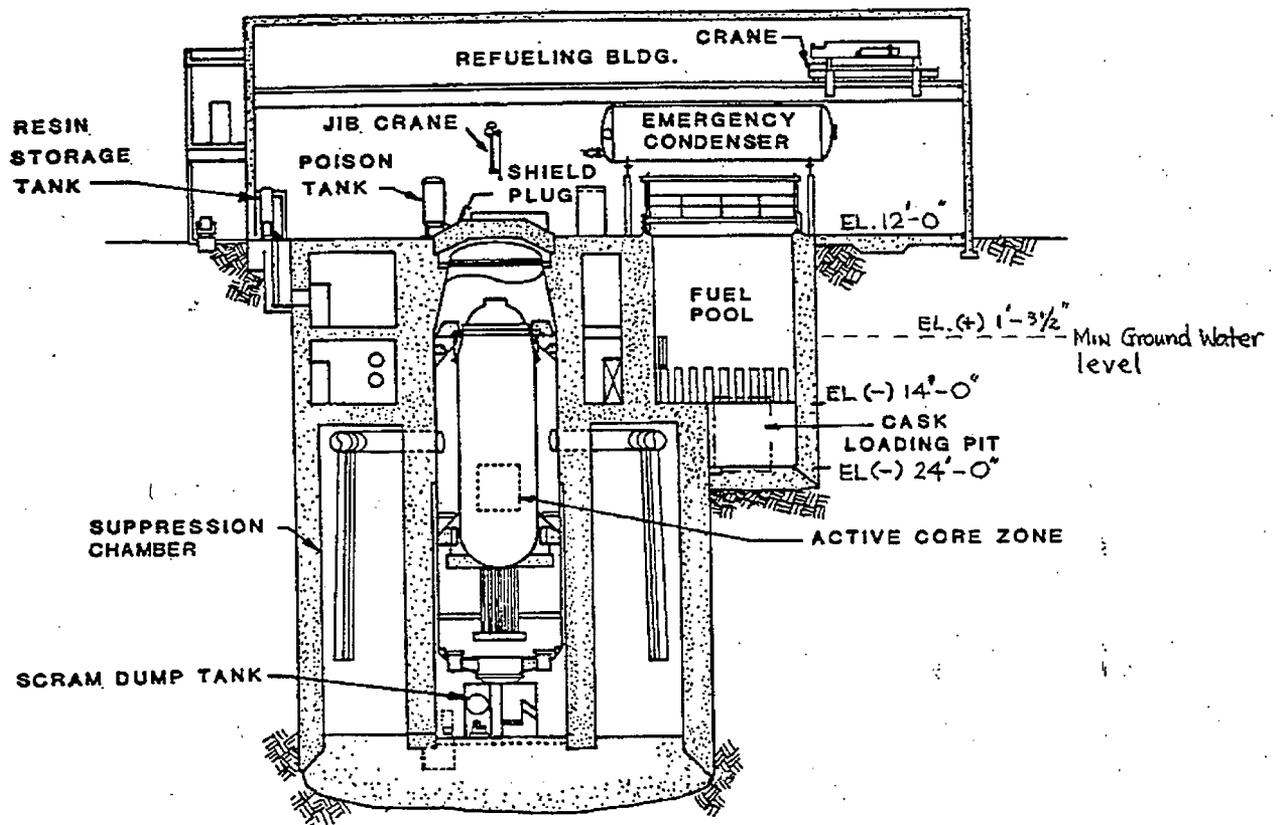


FIGURE 1
HUMBOLDT BAY PLANT SECTIONAL VIEW

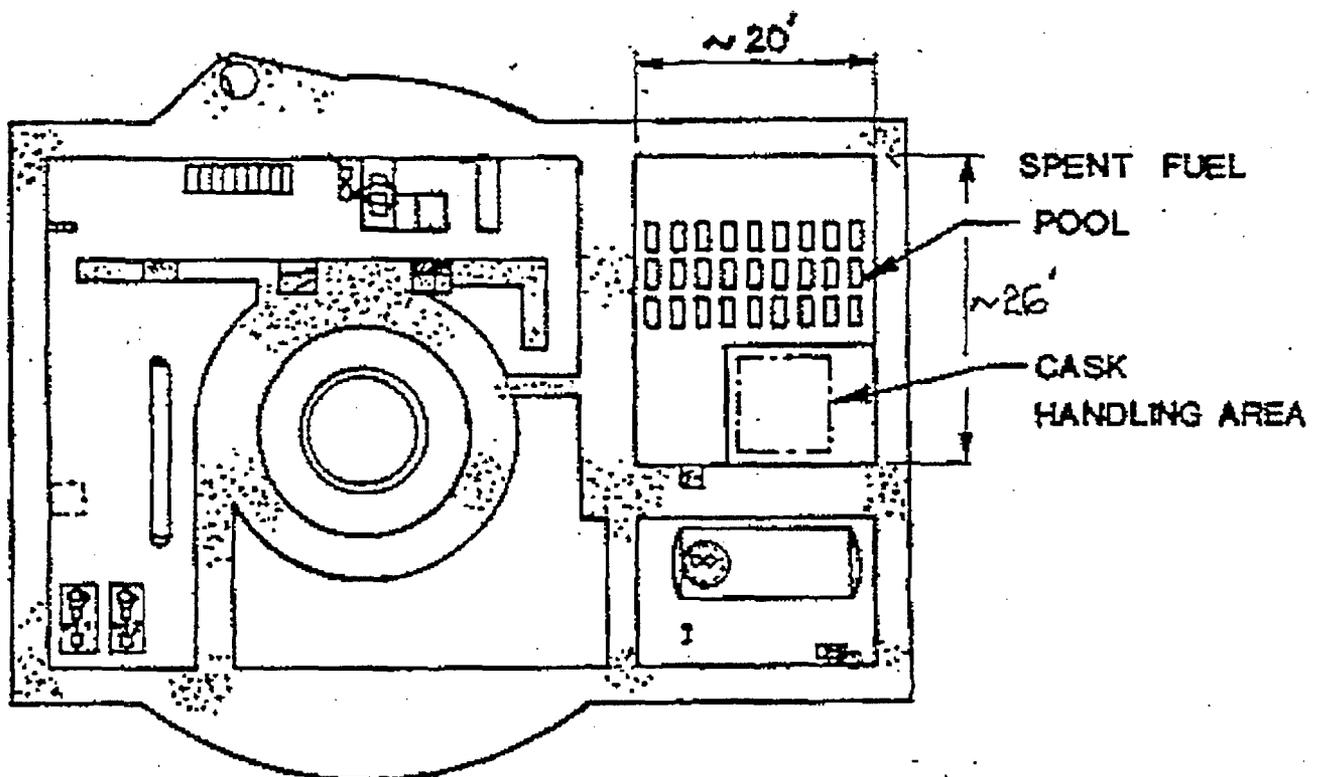


FIGURE 2
HUMBOLDT BAY PLANT PLAN VIEW

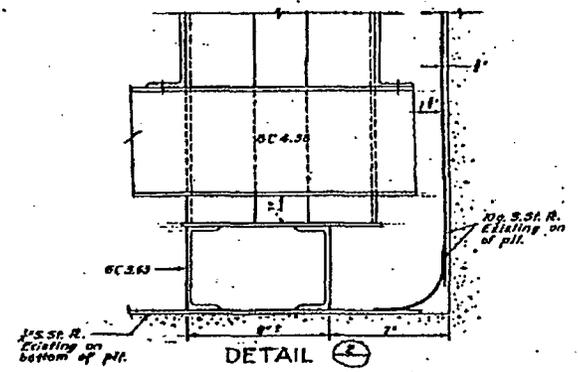
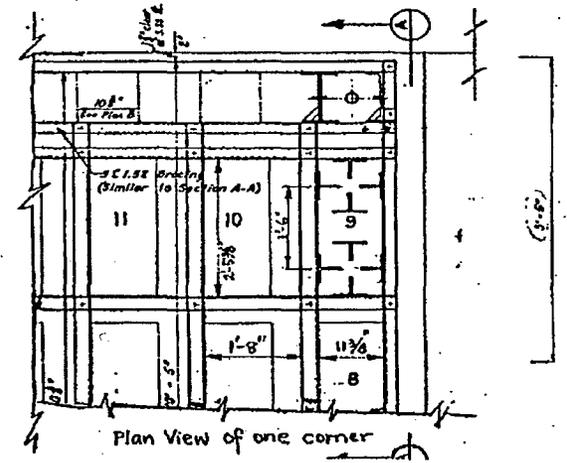
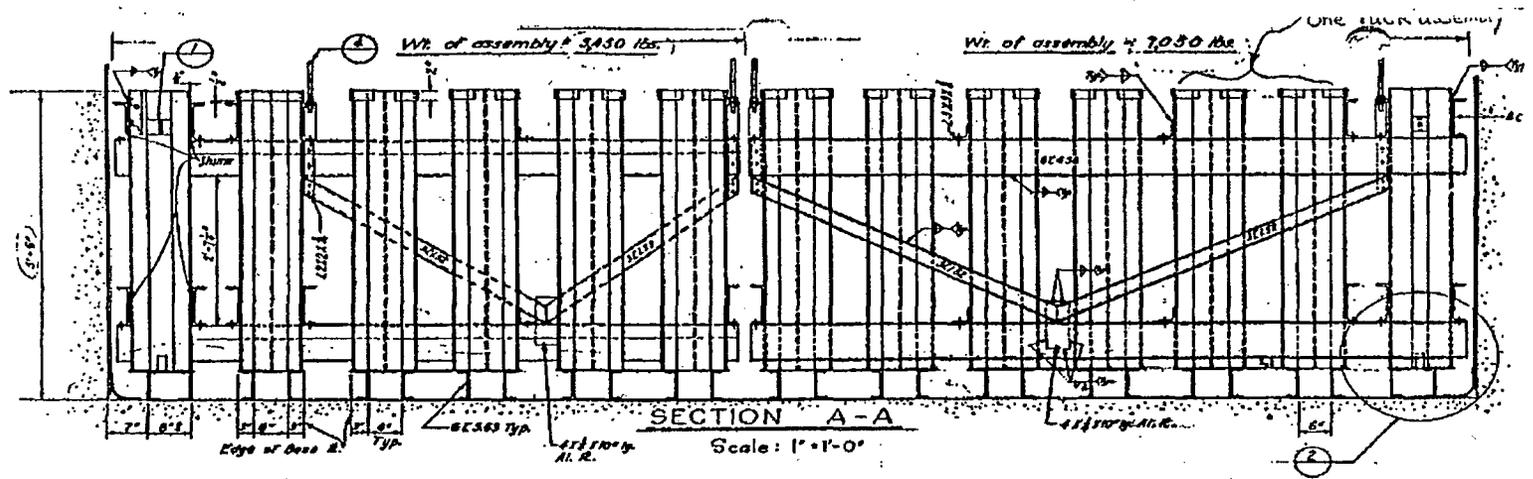
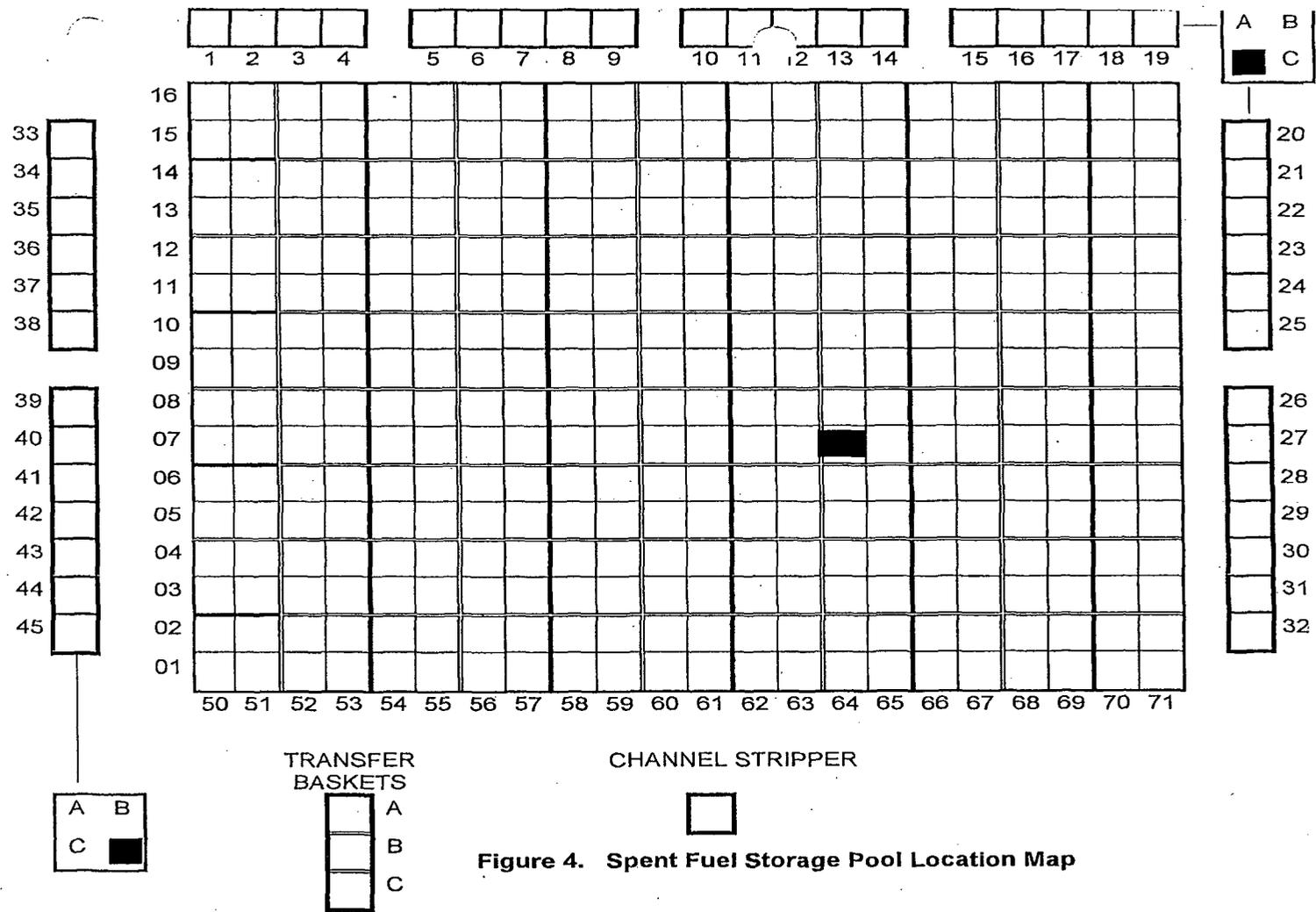


Figure 3. Spent Fuel Rack Arrangement (Typical)



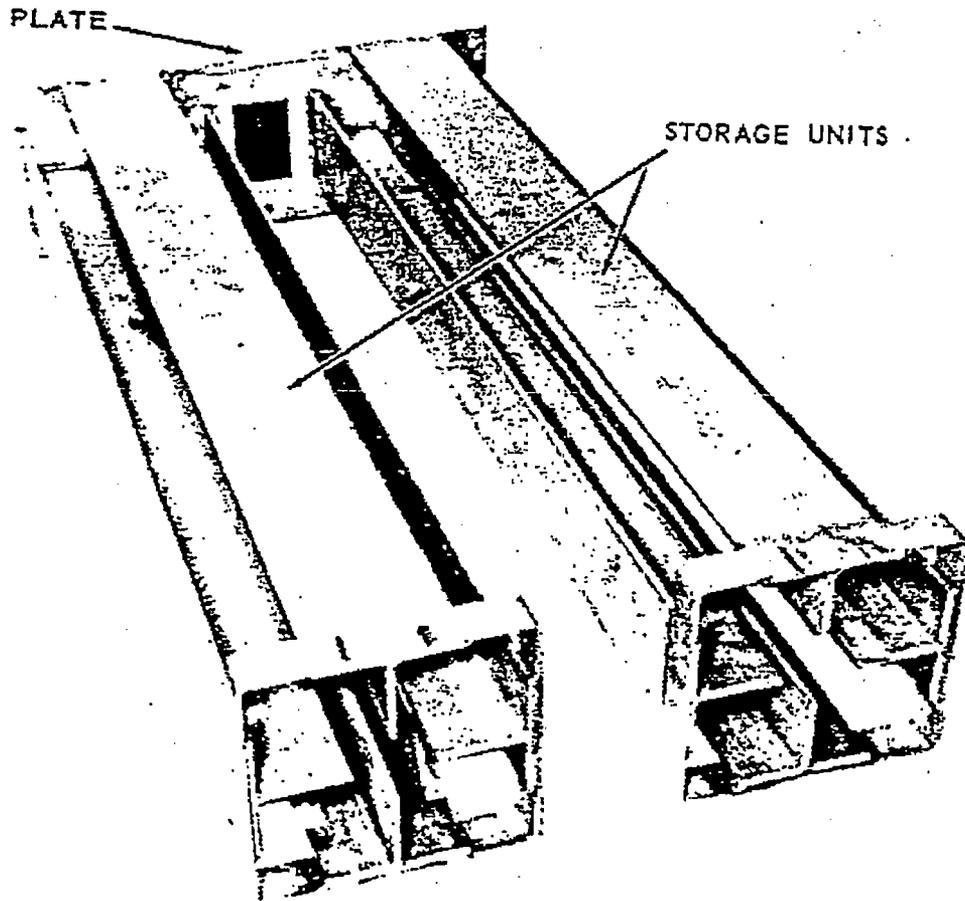


FIGURE 5
HUMBOLDT BAY POWER PLANT STORAGE RACKS

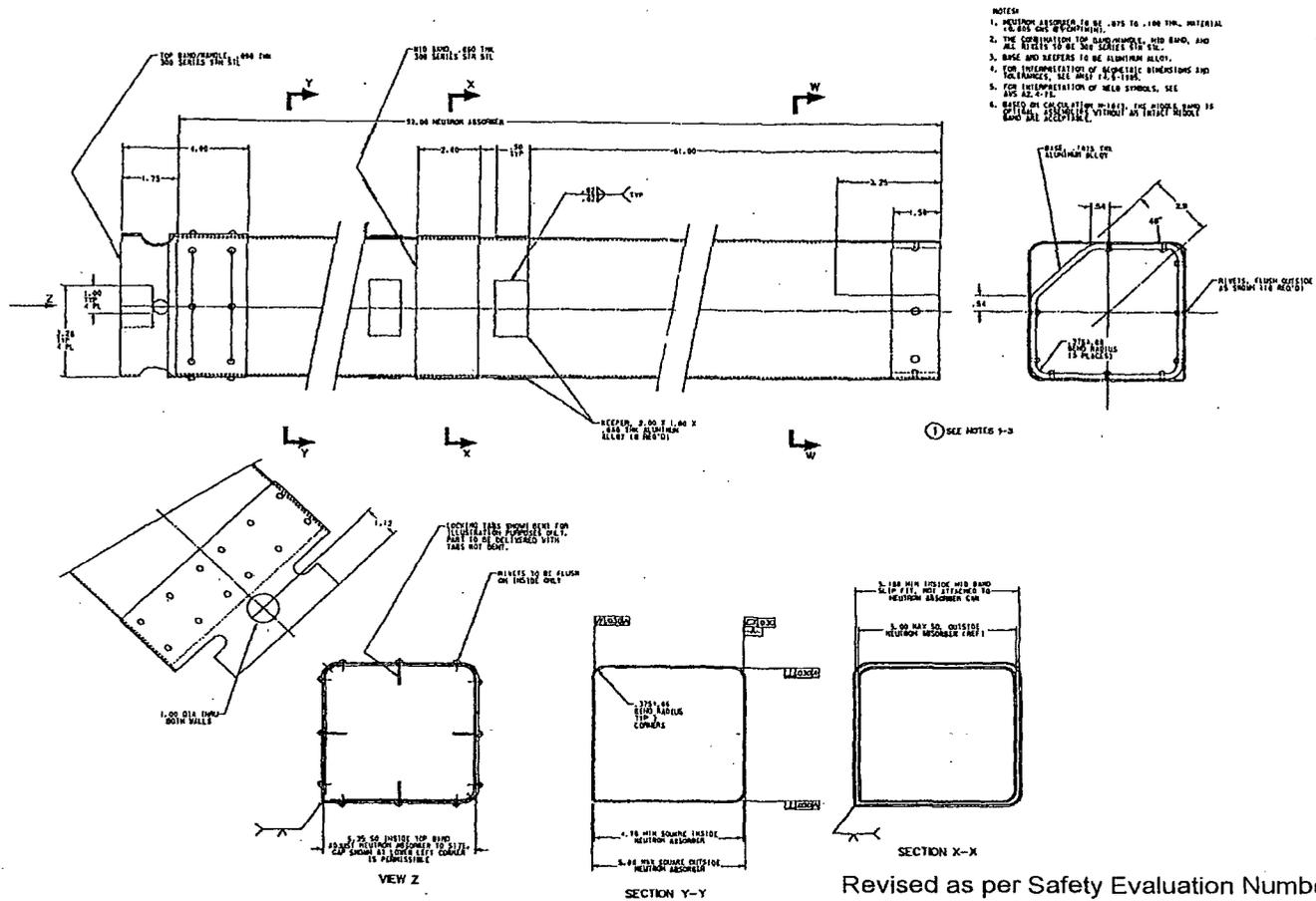
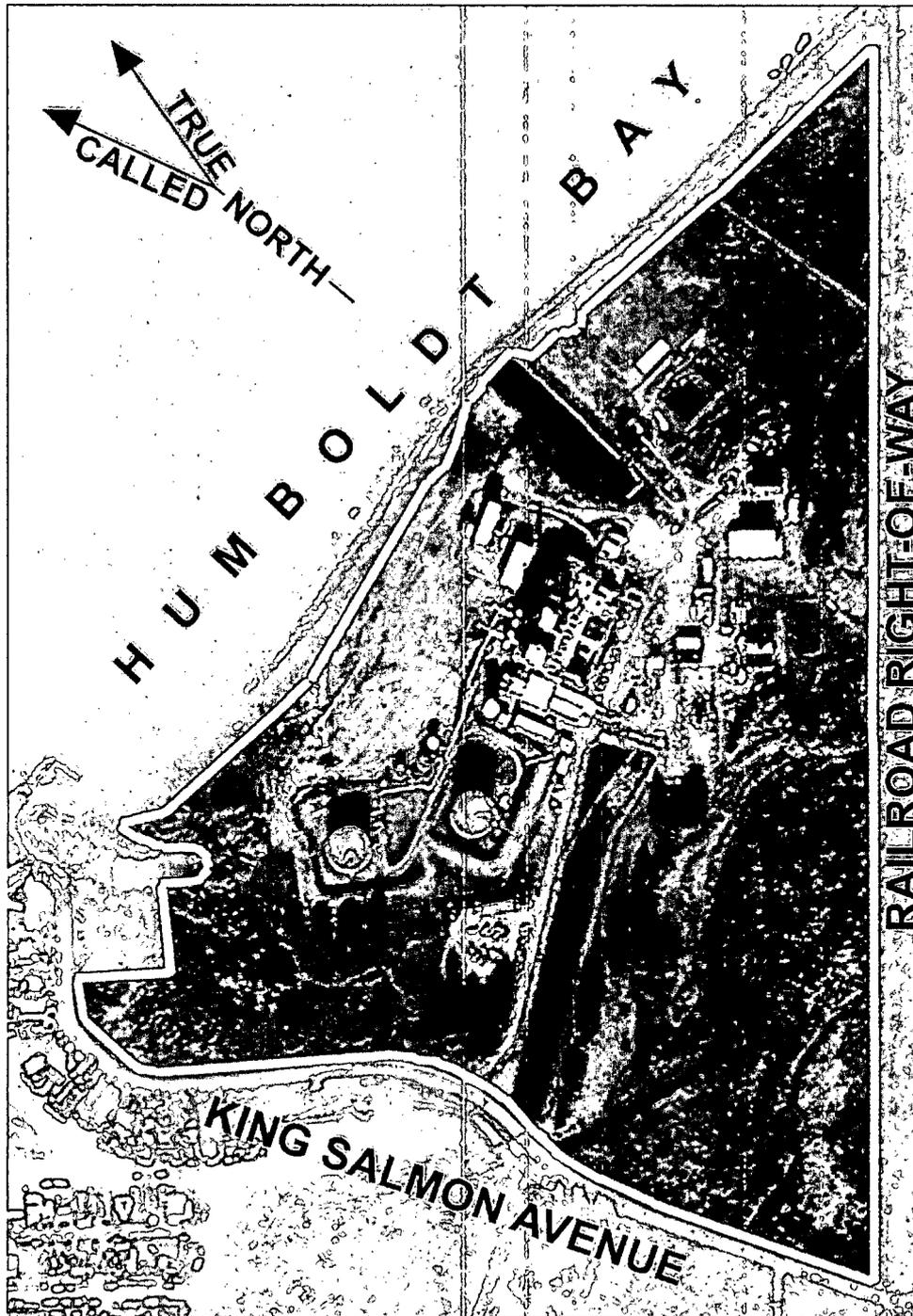


FIGURE 6
FUEL ASSEMBLY PROTECTIVE CAN
 (EXCERPTED FROM 60199924-17)

Revised as per Safety Evaluation Number
 2000-10

Figure 7
Humboldt Bay Power Plant Site Boundary



Note: Licensed material may be received, possessed, or used within the site boundary.

APPENDIX D

IMPLICATIONS OF DECOMMISSIONING ACCIDENTS WITH POTENTIAL FOR RADIOLOGICAL IMPACTS TO THE ENVIRONMENT

CONTENTS

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1.4	Contamination Control Envelope Rupture	D-6
1.5	Oxyacetylene Explosion	D-7
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APPENDIX D

IMPLICATIONS OF DECOMMISSIONING ACCIDENTS WITH POTENTIAL FOR RADIOLOGICAL IMPACTS TO THE ENVIRONMENT

1.0 Introduction and Background

NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," Supplement 1, November 2002 (GEIS), concludes that the environmental impacts of radiological accidents potentially resulting from decommissioning activities are of small significance and that this evaluation of their significance is applicable to all permanently shutdown units, including Humboldt Bay Power Plant (HBPP) Unit 3. Specifically, the GEIS conclusion on the potential impacts of radiological accidents resulting from decommissioning activities states that, "with mitigation procedures in place, the impacts of radiological accidents are neither detectable nor destabilizing. Therefore, the staff makes the generic conclusion that the impacts of non-spent fuel-related radiological accidents are SMALL." For radiological assessments, impacts are of small significance if the Total Effective Dose Equivalent (TEDE) to a member of the public does not exceed the EPA 400 Manual protective action threshold of 1 rem, a small fraction of the limit established in 10 CFR 100.

Postulated accidents have been analyzed for HBPP Unit 3, independently from the NRC staff's evaluation, to address a site specific radiological issue that was not considered in the GEIS. The accidents were selected in accordance with NUREG/CR-0672, "Technology, Safety and Cost of Decommissioning a Reference Boiling Water Reactor."

The following is a brief discussion of the HBPP site specific issue.

At the time that Unit 3 entered commercial service in 1963, the fuel utilized stainless steel clad. The stainless steel clad experienced gross failures during operation. These failures were severe enough that radioactive fuel was released from the clad and dispersed throughout numerous plant systems, contaminating these systems with alpha emitting radionuclides, i.e., uranium and transuranic isotopes. Some external plant surfaces have also been similarly contaminated. HBPP completed the transition from stainless steel to zircaloy assemblies in 1969.

Over the many years since Unit 3 last operated in July 1976, beta and gamma emitting radionuclides have decayed, and alpha has become a more dominant factor in the potential dose contribution. Because alpha causes more severe biological damage when internal exposure occurs, the potential radiological dose consequences are likewise more severe. This issue leads to a plant-specific potential environmental concern for Unit 3 decommissioning.

Sections 1.1 through 1.11 provide a brief description, key assumptions, and summary of the results for each analyzed decommissioning accident. Mitigating measures in the form of administrative controls will be in place (other equivalent controls are acceptable), where appropriate, to minimize the potential radiological environmental impacts of decommissioning activities and to maintain them within regulatory limits.

All accident scenarios, contain surface contamination with the most limiting mixture of radionuclides (i.e., resulting in the highest total specific activity) as a component of the airborne release mixture. This limiting mixture is based on a sample of Spent Fuel Pool (SFP) crud per PG&E Calculation NX-322 (Reference 2.1). The mixture includes alpha emitters and other radioactive nuclides typical of BWR surface contamination. Some accident scenarios contain certain longer lived fission products, and neutron-activated plant metals and concrete as components of the release, depending on the decommissioning activity.

Detailed calculations, descriptions of the evaluations methodologies, and the mixture of radionuclides released for each of the decommissioning accidents in this appendix can be found in PG&E Calculations NX-323 through NX-333.

1.1 Dry Active Waste (DAW) Fire

The following is a brief summary of the postulated DAW fire scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-323 (Reference 2.2).

Description

The purpose of this analysis is to evaluate the offsite radiological dose consequences of a container (B-25 box) filled with DAW generated in alpha zones. Alpha zones are established at HBPP for the purpose of controlling contamination in areas having a high alpha to beta/gamma contamination ratio. Reference 2.1 identifies the limiting mixture of radionuclides used in this analysis and PG&E calculation NX-321 (Reference 2.3) identifies the maximum allowable radionuclide specific activities for qualifying the B-25 box for shipment as a Type A package.

Three different release scenarios have been considered in this analysis

- Filtered stack (elevated) release
- Un-filtered stack (elevated) release
- Un-filtered ground level release (fire in an open yard area)

Assumptions

The following key assumptions have been made:

The contents of the B-25 box are uniformly and totally consumed by the fire within 15-minutes of fire initiation.

Elevated releases, filtered or unfiltered, take place through the 50 foot plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

The NUREG/CR-0672 combustible waste fire accident analysis fraction of $1.5E-04$ of the radioactivity in the package is released by the fire.

Results

Filtered Stack Release -

The TEDE to a member of the public for a filtered stack release is $3.97E-06$ rem which is less than the EPA PAG limit of 1 rem.

Unfiltered Stack Release -

The TEDE to a member of the public for an unfiltered stack release is $3.97E-04$ rem, which is less than the EPA PAG limit of 1 rem. Therefore, the plant stack HEPA filters maintain the dose consequences ALARA (as evidenced by the filtered release results), but they are not required to keep the dose consequences within the EPA PAG limit for this postulated DAW fire scenario.

Unfiltered Ground Level Release -

The TEDE to a member of the public for an unfiltered ground level release is $4.01E-03$ rem, which is less than the EPA PAG limit of 1 rem.

Impact to Radiation Worker -

If a radiation worker is present in the room for the entire 15 minute duration of the fire without respiratory protection, the exposure to the worker would be 665 DAC-hours, which equates to a TEDE of 1.66 rem. The EPA PAG limit does not apply to a radiation worker; however, 10CFR20 radiation worker limits would not be exceeded.

1.2 Explosion of Liquid Propane Gas (LPG) Leaked from a Front-End Loader

The following is a brief summary of the postulated LPG explosion scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies and the graphical depiction of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixture, see PG&E Calculation NX-324 (Reference 2.4).

Description

The purpose of this analysis is to evaluate the offsite radiological dose consequences of an explosion caused by LPG mixing with air to produce a flammable mixture. The source of the LPG would be leakage from a front-end loader. The explosion results in destruction of plant stack HEPA filters and the adjacent ventilation ductwork. The explosive overpressure releases the accumulation of plant surface contamination and neutron-activated concrete dust from the HEPA filter and ventilation ductwork. One scenario occurs inside the plant with the airborne release discharging to atmosphere through the plant ventilation stack without the mitigating benefits of HEPA filtration. A second scenario assumes that the explosion damages the plant ventilation boundary and leads to an unfiltered ground level release.

Assumptions

The explosion occurs when the stack HEPA filters and associated ductwork have the maximum radionuclide inventory.

Results

Administrative Controls -

The following administrative controls maintain the Total Effective Dose Equivalent (TEDE) to a member of the public less than the EPA 400 Manual protective action threshold of 1 rem. It is also acceptable to implement other controls that are judged to either be equivalent or that would preclude the accident scenarios from occurring.

- Use electric equipment in lieu of LPG-operated equipment (such as forklifts) to preclude the conditions in this accident (i.e., an explosion from LPG-powered equipment).
- Where concrete surface contamination is removed prior to LPG-powered equipment operations that could lead to a decommissioning accident through the stack, the TEDE to a member of the public is 1.35E-04 rem.

1.3 Vacuum Filter Bag Rupture

The following is a brief summary of the postulated vacuum bag rupture scenarios, including a short description, key assumptions made, and results.

For all assumptions, as well as the detailed calculations and evaluation methodologies and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-325 (Reference 2.5).

Description

The purpose of this analysis is to evaluate the offsite radiological dose consequences from the rupture of a vacuum filter bag filled with waste generated in Unit 3 alpha zones. Alpha zones are established at HBPP for the purpose of controlling contamination in areas having a high alpha to beta/gamma contamination ratio.

Sharp objects, such as metal shards, could rupture a vacuum filter bag during surface cleaning operations using a vacuum cleaner. Therefore, if a vacuum is used in alpha zones and the filter bag should rupture, there is a potential of generating significant airborne radioactivity. When the filter bag is ruptured, a conservative portion of the collected material becomes airborne depending on the particular scenario. For example, all of the collected material in the bag is assumed to become airborne during a building release because of the motive forces of the vacuum cleaner air flow.

Three different release scenarios have been considered in this analysis:

- Filtered stack (elevated) release
- Un-filtered stack (elevated) release
- Un-filtered ground level release (open yard area)

Assumptions

The following key assumptions have been made:

Elevated releases, filtered or unfiltered, take place through the plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

The vacuum bag contains the equivalent of 30,000 grams (66 pounds) of SFP crud (Reference 2.1).

For a vacuum bag rupture inside a ventilated building, vacuum cleaner air flow results in all 30,000 grams becoming airborne.

Results

Filtered Stack Release -

The TEDE to a member of the public for a filtered stack release is $4.57E-02$ rem, which is less than the EPA 400 Protective Action Guide of 1 rem.

Administrative Controls -

The following administrative controls limit the alternate scenario consequences to within EPA 400 Protective Action Guide of 1 rem. It is also acceptable to implement other controls that are judged to either be equivalent or that would preclude the alternate accident scenarios from occurring.

Unfiltered Stack Release -

- Restrict the maximum vacuum filter bag contact dose rate to 2.02 rem/hr to maintain the maximum radiological impact to less than EPA 400 Protective Action Guide of 1 rem.

Unfiltered Ground Level Release -

- Restrict the maximum vacuum filter bag contact dose rate to 0.20 rem/hr to maintain the maximum radiological impact to less than EPA 400 Protective Action Guide of 1 rem.

1.4 Contamination Control Envelope Rupture

The following is a brief summary of the postulated rupture scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-326 (Reference 2.6).

Description

The purpose of this analysis is to evaluate the offsite radiological dose consequences of a contamination control envelope rupture.

If a contamination control envelope ruptures during oxyacetylene cutting, small globules of molten metal can project up to 9 meters from the cutting operation. It is postulated that such molten particles penetrate the plastic sheet walls of the envelope and increase the leakage from the contamination control envelope from 10 percent to 50 percent. The result of the rupture is an airborne dispersion of dust consisting of Spent Fuel Pool crud and activated metals.

The scenario occurs inside the plant with airborne radioactivity discharging to atmosphere through the plant ventilation stack HEPA filters.

An alternate scenario occurs inside the plant with the release to the atmosphere through the plant stack without HEPA filtration.

Assumptions

The following key assumptions have been made:

Elevated releases, filtered or unfiltered, take place through the plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

The increase in envelope leakage occurs during the removal of the reactor vessel when the highest specific activity material is being cut.

Since this task is remotely controlled, the large leak is ongoing for one hour of cutting before it is detected.

A scenario with the airborne radioactivity discharging directly to atmosphere at ground level is not credible because the airborne radioactivity is generated inside the plant ventilation boundary.

Results

Filtered Stack Release -

The TEDE to a member of the public for a filtered stack release is $2.52E-3$ rem which is less than the EPA 400 Protective Action Guide of 1 rem.

Unfiltered Stack Release -

The TEDE to a member of the public for an unfiltered stack release is 0.252 rem which is less than the EPA 400 Protective Action Guide of 1 rem.

1.5 Oxyacetylene Explosion

The following is a brief summary of the postulated explosion scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-327 (Reference 2.7).

Description

Oxyacetylene cutting torches may be used for removing and segmenting various steel components during dismantlement.

Oxyacetylene explosions can occur from such causes as flow reversals, nozzle obstructions, and flashbacks (a flare going back up the gas hose). The explosion occurs while operating oxyacetylene cutting tools within a 1000 m³ contamination control envelope. The result of the explosion is an airborne dispersion of dust consisting of neutron-activated reactor vessel steel and surface contamination. The accident also releases additional carbon steel from six destroyed HEPA filters.

The scenario occurs inside the plant with the airborne radioactivity discharging to atmosphere through the plant ventilation stack HEPA filters.

An alternate scenario occurs inside the reactor building with the radioactive dust from the explosion going through the plant ventilation system without any HEPA filtration and discharging to atmosphere through the plant stack.

A third scenario considers the unfiltered release at ground level.

Assumptions

Elevated releases, filtered or unfiltered, take place through the plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

Results

Filtered Stack Release

The TEDE to a member of the public for a filtered stack release is 7.29E-03 rem, which is less than the EPA 400 Protective Action Guide of 1 rem.

Unfiltered Stack Release

The TEDE to a member of the public for an unfiltered stack release is 0.729 rem which is less than the EPA 400 Protective Action Guide of 1 rem.

Administrative Controls -

The following administrative controls preclude the third ground level and unfiltered scenario from occurring. It is also acceptable to implement other controls that are judged to either be equivalent or that would preclude the alternate accident scenario from occurring.

- Use arc cutting tools rather than oxyacetylene to avoid the accident scenario or
- Assure that all oxyacetylene usage will be conducted in such a manner that no damage could occur to the plant stack or pressure boundary due to an explosion of the oxyacetylene.

1.6 Filter Damage from a Blasting Surge

The following is a brief summary of the postulated blasting surge scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-328 (Reference 2.8).

Description

The purpose of this analysis is to evaluate the offsite radiological dose consequences of HEPA filter damage due to a blasting surge. The historic background for this accident is the filter damage that occurred during the Elk River reactor decommissioning due to concrete blasting activities. The radionuclide inventories are neutron-activated concrete and concrete surface contamination. As concrete demolition occurs, the filter initially accumulates dust from the contaminated surface layer; subsequent drilling produces dust from activated concrete. Any accumulated neutron-activated concrete dust and surface contamination are released from the HEPA filter by the explosive overpressure.

The scenario occurs inside the plant and involves damage to a local HEPA filter installed as an RP control for the concrete blasting work. Airborne radioactivity from the damaged filter discharges to the atmosphere through the plant ventilation stack HEPA filters. An alternate scenario occurs inside the plant with the release to the atmosphere through the plant stack without HEPA filtration.

Assumptions

A scenario with the airborne radioactivity discharging directly to atmosphere at ground level is not credible based on the source of airborne radioactivity being inside the plant ventilation boundary.

Elevated releases, filtered or unfiltered, take place through the plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

Demolition blasting of the concrete biological shield walls is in progress. The explosion occurs near the end of the demolition so that the filter has a maximum radionuclide inventory.

14,000 grams (31 pounds) of neutron activated concrete dust are released from the damage filter. A much smaller mass of surface contamination is also released.

Results

Filtered Stack Release

The TEDE to a member of the public for a filtered stack release is $1.39\text{E-}03$ rem, which is less than the EPA 400 Protective Action Guide of 1 rem.

Unfiltered Stack Release

The TEDE to a member of the public for an unfiltered stack release is 0.139 rem which is less than the EPA 400 Protective Action Guide of 1 rem.

1.7 Detonation of Unused Explosives

The following is a brief summary of the postulated detonation scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-329 (Reference 2.9).

Description

The purpose of this analysis is to evaluate the offsite radiological dose consequences of the detonation of unused explosives. The explosive charges are onsite for the purpose of removing the biological shield. An accidental explosion occurs inside the Refueling Building with contamination shields and water spray off, causing concrete destruction and releasing a dust cloud of surface contamination.

The resultant airborne radioactivity is discharged to atmosphere through the plant ventilation stack HEPA filters. An alternate scenario occurs inside the plant with the release to the atmosphere through the plant stack without HEPA filtration. A third scenario discharging directly to atmosphere at ground level is considered.

Assumptions

Elevated releases, filtered or unfiltered, take place through the plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

The radionuclide release consists only of surface contamination, since the surface is the most likely material to become airborne in this scenario. The surface contamination is much greater in terms of activity than the structural concrete below, making this an acceptable simplification.

50 grams of concrete surface contamination become airborne.

Results

The TEDE to a member of the public for a filtered stack release is $7.59\text{E-}05$ rem, which is less than the EPA 400 Protective Action Guide of 1 rem.

The TEDE to a member of the public for an unfiltered stack release is $7.59\text{E-}03$ rem which is less than the EPA 400 Protective Action Guide of 1 rem.

The TEDE to a member of the public for a release assumed to occur at ground level with no filtering is $7.68\text{E-}02$ rem which is less than the EPA 400 Protective Action Guide of 1 rem.

1.8 Minor Transportation Accident

The following is a brief summary of the postulated minor transportation scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-330 (Reference 2.10).

Description

Accidents involving trucks carrying radioactive waste from a decommissioning site may result in the release of radioactive material. The purpose of this analysis is to evaluate the offsite radiological consequences of a minor transportation accident involving the release of radioactive material generated at the site. The scenario involves an airborne release from a fire and occurs to a truck shipment of waste containing the bounding mixture of radionuclides (Reference 2.1). Class A limits as defined in 10 CFR 61.55 and Utah Administrative Code R313-15-1008 were used to determine the bounding specific activity for each resident isotope.

Assumptions

As a result of a traffic accident, a Type A 55-gallon container catches fire and a fraction of $5\text{E-}04$ (.0005) of the bounding activity derived in Reference 2.3 becomes airborne.

A member of the public is located 100 meters downwind from the fire.

Results

55-gallon container fire

The TEDE to a member of the public is $4.54\text{E-}03$ rem, which is less than the EPA 400 Protective Action Guide of 1 rem.

1.9 Severe Transportation Accident

The following is a brief summary of the postulated severe transportation scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-331 (Reference 2.11).

Description

Transportation accidents involving trucks carrying radioactive waste from a decommissioning site may result in the atmospheric release of radioactive material. The purpose of this analysis is to evaluate the offsite radiological dose consequences of a severe transportation accident involving radioactive waste generated in alpha zones at the site. The analysis considers an intermodal container catching fire.

Reference 2.1 identifies the limiting mixture of radionuclides used in this analysis and Reference 2.3 identifies the maximum allowable radionuclide specific activities for shipment as Type A packages.

Assumptions

As a result of the traffic accident, an intermodal container catches fire and a fraction of $5E-04$ (.0005) of their activity becomes airborne.

A member of the public is located 100 m downwind from the accident.

Results

The TEDE of $3.04E-1$ rem to a member of the public is less than the EPA 400 Protective Action Guide of 1 rem.

1.10 HEPA Filter Fire

The following is a brief summary of the postulated HEPA filter fire scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-332 (Reference 2.12).

Description

A fire in a portable ventilation unit HEPA filter connected to an alpha zone contamination control envelope has the potential of generating significant airborne radioactivity. The purpose of this analysis is to evaluate the offsite radiological dose consequences of a fire in a HEPA filter being used in this configuration. Three different release scenarios have been considered in this analysis.

- Filtered stack (elevated) release
- Un-filtered stack (elevated) release
- Un-filtered ground level release (fire in an open yard area)

Alpha zones are established for the purpose of controlling contamination in areas having a high alpha to beta/gamma contamination level. In this analysis a portable ventilation unit is connected to a contamination control envelope and is processing material being generated within unit alpha zones. A typical portable filtered ventilation enclosure unit consists of a large blower coupled with a HEPA filter preceded by a glass-fiber roughing filter, all mounted on a wheeled cart. A flexible duct couples the cart unit to the contamination control envelope; the envelope surrounds the work area and confines the materials being generated. Roughing filters are installed at both the inlet and the outlet of the ventilation enclosure unit.

Assumptions

The HEPA filter is loaded with radioactive material with the limiting radionuclide mixture of Reference 2.1.

Elevated releases, filtered or unfiltered, take place through the plant stack.

The entire filter is uniformly consumed by the fire during a 15-minute period.

Results

Administrative Control

Based on current radiological practices at HBPP, the filter is changed out at a contact dose rate of 1 mrem/hr.

Filtered Stack Release

The TEDE is $8.16\text{E-}06$ rem which is less than the EPA 400 Protective Action Guide of 1 rem.

Unfiltered Stack Release

The TEDE is $8.16\text{E-}04$ rem which is less than the EPA 400 Protective Action Guide of 1 rem.

Unfiltered Ground Level Release

The TEDE is $8.25\text{E-}03$ rem which is less than the EPA 400 Protective Action Guide of 1 rem.

1.11 Loss of HEPA Filtration

The following is a brief summary of the postulated filtration loss scenarios, including a short description, key assumptions made, and results. For all assumptions, as well as the detailed calculations and evaluation methodologies, and the graphical depictions of the relationship between TEDE and the total curies released for the relevant radioactive nuclide mixtures, see PG&E Calculation NX-333 (Reference 2.13).

Description

Loss of portable ventilation unit HEPA filter capability while connected to an alpha zone contamination control envelope has the potential of generating significant airborne radioactivity. The purpose of this analysis is to evaluate the offsite radiological dose consequences of a loss of HEPA filtration capability while the filter is being used in this configuration.

Three different release scenarios have been considered in this analysis.

- Filtered stack (elevated) release
- Un-filtered stack (elevated) release
- Un-filtered ground level release

Alpha zones are established for the purpose of controlling contamination in areas having a high alpha to beta/gamma contamination level. In this analysis a portable ventilation unit is connected to a contamination control envelope and is processing material being generated within unit alpha zones. A typical portable filtered ventilation enclosure unit consists of a large blower coupled with a HEPA filter preceded by a glass-fiber roughing filter, all mounted on a wheeled cart. A flexible duct couples the cart unit to the contamination control envelope; the envelope surrounds the work area and confines the materials being generated. Roughing filters are installed at both the inlet and the outlet of the ventilation enclosure unit.

Assumptions

Elevated releases, filtered or unfiltered, take place through the plant stack.

A 99% removal efficiency is conservatively assumed for plant stack HEPA filtration rather than the design value of 99.7%, which reduces the airborne release by a factor of 0.01.

The unfiltered release would continue for a 15 minute period.

30,000 g (66 pounds) of the radionuclide mixture identified in Reference 2.1 is released in each scenario.

Results

Filtered Stack Release

The TEDE is 3.43E-02 rem, which is less than the EPA 400 Protective Action Guide of 1 rem.

Administrative Controls

The following administrative controls limit the alternate scenario consequences to within regulatory limits or preclude their occurrence. It is also acceptable to implement other controls that are judged to either be equivalent or that would preclude the alternate accident scenario from occurring.

Unfiltered Stack Release

- Incorporate HEPA filter redundancy into the ventilation unit to preclude an unfiltered stack release on the loss of a single HEPA filter or
- Limit the contamination control envelope Am-241 concentration to less than 1.06E-05 $\mu\text{Ci/ml}$.

Unfiltered Ground Level Release -

- Incorporate HEPA filter redundancy into the ventilation unit to preclude a ground level release on the loss of a single HEPA filter or
- Limit the contamination control envelope Am-241 concentration to less than 1.05E-06 $\mu\text{Ci/ml}$.

2.0 References

- 2.1 PG&E Calculation NX-322, "Decommissioning Accident Analysis-B, HBPP SFP Crud Sample Activity"
- 2.2 PG&E Calculation NX-323, "HBPP Dry Active Waste Fire Analysis"
- 2.3 PG&E Calculation NX-321, "Decommissioning Accident Analysis - A, HBPP Crud Waste Class A Limit"
- 2.4 PG&E Calculation NX-324, "Explosion of LPG Leaked from a Front-End Loader Accident Analysis"
- 2.5 PG&E Calculation NX-325, "Vacuum Filter-Bag Rupture Accident Analysis"
- 2.6 PG&E Calculation NX-326, "Contamination Control Envelope Rupture Accident Analysis"
- 2.7 PG&E Calculation NX-327, "Oxyacetylene Explosion Accident Analysis"
- 2.8 PG&E Calculation NX-328, "Filter Damage from a Blasting Surge Accident Analysis"
- 2.9 PG&E Calculation NX-329, "Detonation of Unused Explosives Accident Analysis"
- 2.10 PG&E Calculation NX-330, "HBPP Minor Transportation Accident Analysis"
- 2.11 PG&E Calculation NX-331, "HBPP Severe Transportation Accident Analysis"
- 2.12 PG&E Calculation NX-332, "HBPP HEPA Filter Fire Accident Analysis"
- 2.13 PG&E Calculation NX-333, "Loss of HEPA Filtration Accident Analysis"

Enclosure 2
PG&E Letter HBL-10-005

**HUMBOLDT BAY QUALITY
ASSURANCE PLAN, REVISION 26**



Nuclear Power Generation Humboldt Bay Power Plant

NUMBER L-4
VOLUME 4
REVISION 26
EFFEC DATE 12-2-09 @1700
PAGE 1 of 1

TITLE

HUMBOLDT BAY
QUALITY ASSURANCE PLAN

APPROVED BY

ORIGINAL SIGNED 12-2-09

DIRECTOR/PLANT MANAGER / DATE
HB NUCLEAR

(Procedure Classification - Quality Related)

1.0 SCOPE

- 1.1 The Humboldt Bay (HB) Quality Assurance Plan (QAP) is applicable to Humboldt Bay Power Plant (HBPP) Unit 3 SAFSTOR and decommissioning activities, as well as HB Independent Spent Fuel Storage Installation (ISFSI) activities.
- 1.2 The scope of Attachment 4.1 (Humboldt Bay Power Plant Unit 3 SAFSTOR Quality Assurance Plan) applies to HBPP Unit 3 SAFSTOR and decommissioning activities described therein.
- 1.3 The scope of Attachment 4.2 (Quality Assurance Requirements for the Humboldt Bay ISFSI) applies to the Humboldt Bay ISFSI activities described therein.

2.0 DISCUSSION

- 2.1 Attachment 4.1 is the HBPP Unit 3 SAFSTOR Quality Assurance (QA) Plan.
- 2.2 Attachment 4.2 contains the QA Requirements for the HB ISFSI.

3.0 INSTRUCTIONS

After the issuance of the HB QAP, revision 25, on 9/24/2009, any references within HBPP documents (procedures, programs, etc.) to either the SAFSTOR QA Plan or the Quality Assurance Program for the Diablo Canyon Power Plant, Units 1 & 2 (or any variations in title thereof), shall be construed to mean the HB QAP.

4.0 ATTACHMENTS

- 4.1 Humboldt Bay Power Plant Unit 3 SAFSTOR Quality Assurance Plan
- 4.2 Humboldt Bay ISFSI Quality Assurance Requirements

5.0 RESPONSIBLE ORGANIZATION

Quality Verification

PACIFIC GAS AND ELECTRIC COMPANY

HUMBOLDT BAY POWER PLANT UNIT 3

SAFSTOR QUALITY ASSURANCE PLAN

HUMBOLDT BAY POWER PLANT UNIT 3 SAFSTOR QUALITY ASSURANCE PLAN

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1.0 INTRODUCTION

1.1 Plan Objective

The objectives of Pacific Gas and Electric Company's (PG&E) Quality Assurance Plan (QA Plan) for Humboldt Bay Power Plant Unit 3 (HBPP-3) during SAFSTOR operation are:

- a) To establish and implement a graded quality assurance program based upon the appropriate criteria of 10 CFR 50, Appendix B to an extent that is commensurate with the scope of SAFSTOR activities and the required function of the HBPP Unit 3 systems, structures and components.
- b) To meet the regulatory requirements for quality assurance programs as specified in 10 CFR 71, "Packaging and Transportation of Radioactive Material", and in U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide 4.15 (December 1977), "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment".
- c) To assure compliance with the plant Technical Specifications.
- d) To implement administrative controls that were relocated from the Technical Specifications.
- e) To establish and implement QA requirements for major modifications and for significant dismantlement and/or decommissioning activities.

1.2 Plan Scope

- a) This QA Plan applies to the following HBPP-3 SAFSTOR activities:
 - Radiological monitoring of gaseous and liquid effluents and the environment.
 - Packaging and transportation of radioactive material within the scope of 10 CFR 71.
 - Implementing plant Technical Specifications.
 - Implementation of administrative controls relocated from the Technical Specifications.
- b) This QA Plan applies to all personnel involved in HBPP Unit 3 SAFSTOR activities.

1.3 QA Plan Control

The Director, Quality Verification (QV Director) shall be responsible to identify, prepare, submit for approval, and issue changes as are necessary to maintain the QA Plan current and in conformance with applicable regulatory requirements and PG&E commitments to the USNRC.

Changes to the commitments contained herein shall be submitted to the USNRC in accordance with 10 CFR 50.54.

1.4 QA Plan Implementation

The QA Plan consists of all the control measures that are established and implemented to achieve the objective of Section 1.1. These control measures, and the methods for implementing them, are described in a series of documents as follows:

- a) Quality Assurance Manual - (QA Manual) A corporate-level document that contains procedures that provide requirements to PG&E departments for implementing the QA Plan.
- b) Department Programmatic Procedures - Department-level documents that describe how the requirements of the QA Manual are implemented.

The measures taken to implement the criteria established within the QA Plan will be executed in a graded approach to an extent that is commensurate with the importance to safety.

2.0 ORGANIZATION

2.1 Background

- a) PG&E's program for assuring the quality and safety of the decommissioning of HBPP Unit 3 is organized in a structured manner with clearly defined levels of authority, assignments of responsibility, and lines of communication. Assignment of responsibility for an item or activity includes responsibility for its quality.

PG&E acknowledges full responsibility to its employees, stockholders, the general public, and affected governmental regulatory agencies for the establishment and execution of this QA Plan. The work of executing selected portions of this QA Plan may be delegated to organizations external to PG&E; however, in all such instances PG&E retains overall responsibility.

- b) Specific responsibilities pertaining to quality assurance matters are assigned to various individuals throughout the Company. In each instance, the assignment of a responsibility to an individual includes with it a commensurate delegation of sufficient authority that the person can, in fact, fulfill that responsibility.

Unless otherwise specifically prohibited, it is understood that the functions, tasks, and activities necessary to implement a responsibility may be delegated to and performed by other qualified individuals. Instances are documented in which authority is to be delegated or support services are to be provided.

That individual within PG&E who has been assigned a particular responsibility in this QA Plan is the only person within the Company who is authorized to perform the activities necessary to discharge that responsibility. Normally, the activities related to discharging that particular responsibility will be performed either by the person who has been assigned that responsibility or by personnel who are directly subordinate to and under the control of that person. However, circumstances may arise where it is considered either necessary or desirable to have such activities, or some portion of them, actually performed by someone else. In such cases, the assigning person retains responsibility.

- c) Verification of conformance to established requirements is accomplished by individuals or groups within the QV organization who do not have direct responsibility for performing the work being verified or by individuals or groups trained and qualified in quality assurance concepts and practices and independent of the performance of the task. The persons and organizations performing quality assurance functions have direct access to management levels which assure the ability to identify quality problems, recommend or provide solutions through designated channels, and verify implementation of solutions.

They are sufficiently free from direct pressure for cost/schedule and have the responsibility to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material. (The organizational positions with stop work authority are identified in the implementing procedures.) The Quality Assurance Organization reviews and evaluates quality-related procedures that provide different methods to either implement or may potentially deviate from the requirements of the QA Plan.

- d) This QA Plan uses generic titles and identifies functions and responsibilities for those titles used in this document. Subsequent changes (if any) to actual titles used in the organization are traceable to the QA Plan titles by the use of administrative procedures.

2.2 Organization Description and Responsibilities

- a) The President and Chief Executive Officer, Pacific Gas and Electric Company, has overall responsibility for the decommissioning and continued care of HBPP Unit 3. The President and Chief Executive Officer, Pacific Gas and Electric Company, reports directly to the Chairman of the Board of PG & E Corporation. Reporting to the President and Chief Executive Officer, Pacific Gas and Electric Company, is the Senior Vice President, Generation and Chief Nuclear Officer.

- b) The Senior Vice President, Generation and Chief Nuclear Officer (SVP & CNO) is responsible for the design, construction, and operation of PG&E's nuclear facilities, including the safe and efficient decommissioning of HBPP Unit 3. The SVP & CNO, or his delegate, approves and signs all official Company correspondence with the USNRC or its representatives.

The SVP & CNO is responsible for providing QA oversight and regulatory services, and upon request, specialized support to HBPP in the areas of security, operations, procurement, emergency planning, radiation protection, radwaste and effluents management. Specialized support is also provided by outside consultants as needed. Reporting directly to the SVP & CNO are the Site Vice President, the HBPP Nuclear Oversight Committee (NSOC), the Director, Quality Verification, and the HBPP Unit 3 Plant Staff Review Committee (PSRC). The Director and Plant Manager, Humboldt Bay Nuclear (Nuclear Plant Manager), reports directly to the Site Vice President, who reports to the SVP & CNO. The Nuclear Plant Manager is the individual designated for approving HBPP Unit 3 SAFSTOR operations and future decommissioning projects.

- c) The Nuclear Plant Manager has overall responsibility for SAFSTOR operations and future decommissioning projects at HBPP Unit 3.

The Nuclear Plant Manager is responsible for the conduct of all activities related to SAFSTOR and decommissioning of HBPP Unit 3. This includes responsibility for operation, maintenance, engineering, radiation protection, training, and security. The Nuclear Plant Manager is the chairman of the PSRC and the Plant ALARA Committee.

The Nuclear Plant Manager is responsible to develop, and is authorized to approve and direct the implementation of those programs, procedures, and instructions required for HBPP Unit 3 within limits established by this QA Plan, the HBPP Unit 3 Technical Specifications, the Defueled Safety Analysis Report (DSAR), and administrative guidelines established by the SVP & CNO. Design authority for HBPP Unit 3 has also been delegated to the Nuclear Plant Manager.

The Nuclear Plant Manager shall have the overall onsite responsibility for activities associated with Unit 3. He shall be accountable for adherence to the operating limits and requirements contained in the Technical Specifications. He shall be responsible for the operational command function. He shall delegate these responsibilities to other specific members of the plant staff during his absence.

- d) The Director, Engineering Services, Diablo Canyon, reports to the Senior Director of Engineering Services, Diablo Canyon, and is charged with the development, evaluation, qualification, testing and improvement of nondestructive examination procedures by PG&E, and for evaluation of these types of procedures that are used at HBPP by other organizations. The Senior Director of Engineering Services, Diablo Canyon, reports to the Site Vice President.
- e) The Manager, Regulatory Services, Diablo Canyon, is the principal corporate interface with the USNRC and other regulatory agencies on matters related to obtaining and maintaining licenses and permits for HBPP Unit 3.

The Manager, Regulatory Services, Diablo Canyon, reports directly to the Site Services Director, Diablo Canyon, who reports to the Site Vice President.

- f) The Director, Applied Technology Services, is responsible to provide, upon request, specialized technical investigations, tests, analysis, examinations and calibration services.

In addition, the Director, Applied Technology Services has been specifically charged with development, evaluation, qualification, testing and improvement of welding, brazing and heat treating procedures required by the company and evaluation of these procedures used at Humboldt Bay Power Plant. The Director, Applied Technology Services reports to the Senior Vice President, Engineering & Operations, who reports to the President and Chief Executive Officer, Pacific Gas and Electric Company.

- g) The quality control functions are provided by the HBPP Quality Control organization. This includes participating in and monitoring those day-to-day activities identified in the QA Plan, such as identifying quality problems, recommending or providing solutions to quality problems, verifying implementation of solutions to quality problems, evaluating inspection results, and assuring inspection requirements are satisfied. The designated quality control representative is the HBPP Quality Control Supervisor, who reports to the DCCP QV Director.
- h) The Director, Quality Verification, Diablo Canyon, (QV Director) is responsible for management of this QA Plan, and to assure that this QA Plan is established and effectively implemented by all involved organizations, both internal and external to PG&E. The responsibilities and qualifications of the QV Director, and the reporting relationships, internal organization and responsibilities of the Quality Verification department are described in the DCCP QA Program, contained in Chapter 17 of the DCCP Final Safety Analysis Report (FSAR) Update, and in applicable procedures. The title Director, Quality Verification, or Quality Verification Manager, may be used to describe the senior position in the Quality Assurance organization.

The Chairman of the Board, the President and Chief Executive Officer, Pacific Gas and Electric Company, the SVP & CNO, have given the QV Director the organizational freedom and delegated the requisite authority to investigate any area or aspect of the Company's operations as necessary to identify and define problems associated with establishment or execution of this QA Plan.

They have also delegated the authority to assess, audit and monitor the conduct of quality-related activities performed by or for PG&E to assure compliance with this QA Plan and other regulatory requirements; to recommend solutions for such problems to whatever management level is necessary; and to verify that effective corrective action is taken in a timely manner.

The QV Director has access to the SVP & CNO; the DCPD Site Vice President, the DCPD Station Director; the HBPP Nuclear Plant Manager, and appropriate directors, managers and supervisors for any significant quality-related problem or deficiency.

The QV Director has the authority and responsibility to stop work should there be a serious breach of any part of this QA Plan, or of technical or regulatory requirements wherein public health or safety could be involved; and is authorized to prepare, approve, and issue standard procedures prescribing a uniform, Company-wide method of assuring quality when such standardization is essential to the effectiveness of this QA Plan.

- i) The DCPD Quality Assurance Supervisor is designated by and reports to the QV Director. Specific responsibilities pertaining to quality assurance matters are assigned by the QA Plan and its implementing procedures and instructions. The QV Director, the DCPD Quality Assurance Supervisor and other off-site quality verification personnel are responsible for certain quality assurance activities as delegated by the QV Director. These activities include performance monitoring, quality assurance program maintenance, quality assessments and quality audits of Humboldt Bay Power Plant and Humboldt Bay Power Plant support activities.
- j) The Engineering, Environmental & Safety Manager reports to the Nuclear Plant Manager, and is responsible for oversight and implementation of engineering activities at HBPP.
- k) The Radiation Protection Manager reports to the Nuclear Plant Manager. The Radiation Protection Manager is responsible for implementing the radiation protection program at HBPP for the protection of the workers and members of the public.
- l) The Decommissioning Manager reports to the Nuclear Plant Manager. The Decommissioning Manager is responsible for management of decommissioning activities for HBPP.

2.3 Facility Staff Qualifications

Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1 1971 for comparable positions, except for :

- a) The Radiation Protection Manager shall meet or exceed Regulatory Guide 1.8, Revision 2, April 1987

2.4 Plant Staff Review Committee

- a) Purpose

The Plant Staff Review Committee (PSRC) shall meet on a regular basis to review overall operating and maintenance experience, proposed changes and tests, adequacy of procedures, and other matters that may have a bearing on nuclear or radiological safety at the plant.

- b) Membership

The PSRC shall be composed of members of the Plant staff who have responsibility in the areas of:

- Operations
- Mechanical maintenance
- Electrical maintenance
- Instrumentation and control maintenance
- Radiation protection
- Nuclear engineering

The Nuclear Plant Manager (or designee) shall chair the PSRC.

- c) Alternates

In the absence of a regular member, the Chair may designate an alternate from the plant staff to carry out review functions. A regular member shall be designated to serve as Chair in the absence of the Nuclear Plant Manager.

- d) Meeting Frequency

Once per calendar quarter and at other times at the discretion of the Chair.

- e) Quorum

A quorum shall consist of four regular members or three regular members and an alternate.

f) Responsibilities

- 1) The PSRC shall review the following items prior to implementation to determine if a change in the Technical Specifications or prior NRC approval as defined in 10 CFR 50.59 is involved, and shall render such determinations in writing:
 - a) Proposed tests and experiments determined by a Committee member to have nuclear safety significance.
 - b) Proposed changes or modifications to Unit 3 systems or equipment.
 - c) Proposed normal, abnormal, and emergency operating procedures, maintenance procedures, security procedures, administrative procedures, and other procedures determined by a Committee member to be significant to the maintenance of Unit 3 in SAFSTOR.
 - d) Proposed changes to approved procedures of the type described in item (c) above.
 - e) Proposed changes to the Technical Specifications and DSAR.
- 2) The PSRC shall periodically review:
 - a) Approved procedures of the type described in item (1)(c) above for currentness and applicability.
 - b) Maintenance and surveillance testing experience to ensure safe and efficient maintenance of the Unit and to determine if changes to equipment or procedures are needed.
- 3) The PSRC shall investigate any violation of the Technical Specifications and prepare and forward a report to the SVP & CNO and the Nuclear Safety Oversight Committee (NSOC) covering their evaluation and recommendations to prevent recurrence. The format for this report shall be identified in procedures that describe PSRC functions.
- 4) The PSRC shall conduct a biennial review of:
 - a) The Plant Security Plan and implementing procedures to determine the need for changes in the plan or its implementing procedures.
 - b) The Site Emergency Plan and its implementing procedures to determine the need for changes in the plan or its implementing procedures.

g) Authority

- 1) The PSRC shall recommend to the Nuclear Plant Manager approval or disapproval of proposals reviewed under items (f) (1) through (3) above.
- 2) The PSRC shall render written determinations regarding whether or not a proposed change or test or other such matter which has been reviewed requires prior NRC approval as defined in 10 CFR 50.59(c) or a change in the Technical Specifications.
- 3) In the event of disagreement between PSRC members on a matter affecting nuclear or radiological safety, a conservative course shall be followed as determined by the Nuclear Plant Manager. Records of such disagreements shall be included in the meeting minutes, described in item (h) below, and distributed promptly.

h) Records

Minutes of each PSRC meeting shall be prepared and maintained at the Plant. Copies of minutes shall be sent to the SVP & CNO, and to the NSOC.

2.5 Nuclear Safety Oversight Committee

a) Purpose

The Nuclear Safety Oversight Committee (NSOC) shall function to provide independent review and audit of designated activities in the areas of:

- Nuclear power plant operations
- Chemistry and radiochemistry
- Instrumentation and controls
- Mechanical and electrical engineering
- Nuclear engineering
- Metallurgy
- Radiological safety
- Quality assurance practices

The NSOC shall report to and advise the SVP & CNO on those areas of responsibility specified in item (f) below.

b) Composition

The NSOC shall be composed of a Chair and a minimum of four members. The NSOC Chair and members shall be appointed in writing by the SVP & CNO.

The NSOC Chair shall have a minimum of six years of professional level managerial experience in the power field and the NSOC members shall have a minimum of five years of professional level experience in the field of their specialty.

The NSOC Chair and all members shall have qualifications that meet or exceed the requirements and recommendations of Section 4.7 of ANSI/ANS 3.1-1978.

c) Consultants

Consultants shall be utilized as determined by the NSOC Chair to provide expert advice to the NSOC.

d) Meeting Frequency

The NSOC shall meet not less than twice a year in accordance with Regulatory Guide 1.33 – 1978, which authorizes the practice in standard ANSI N18.7-1976.

e) Quorum

A quorum of the NSOC necessary for the performance of the NSOC function shall be a majority (one-half or more) of the members in accordance with Regulatory Guide 1.33 – 1978, which authorizes the practice in standard ANSI N18.7-1976. In addition, no more than a minority of the quorum shall have line responsibility for the plant.

f) Review

The NSOC shall review:

- 1) The 10 CFR 50.59 evaluations for (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions did not require prior NRC approval.
- 2) Proposed changes to procedures, equipment, or systems that require prior NRC approval as defined in 10 CFR 50.59.
- 3) Proposed tests or experiments that require prior NRC approval as defined in 10 CFR 50.59.
- 4) Proposed changes to Technical Specifications or the License.
- 5) Violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.

6) Significant operating abnormalities or deviations from normal and expected performance of Unit equipment.

7) Events requiring written notification to the Commission.

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

9) The PSRC's reports and meeting minutes.

g) Records

Records of the NSOC activities, including minutes of meetings, reports of reviews, and audit reports shall be prepared, approved and distributed as required by the approved NSOC Charter.

3.0 QUALITY ASSURANCE REQUIREMENTS

3.1 General Requirements

Quality assurance requirements applicable to all the activities within the scope of this QA Plan are specified in this section. Additional requirements for radiological monitoring, radioactive material packaging and transportation, and Technical Specification activities are specified in Sections 3.2, 3.3, and 3.4, respectively. Administrative Controls requirements relocated from the Technical Specifications per License Amendment 41 are contained in sections 3.5, 3.6 and 3.7. The requirements for activities outside the scope of this QA Plan are specified in Section 3.8.

3.1.1 Procedures, Instructions, and Drawings

This QA Plan shall be implemented by procedures, instructions, or drawings prepared and utilized by organizations having responsibility to perform the activities described herein.

Standard guidelines for the format, content, and review and approval processes shall be established and set forth in written procedures or instructions issued by the organizational units.

Procedures and instructions shall identify the required interfaces with other organizations and shall delineate the responsibilities of each for the specific activity. Procedures and instructions shall be reviewed by other organizations with interface responsibilities and comments forwarded to the issuing organization for resolution.

Procedures and instructions shall be reviewed and concurred with by independent personnel, trained in quality assurance concepts and practices, for compliance with and implementation of the requirements of this QA Plan.

a) Procedure Review

The following procedures, and changes thereto, shall be reviewed by the PSRC and approved by the Nuclear Plant Manager, prior to implementation, except as provided in items (b) and (c) below:

- 1) Normal startup, operation, and shutdown of systems and components required during SAFSTOR
- 2) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components
- 3) Actions to be taken during emergency conditions involving unplanned releases of radioactivity
- 4) Abnormal and emergency operation of all systems and components required to maintain the SAFSTOR condition of the Plant
- 5) Surveillance activities required to demonstrate compliance with the Technical Specifications
- 6) Calibration of instrumentation used to demonstrate compliance with Technical Specifications
- 7) Shipping and disposal of radioactive materials
- 8) Process Control Program
- 9) Fire Protection Program implementation

b) Procedure Changes

Rules shall be established that provide methods by which temporary changes can be made to approved procedures, including the designation of those persons authorized to approve such changes.

Temporary changes that clearly do not change the intent of the approved procedure from the standpoint of nuclear safety may be approved by two members of the plant management staff. Such changes shall be documented and, if appropriate, incorporated into the next revision of the affected procedure.

c) Emergencies Not Covered by a Procedure

In the event of an emergency not covered by an approved procedure, operations personnel shall be instructed to take action to minimize personnel injury and damage to the facility.

3.1.2 Document Control

Documents and changes to documents that prescribe and verify activities affecting quality shall be controlled in a manner that precludes the use of inappropriate or outdated documents.

Procedures and instructions shall provide means to assure that documents, including changes, are prepared, reviewed, and approved for release by authorized personnel; distributed prior to commencing work; and used in performing the activity.

The organization responsible for establishing instructions, procedures, drawings, or other documents prescribing and verifying activities affecting quality shall also be responsible for developing and implementing systematic methods for the control of such documents.

A document control system shall be established to: (a) identify the current revision of instructions, procedures, and drawings; and (b) assure the use thereof.

The organization issuing procedures shall be responsible to maintain a file of all procedure revisions issued.

3.1.3 Inspection

Inspections of radioactive material packaging and transportation activities for greater than Type A quantities of radioactive materials (covered by 10 CFR 71, "Packaging and Transportation of Radioactive Material"), shall be performed by individuals independent of the individuals who performed the work.

Operational inspections of equipment in service or being returned to service, and of activities conducted in accordance with USNRC Regulatory Guide 4.15 (December 1977); of radioactive material packaging and transportation activities for Type A quantities (or less) of radioactive materials (covered by 10 CFR 71, "Packaging and Transportation of Radioactive Material"); and the Unit 3 Technical Specifications, shall be performed by individuals performing the work or by other individuals within the organization.

3.1.4 Corrective Action and Nonconformance Control

Measures shall be established in written procedures and utilized for documenting, reviewing, and dispositioning of quality problems and nonconformances occurring in the conduct of the activities within the scope of this QA Plan.

Technical decisions for the disposition of nonconformances shall be made by personnel with assigned authority in the relevant discipline.

These measures shall include provisions for identification of deficiencies and implementing corrective action to prevent recurrence

3.1.5 Indoctrination and Training

Personnel involved in implementing the activities within the scope of this QA Plan shall be responsible for the quality of their work. These personnel shall receive:

- Indoctrination in the requirements of this QA Plan.
- Indoctrination in their organization's implementing procedures.
- Training and qualification in tasks requiring special skills or knowledge in accordance with the requirements referenced in Sections 3.2, 3.3, and 3.4.

Indoctrination, training, qualification, and re-qualification (when applicable) shall be prescribed and performed in accordance with written procedures which specify the management responsibilities; training areas; frequency of training; method of qualification and requalification; and documentation requirements.

Each organization shall be responsible for the training of its own personnel. The Quality Verification department shall assist applicable organizations by providing indoctrination in the purposes and requirements of this QA Plan.

3.1.6 Records

Records shall be maintained, in accordance with written procedures, to furnish evidence that items or activities affecting quality meet: (a) technical requirements, applicable procedures, instructions, drawings, and other documents; and (b) regulatory requirements.

Participating organizations shall establish a control system for the collection, storage, and maintenance of completed quality assurance records in accordance with USNRC Regulatory Guide 1.88 (October 1976), Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records." (PG&E shall comply with a 1-hour fire rating for SAFSTOR rather than that specified therein.)

Records shall be assigned a retention period in conformance with applicable regulatory requirements and the following:

a) Record Retention

1) Five-Year Retention

All records and logs relative to the following areas shall be retained for at least 5 years:

- a) Records and logs of normal SAFSTOR operations.
- b) Records and logs of principal maintenance activities, including inspection, repair, substitution, or replacement of principal items of equipment described in the Technical Specifications.
- c) Reportable Occurrence Reports
- d) Records of periodic checks, inspections, and calibrations performed to verify that surveillance requirements are being met.
- e) Records of radioactive shipments.
- f) Records of sealed source leak tests and results.
- g) Records of the annual physical inventory of all source material of record.
- h) Records of tests or experiments associated with SPENT FUEL storage.
- i) Records of changes made in operating procedures.

2) SAFSTOR Duration

All records relative to the following areas shall be retained for the duration of SAFSTOR:

- a) Records and prints of changes made to the Plant.
- b) Records of spent fuel inventory, transfers of fuel, and assembly histories.
- c) Records of plant radiation and contamination surveys.
- d) Records of offsite environmental monitoring surveys.
- e) Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR 20.
- f) Records of radioactivity in liquid and gaseous wastes released to the environment.
- g) Records of training and qualification for current members of the plant staff.
- h) Minutes of meetings of the PSRC and NSOC.
- i) Records of Quality Assurance activities required by the SAFSTOR Quality Assurance Plan.
- j) Records of reviews performed for changes made to procedures or equipment, or reviews of tests and experiments pursuant to the Defueled Safety Analysis Report (DSAR).
- k) Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

3.1.7 Audits

Audits shall be conducted in accordance with USNRC Regulatory Guide 1.144 (January 1979), "Auditing of Quality Assurance Programs for Nuclear Power Plants," with the exception that audit frequencies shall be as specified herein.

For audits other than the Emergency Plan and Security Plan, a grace period of up to 90 days may be utilized when the urgency of other priorities makes meeting the specified schedule date impractical.

For audit activities deferred by using the grace period, the next scheduled audit due date shall be based on the originally scheduled date, and may not exceed the original due date plus 90 days.

Auditors shall be independent of direct responsibility for the performance of the activities they audit; have experience or training commensurate with the scope and complexity of their audit responsibility; and be qualified in accordance with USNRC Regulatory Guide 1.146 (August, 1980), "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants."

The above noted "90 day grace period" may also be applied to Auditor re-qualification frequencies if necessary. If an auditor's re-qualification is deferred by using the grace period, the next scheduled due date shall be based on the originally scheduled date, and may not exceed the original due date plus 90 days.

Audit reports shall be prepared, signed by the lead auditor and issued to responsible management of both the audited and auditing organizations.

Management of the audited organization shall review the audit report and investigate any adverse findings to identify their cause and determine the extent of corrective action required, including action to prevent recurrence. They shall schedule such corrective action and also take appropriate action to assure it is accomplished as scheduled. They shall respond to the Quality Verification Director regarding each adverse finding, give the results of their review and investigation, and clearly state the corrective action taken or planned.

The QV Director shall: receive the written response to audit findings; evaluate the adequacy of each response; assure that corrective action is identified and taken for each adverse finding; and confirm that corrective action is accomplished as scheduled.

Audits of Unit 3 activities shall be performed under the cognizance of the NSOC. The NSOC shall report to and advise the SVP & CNO on the audit program. These audits shall encompass:

- 1) The conformance of Unit 3 operation to provisions contained within the Technical Specifications and applicable license conditions - at least once per 24 months.
- 2) The performance, training, and qualifications of the entire Unit staff - at least once per 24 months.
- 3) The results of actions taken to correct significant deficiencies occurring in Unit equipment, structures, systems or methods of operation - at least once per 24 months.

- 4) The performance of the following activities required by the Quality Assurance Program – at least once per 24 months:
 - a) The Quality Assurance Program.
 - b) The Radiation Protection Program.
 - c) Radiological Effluents Program.
 - d) Radiological Environmental Monitoring Program.
 - e) Radioactive Material Packaging and Transportation.
 - f) Radioactive Waste Processing & Process Control Program
- 5) The Emergency Plan and implementing procedures – at least once per 24 months.
- 6) The Fire Protection and Loss Prevention Program and Implementing Procedures - at least once per 24 months. The audit team will include qualified offsite licensee personnel or an outside consultant.
- 7) Supplemental audits (or independent assessments) shall be performed as authorized by the Quality Verification Director based on the following considerations:
 - When significant changes are made in functional areas of the QA program such as significant reorganizations or procedure changes.
 - When it is suspected that the quality of an item or activity is in jeopardy due to deficiencies in the QA program.
 - When a systematic, independent assessment of program effectiveness is considered necessary.
 - When supplemental audits or assessments are necessary to verify implementation of required corrective action.

3.2 Radiological Monitoring

In addition to the quality assurance requirements specified in Section 3.1 of this QA Plan, radiological monitoring of gaseous and liquid effluents and the environment shall be controlled in accordance with USNRC Regulatory Guide 4.15 (December 1977).

3.3 Radioactive Material Packaging and Transportation

In addition to the quality assurance requirements specified in Section 3.1 of this QA Plan, containers used for packaging and transportation of radioactive materials within the scope of 10 CFR 71 shall be controlled in accordance with the NRC-approved DCPD Quality Assurance Program, as contained in the relevant sections of Chapter 17 of the DCPD FSAR Update and thereby in compliance with the quality assurance requirements of Subpart H of 10 CFR 71.

Relative to Section 17.1, DCPD FSAR Update, the plant organization referenced therein shall be the organization described in this QA Program.

3.4 Technical Specification Activities

In addition to the quality assurance requirements specified in Section 3.1 of this QA Plan, Technical Specification activities shall be controlled in accordance with the Limiting Conditions for Operations (LCO) and Surveillance Requirements (SR).

3.5 Offsite Dose Calculation Manual

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Monitoring Report, and Annual Radioactive Effluent Release Report, required by Section 3.7.2 and Section 3.7.3, of this QA Plan.
- c. Licensee initiated changes to the ODCM:
 1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 - i) sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
 - ii) a determination that the change(s) will maintain the level of radioactive effluent control required by 10CFR 20.1302, 40CFR Part 190, 10CFR 50.36a and Appendix I to 10CFR 50, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

2. Shall become effective after review and acceptance by the Plant Staff Review Committee and approval of the Nuclear Plant Manager; and
3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

3.6 Radioactive Effluent Controls Program

This program conforms with 10CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable (ALARA). The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,
2. Limitations on the instantaneous (average over a one-hour period) concentrations of radioactive material released in liquid effluents to Humboldt Bay conforming to ten times the effluent concentration limits of 10CFR Part 20, Appendix B, Table 2, Column 2,
3. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10CFR 20.1302 and with the methodology and parameters in the ODCM,
4. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to Humboldt Bay conforming to the dose design objectives of Appendix I to 10CFR Part 50,
5. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6. Limitations on the operability and use of the liquid effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to the dose design objectives of Appendix I to 10CFR Part 50,
7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be established as follows:
 - a. For noble gases: less than or equal to an instantaneous dose rate (average over a one-hour period) of less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and
 - b. For tritium and radionuclides in particulate form with half-lives greater than 8 days: less than or equal to a dose rate (averaged over a one-week period) of 1500 mrem/yr to any organ.
8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary conforming to Appendix I to 10CFR Part 50,
9. Limitation on the annual and quarterly doses to a member of the public from tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the site boundary conforming to the dose design objectives of Appendix I to 10CFR Part 50, and
10. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40CFR Part 190.

3.7 Reporting Requirements

The following reports shall be submitted in accordance with 10CFR 50.4.

3.7.1 Occupational Radiation Exposure Report

An annual report shall be made of personnel exposure, in accordance with the requirements of 10CFR Part 20.2206. The report shall be submitted by April 30 of each year.

3.7.2 Annual Radiological Environmental Monitoring Report

The Annual Radiological Environmental Monitoring Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Monitoring Report shall include the results of analyses of radiological environmental samples and of environmental radiation measurements taken during the period pursuant to the quality related locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in the next annual report.

3.7.3 Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the activities of the unit in the previous year shall be submitted prior to April 1 of each year in accordance with 10CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR 50.36a and 10CFR Part 50, Appendix I, Section IV.B.1.

3.8 Activities Outside the SAFSTOR QA Plan Scope

In addition to the quality assurance requirements specified in Section 3.1 of this QA Plan, a Project Quality Plan shall be developed and approved prior to conducting projects or activities that are outside the scope of this QA Plan

These activities shall include major modifications and significant dismantlement and/or decommissioning activities that affect quality-related SAFSTOR systems, structures and components described in the Unit 3 Defueled Safety Analysis Report (DSAR).

Project Quality Plans provide a method to identify and supplement the quality assurance program controls for a specific project or activity. Project Quality Plans also provide a description of the project or activity scope, identify the affected organizations and responsibilities, and include the quality verification activities used to assure the project or activity meets QA program requirements.

QUALITY ASSURANCE REQUIREMENTS FOR THE HUMBOLDT BAY ISFSI

QUALITY ASSURANCE REQUIREMENTS FOR THE HUMBOLDT BAY ISFSI

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QUALITY ASSURANCE REQUIREMENTS FOR THE HUMBOLDT BAY ISFSI

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QUALITY ASSURANCE REQUIREMENTS FOR THE HUMBOLDT BAY ISFSI

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QUALITY ASSURANCE REQUIREMENTS FOR THE HUMBOLDT BAY ISFSI

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QUALITY ASSURANCE REQUIREMENTS FOR THE HUMBOLDT BAY ISFSI

17.1 ORGANIZATION

The Pacific Gas and Electric Company's (PG&E) efforts to assure the quality and safety of the independent spent fuel storage installation (ISFSI) is organized in a structured manner with clearly defined levels of authority, assignments of responsibility, and lines of communication. Assignment of responsibility for an item or activity includes responsibility for its quality. Figure 17.1-1 depicts the organizational structure of PG&E. The position of the quality verification (QV) organization in the utility organization is shown in Figure 17.1-2.

PG&E has assumed full responsibility to its employees, stockholders, the general public, and affected governmental regulatory agencies for the establishment and execution of the Quality Assurance (QA) Program prescribed herein, quality related program directives and administrative procedures. The work of executing selected portions of the QA Program may be delegated to organizations external to PG&E; however, in all such instances, PG&E retains overall responsibility.

Specific responsibilities pertaining to quality assurance matters are assigned by the QA Program and its implementing procedures and instructions to various individuals throughout PG&E. In each instance, the assignment of a responsibility to an individual includes with it a commensurate delegation of sufficient authority that the person can, in fact, fulfill that responsibility. Unless otherwise specifically prohibited, it is understood that the functions, tasks, and activities necessary to carry out a responsibility may be delegated to and performed by other qualified individuals. All delegations of functions, tasks, activities, and authority shall be documented.

Figure 17.1-2 identifies those individuals and organizational components of PG&E with direct responsibilities related to the quality of the design, fabrication, construction, testing, operation, maintenance, modification, and decommissioning of ISFSI structures, systems, and components (SSCs) that are important to safety.

The narrative description throughout this section is based on Figures 17.1-1 and 17.1-2.

THE BOARD OF DIRECTORS OF PG&E CORPORATION is responsible for all facets of PG&E's utility business.

THE CHAIRMAN, CEO, AND PRESIDENT, PG&E CORPORATION, is accountable to the Board of Directors and establishes the corporate policies, goals, and objectives related to all of PG&E's activities and operations. Reporting to the Chairman, CEO, and President is the President and Chief Executive Officer - PG&E Company.

THE PRESIDENT AND CHIEF EXECUTIVE OFFICER - PG&E, is a member of the Board of Directors and is responsible for and directs the planning, distribution, and development of all the Company's energy resources and nuclear power generation. These functions include such activities as planning and development, engineering, construction, and fossil and nuclear power plant and ISFSI operations. Reporting to the President and Chief Executive Officer is the Senior Vice President and Chief Operating Officer.

The SENIOR VICE PRESIDENT and CHIEF OPERATING OFFICER is responsible for leading and managing the utility's day-to-day operations, including oversight of energy delivery, engineering and operations, generation, ISTS, and shared services. Reporting to the Senior Vice President and Chief Operating Officer is the Senior Vice President - Generation and Chief Nuclear Officer; the Senior Vice President – Engineering & Operations; and the Vice President Shared Services.

THE SENIOR VICE PRESIDENT – ENGINEERING AND OPERATIONS, through the Director – Applied Technology Services, is responsible for providing, upon request: (1) technical investigations, tests, analyses, examinations, and calibration services in support of the Humboldt Bay Power Plant and the ISFSI; (2) developing, evaluating, qualifying, testing, and improving welding, brazing, and heat-treating procedures required by the company; and (3) providing evaluation support of these procedures.

THE VICE PRESIDENT – SHARED SERVICES AND CHIEF PROCUREMENT OFFICER, through the Support Services Supervisor – Engineering Records Unit, is responsible for providing document services support for the Humboldt Bay Power Plant and the Humboldt Bay ISFSI. These services include indexing, preparing, and duplicating microfiche for the drawing control system; storing the master microfiche and drawings that cannot be microfilmed; and scanning and indexing drawings when requested. They also provide remote storage of master microfilm reels for the records management system (RMS) and storage of vendor manuals. The Vice President – Shared Services and Chief Procurement Officer, through the Manager, Procurement Services, is responsible for administering, coordinating, planning, and operation of warehousing and procurement of materials in support of HBPP and ISFSI operations and construction, as well as for contract services.

THE SENIOR VICE PRESIDENT - GENERATION AND CHIEF NUCLEAR OFFICER, is responsible for overall ISFSI safety and for taking measures needed to ensure acceptable performance of the ISFSI staff in designing, fabricating, constructing, testing, operating, modifying, decommissioning, and providing technical support to the ISFSI. Reporting directly to the Senior Vice President - Generation and Chief Nuclear Officer is the Site Vice President and the Director, Quality Verification. The Director and Plant Manager - Humboldt Bay Nuclear; reports directly to the Site Vice President.

Verification; and the Employee Concerns Program supervisor. The Senior Vice President - Generation and Chief Nuclear Officer, or his designee, as specified in administrative procedures, approves and signs official company correspondence to the U.S. Nuclear Regulatory Commission (NRC) or its representatives.

The Independent Review and Audit Program reports to the Senior Vice President – Generation and Chief Nuclear Officer. He approves revisions to the QA Program as described herein that constitute a reduction in a commitment made to the NRC. He also approves revisions to program directives.

The DIRECTOR AND PLANT MANAGER - HUMBOLDT BAY NUCLEAR, is responsible for the conduct of all activities related to the Humboldt Bay ISFSI. This includes responsibility for operation, maintenance, engineering, radiation protection, training, and security. He is the chairman of the Humboldt Bay PSRC. He is responsible to develop, and is authorized to approve and direct the implementation of those programs, procedures, and instructions required for the ISFSI within limits established by this QA Program, the Humboldt Bay ISFSI Technical Specifications, and administrative guidelines established in the Humboldt Bay ISFSI Final Safety Analysis Report (FSAR) Update. Design authority for the Humboldt Bay ISFSI has also been delegated to the Director and Plant Manager - Humboldt Bay Nuclear.

THE HBPP ENGINEERING MANAGER reports directly to the Director and Plant Manager - Humboldt Bay Nuclear, and is responsible for technical aspects of the engineering and design of Humboldt Bay ISFSI SSCs for monitoring system performance and trends; for performance of modifications to the Humboldt Bay ISFSI; for configuration control and design bases defense and management; for quality classification of Humboldt Bay ISFSI SSCs; and for the specification of technical and quality requirements for the purchase of Humboldt Bay ISFSI material and equipment.

THE DIRECTOR - QUALITY VERIFICATION, is responsible for management of the QA Program and for assuring that the QA Program prescribed herein, program directives, and administrative procedures are effectively implemented and complied with by all involved organizations, both internal and external to PG&E. The Chairman, CEO, and President - PG&E Corporation; the President and Chief Executive Officer - PG&E; the Senior Vice President and Chief Operating Officer; and the Senior Vice President - Generation and Chief Nuclear Officer have given the Director, Quality Verification, the organizational freedom and delegated the requisite authority to investigate any area or aspect of PG&E's operations as necessary to identify and define problems associated with establishment or execution of the QA Program. They have also delegated to the Director, Quality Verification, the authority to initiate, recommend, or provide solutions for such problems to whatever management level is necessary, and to verify that effective corrective action is taken in a timely manner. This delegation includes the authority to assess, review, inspect, audit, and monitor the conduct of quality-related activities performed by or for PG&E to assure compliance with the QA Program and other regulatory requirements.

The Director - QV, reports directly to the Senior Vice President - Generation and Chief Nuclear Officer and has access to the Chairman, CEO, and President - PG&E Corporation; the President and Chief Executive Officer - PG&E; the Senior Vice President and Chief Operating Officer; and the Director and Plant Manager - Humboldt Bay Nuclear; and appropriate directors and managers for any significant quality-related problem or deficiency. He is authorized to prescribe a uniform company-wide method of performing an activity affecting quality by sponsoring or requiring the issuance of procedures when such standardization is considered desirable or essential to the effectiveness of the QA Program. Such uniform methods are contained in program directives and administrative procedures, and compliance with their requirements by all PG&E personnel is mandatory.

The Director - QV, will not be responsible for any activities unrelated to responsibilities described in the QA Program that would prevent the required attention to QA matters. Further, the responsibility of the implementation of the QA Program will take precedence over the other non-QA duties.

The Director - QV, shall meet the following qualification requirements: management experience through assignments to responsible positions; knowledge of QA regulations, policies, practices, and standards; and experience working in QA or related activity in reactor design, construction, or operation or in a similar highly technological industry. At the time assignment to the active position, the Director - QV, shall have six years experience in implementing quality assurance, preferably at an operating nuclear plant, or operations supervisory experience. At least one year of these six years of experience shall be nuclear power plant experience in the overall implementation of the QA Program. A minimum of one year of this six-year experience requirement shall be related technical or academic training. A maximum of four years of this six-year experience requirement may be fulfilled by related technical or academic training.

The Director - QV, is responsible to regularly assess and report on the status, adequacy, and effectiveness of this QA Program to the Senior Vice President - Generation and Chief Nuclear Officer and other affected PG&E management and nuclear oversight committees. He is responsible to identify, prepare, and submit for approval such changes to the QA Program prescribed herein as are necessary to maintain the QA Program up to date and in conformance with current regulatory requirements and PG&E commitments to the NRC. He is responsible for the review of all regulatory submittals as they pertain to the QA Program, and his concurrence is required prior to submittal. He is responsible for assessing and assuring that the QA Program is effectively implemented at the ISFSI site. He assures timely and effective corrective actions through audits, regular assessments, and quality assessment status reports. Reporting to the Director - QV, are the quality assurance, supplier quality, project quality, and independent quality control inspection functions.

The Director - QV, is responsible for providing recommendations on solutions to quality problems and performing monitoring, assessments, independent QC inspections, reviews, and audits for the areas covered by the QA Program including supplier quality. The Director - QV, is also responsible for quality assurance associated with the Humboldt Bay Power Plant.

The Director - QV, has the authority and responsibility to stop work should there be a serious breach of any part of the QA Program, or of technical or regulatory requirements wherein public health or safety could be involved.

Through the conduct of assessments, audits, reviews, monitors, and independent QC inspections, the Director - QV, is responsible for quality overview of ISFSI design, fabrication, construction, testing, operation, modification, decommissioning, and related activities to verify independently that these activities are performed correctly and that human errors are reduced as much as practicable.

THE EMPLOYEE CONCERNS PROGRAM SUPERVISOR reports to the Senior Vice President - Generation and Chief Nuclear Officer.

THE DCPM MANAGER - PROCUREMENT SERVICES, reports through the Director, Generation Supply Chain, to the Vice President – Shared Services and Chief Procurement Officer and is matrixed to the DCPM Director - Site Services. The DCPM Manager – Procurement Services, is responsible for administering, coordinating, planning, and operation of warehousing and procurement of materials in support of HBPP and ISFSI operations and construction, as well as for contract services. This position is responsible for the functions within the materials procurement group including: the procurement specialist group, warehousing operations, administrative coordination of warehouse quality control receipt inspection activities, and materials coordination.

The DIRECTOR - GEOSCIENCES, is matrixed to the Director and Plant Manager, HBPP, and is responsible for providing geo-scientific studies; reports, and calculations (including geology, seismology, vibration ground motion studies, surface faulting, stability of subsurface materials, and slope stability) in support of the ISFSI and HBPP.

The following committees function at the managerial level within PG&E to provide review of ISFSI design, maintenance, and operation activities.

THE NUCLEAR SAFETY OVERSIGHT COMMITTEE, which reports to the Senior Vice President - Generation and Chief Nuclear Officer, implements the Independent Review and is described in Section 17.2.3.

THE HBPP PLANT STAFF REVIEW COMMITTEE reports to the Senior Vice President - Generation and Chief Nuclear Officer, and is responsible to advise on matters related to nuclear safety. The Committee is responsible for providing timely and continuing monitoring of ISFSI operating activities to assist the Director and Plant Manager - Humboldt Bay Nuclear, in keeping aware of general ISFSI conditions and to verify that day-to-day operating activities are conducted safely and in accordance with applicable administrative controls. The committee

performs periodic reviews of ISFSI operating activities to evaluate operations and to plan future activities. In addition, the HBPP PSRC performs special reviews, investigations or analyses, and screens subjects of special concern as requested by NSOC. HBPP PSRC functions, responsibilities, and meeting requirements are described in Section 17.2.

Administrative procedures or charters for the above committees or programs provide detailed responsibilities and functions, as well as membership, authority, and reporting requirements. The reporting relationships of the committee are identified in the organization chart on Figure 17.1-2.

Verification of conformance to established requirements (except designs) is accomplished by individuals or groups within QV who do not have direct responsibility for performing the work being verified or by individuals or groups trained and qualified in QA concepts and practices and independent of the organization responsible for performing the task. The persons and organizations performing QA and quality control functions have direct access to management levels that assure the ability to: (a) identify quality problems; (b) initiate, recommend, or provide solutions through designated channels; and (c) verify implementation of solutions. They are sufficiently free from direct pressures for cost and schedule and have the responsibility to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material. (The organizational positions with stop work authority are identified in the implementing procedures.) QV reviews and documents concurrence with all procedures and instructions that define methods for implementing the QA Program.

Each organization that supports the ISFSI documents and maintains current a written description of its internal organization. This documentation describes the business unit or department's structure, levels of authority, lines of communication, and assignments of responsibility. Such documentation takes the form of organization charts supported by written job descriptions or other narrative material in sufficient detail that the duties and authority of each individual whose work affects quality is clear. Interfaces between organizations are described in administrative procedures or other documents controlled in accordance with the appropriate requirements of Section 17.6.

The individuals assigned to the positions having a particular responsibility in program directives and administrative procedures (as described above) are the only individuals who are authorized to perform these activities. However, circumstances may arise where it is considered either necessary or desirable to have such activities, or some portion of them, actually performed by someone else. In such cases, the assigning organization retains responsibility and shall verify that the procedures and instructions to be followed in performing the work are adequate for controlling the work and meet applicable requirements. In such circumstances, the detailed procedures and instructions to be followed in performing the work are reviewed and approved by the person assigned responsibility for the work prior to the commencement of work. The purpose of such a review and approval is to verify that such procedures and instructions reflect an acceptable method of performing the work and are in compliance with the requirements of the QA Program. All instances in which authority is to be delegated or support services are to be

provided are documented.

ISFSI suppliers are required to conform to this QA Program or to their own program approved by PG&E. Supplier QA Programs are required to comply with the applicable portions of both 10 CFR 50, Appendix B, and 10 CFR 72, Subpart G, and the applicable regulatory documents and industry standards identified in Table 17.1-1. The quality program is defined in the contract or similar procurement document. Suppliers to PG&E are required to document their internal organizational arrangements to the extent necessary for PG&E to assure the supplier is capable of effectively managing, directing, and executing the requirements of the procurement documents. The authority and responsibility of persons and organizations who perform activities that might affect the quality of the procured items or services shall be clearly established. The Suppliers' organizational structure, levels of authority, and functional assignments of responsibility shall be such that:

- (1) The QA function of formally verifying conformance to the technical and quality requirements of the procurement documents is accomplished by qualified personnel who are independent of those who performed or directly supervised the work.
- (2) Personnel who perform QA functions have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; to verify implementation of those solutions; and to control further processing of the items or services until proper dispositioning has occurred.

17.2 QUALITY ASSURANCE PROGRAM

17.2.1 PROGRAM APPLICABILITY

The quality of the important-to-safety aspects related to the design, fabrication, construction, testing, operation, maintenance, modification, and decommissioning of the Humboldt Bay ISFSI structures, systems, and components (SSCs) shall be assured through the QA Program prescribed herein, quality-related program directives, and administrative procedures. The QA Program requirements, as a minimum, apply to the HB ISFSI SSCs classified as important to safety in the HB ISFSI FSAR Update, Section 4.5. The applicable QA criteria are executed to an extent that is commensurate with the importance to safety.

The QA Program also applies to the following:

- (1) Managerial and administrative controls to ensure safe operation of the ISFSI, both prior to issuance of a license and throughout the life of the licensed activity.
- (2) Activities that provide confidence that an ISFSI SSC will perform satisfactorily in service, including activities that determine that physical characteristics and quality of materials or components adhere to predetermined requirements.

In addition, the QA Program includes requirements that apply to the following ISFSI nonsafety-related programs:

- (1) Emergency Preparedness
- (2) Security
- (3) Radiation Protection
- (4) ISFSI Radiological Environmental Monitoring
- (5) Radioactive Waste Management

17.2.2 PROGRAM CONTROL

The status and adequacy of this QA Program shall be regularly monitored, and it shall be revised as necessary to improve its effectiveness or to reflect changing conditions.

The Director - Quality Verification (QV), is responsible for the preparation, issue, interpretation, and control of this QA Program, and for concurring with changes to

quality-related administrative procedures that propose a change to the QA Program as it is described in a commitment to a regulatory agency. The Director - QV, is responsible to assure the requirements set forth in this QA Program, quality-related program directives, and administrative procedures are in compliance with current regulatory requirements and PG&E commitments to the NRC as shown in Table 17.1-1. Proposed changes to program directives are also approved by the Senior Vice President - Generation and Chief Nuclear Officer.

The QA Program documents, including any changes, supplements, or appendices, are issued and maintained as controlled documents. Changes to the HB ISFSI-specific QA Program requirements shall be made in accordance with 10 CFR 50.54. Proposed changes to this QA Program that reduce commitments are reviewed and concurred with in writing by the Director - QV, and are approved by the Senior Vice President - Generation and Chief Nuclear Officer, or his designee, prior to being submitted to and approved by the NRC in accordance with 10 CFR 50.54 prior to issue for use.

Implementation of the QA Program is accomplished through separately issued procedures, instructions, and drawings. Each vice president, director, and manager is responsible for the establishment and implementation of detailed procedures and instructions prescribing the activities for which he is responsible. Such documents are derived from the requirements and reflect the responsibilities specified in the QA Program. Activities affecting quality are accomplished in accordance with these instructions, procedures, and drawings. All personnel are instructed that compliance with those requirements; and the requirements of the QA Program, is mandatory.

Questions or disputes involving interpretations of QA Program requirements, or of the commitments and requirements upon which it is based, are referred to the Director - QV, for resolution. Questions or disputes involving the responsibilities defined in this chapter and program directives are referred to the Senior Vice President - Generation and Chief Nuclear Officer. Questions or disputes involving other quality matters are resolved by referring the matter in a timely manner to successively higher levels of management until, if necessary, the matter reaches that level which has direct authority over all contesting parties.

Personnel who perform functions addressed by the QA Program are responsible for the quality of their work. They are indoctrinated, trained, and appropriately qualified to assure that they have achieved and maintained suitable proficiency to perform those functions. Qualifications of such personnel are in accordance with applicable codes, standards, and regulatory requirements.

The Director - QV, or his designated representative, regularly reports to the Senior Vice President - Generation and Chief Nuclear Officer, responsible company management, and NSOC on the effectiveness of the QA Program as it relates to ISFSI design,

maintenance, and operation. Such reports are based on the results of audits, reviews, inspections, tests, and other observations of activities as prescribed by the QA Program.

Annually, the Director - QV, shall report to the Senior Vice President - Generation and Chief Nuclear Officer, on the effectiveness of the QA Program and results of the Audit Program. The report shall include an evaluation of compliance with current regulatory requirements and commitments to the NRC.

17.2.3 INDEPENDENT REVIEW PROGRAM

The QA Program also includes an independent review, implemented by NSOC. This function provides an independent review of ISFSI changes, tests, and procedures, which constitute a change to the ISFSI as described in the HB ISFSI FSAR Update. In addition, the independent review function will verify that reportable events are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not appear to a day-to-day observer.

The individuals assigned responsibility for independent reviews shall be qualified in specific disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

- (1) ISFSI operations
- (2) Nuclear engineering
- (3) Chemistry and radiochemistry
- (4) Metallurgy
- (5) Nondestructive testing
- (6) Instrument and control
- (7) Radiological safety
- (8) Mechanical and electrical engineering
- (9) Administrative controls
- (10) Quality assurance practices

(11) Other appropriate fields

NSOC shall report to and advise the Senior Vice President - Generation and Chief Nuclear Officer, on those areas of responsibility specified in the sections below.

Composition – NSOC membership shall be comprised of site representatives and offsite members. Membership will normally include the Director of Quality Verification and four members. The NSOC Chair shall have a minimum of 6 years of professional level managerial experience in the power field and NSOC members shall have a minimum of 5 years of professional level experience in the power field.

The NSOC Chair and all members shall have qualifications that meet or exceed the requirements and recommendations of Section 4.7 of ANSI/ANS 3.1 1978.

An individual may possess competence in more than one specialty area.

Consultants: Consultants shall be used as determined by the NSOC Chair to provide expert advice to NSOC.

Meeting Frequency: NSOC shall meet at least twice a year.

Quorum: A quorum of NSOC is necessary for the performance of the NSOC function required by the QA Program. The quorum shall be a majority of the members, and shall include the Chair (or appointed Vice-Chair).

Review: NSOC shall review:

- (1) The evaluations for: (a) changes to procedures, equipment, or systems, and (b) tests or experiments completed under the provision of 10 CFR 50.59 or 10 CFR 72.48, to verify that such actions did not require prior NRC approval
- (2) Proposed changes to procedures, equipment, or systems, that require prior NRC approval as defined in 10 CFR 50.59 or 10 CFR 72.48
- (3) Proposed tests or experiments that require prior NRC approval as defined in 10 CFR 50.59 or 10 CFR 72.48
- (4) Proposed changes to the HB ISFSI Technical Specifications or license

- (5) Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance
- (6) Significant operating abnormalities or deviations from normal and expected performance of ISFSI equipment that affect nuclear safety
- (7) All reportable events
- (8) All recognized indications of an unanticipated deficiency in some aspect of ISFSI design or operation of important-to-safety SSCs that could affect nuclear safety
- (9) Reports and meeting minutes of the PSRC.
- (10) Any other matter involving safe operation the ISFSI that the quality verification director deems appropriate for consideration, or which is referred to the director by organizational units.

NSOC may delegate reviews of selected topics such as changes processed under 10 CFR 50.59 and 10 CFR 72.48 to QV. The appropriate NSOC subcommittee will consider QV's reviews of those topics in their meetings.

Records: Records of NSOC reviews and activities shall be prepared, approved, and distributed as indicated below:

- (1) A summary report shall be prepared, approved, and forwarded to the Senior Vice President - Generation and Chief Nuclear Officer and the Plant Manager - Humboldt Bay Nuclear.
- (2) Minutes of each NSOC meeting shall be prepared, approved, and forwarded to the Senior Vice President - Generation and Chief Nuclear Officer, within 30 days following each meeting

17.2.4 PLANT STAFF REVIEW COMMITTEE

A PSRC has been established for the HB ISFSI. The committee satisfies applicable requirements of ANSI N18.7, 1976, and its activities are controlled as described below:

PSRC Function - The PSRC shall function to advise the Director and Plant Manager - Humboldt Bay Nuclear, on all matters related to nuclear safety.

Composition - The PSRC shall be chaired by the Director and Plant Manager - Humboldt Bay Nuclear, or delegate, and shall be composed of members of the plant staff who have responsibility in the areas of ISFSI operations, mechanical maintenance, instrumentation and control maintenance; radiation protection, and nuclear engineering. The PSRC Chairman shall appoint all members in writing. Each PSRC member shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978, Section 4.7, for comparable positions, except for ISFSI operations and radiation protection. The radiation protection member shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987.

Alternates - The Chairman may designate in writing other regular members who may serve as the Acting Chairman of PSRC meetings. All alternate members shall be appointed in writing by the PSRC Chairman. Alternates may be designated for specific PSRC members and shall have expertise and qualifications in the same general area as the regular PSRC member they represent. No more than two alternates shall participate as voting members in PSRC activities at any one time.

Meeting Frequency - The PSRC shall meet at least once per calendar quarter and as convened by the PSRC Chairman or his designated alternate.

Quorum - The minimum quorum of the PSRC necessary for performance of the PSRC responsibility and authority provisions of this QA Program shall be a majority (more than one-half) of the members of the PSRC. For purposes of the quorum, this majority shall include the Chairman or the acting chairman, and no more than two alternate members.

The PSRC shall be responsible for:

- (1) Reviewing the documents listed below to verify that proposed actions do not require prior NRC approval or require a change to the Technical Specifications and recommending approval or disapproval in writing to the appropriate approval authority
 - (a) Evaluations of proposed procedures and procedure changes completed under the provisions of 10 CFR 50.59 or 10 CFR 72.48
 - (b) Evaluations of proposed tests or experiments completed under the provisions of 10 CFR 50.59 or 10 CFR 72.48
 - (c) Evaluations of proposed changes or modifications to plant structures, systems, or equipment completed under the provisions of 10 CFR 50.59 or 10 CFR 72.48

- (d) Evaluations of proposed changes to the following plans and programs completed under the provisions of 10 CFR 50.59, 10 CFR 72.48, or other applicable regulations:
1. Security Plan
 2. Emergency Plan
- (2) Reviewing all proposed changes to the ISFSI Technical Specifications and advising the Director and Plant Manager - Humboldt Bay Nuclear, on their acceptability
- (3) Investigating all violations of the ISFSI Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Senior Vice President - Generation and Chief Nuclear Officer. The assessment shall include an assessment of the safety significance of each violation
- (4) Reviewing all reportable events in order to advise the Director and Plant Manager - Humboldt Bay Nuclear, on the acceptability of proposed corrective actions, and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Senior Vice President - Generation and Chief Nuclear Officer.
- (5) Reviewing significant ISFSI operating experience or events that may indicate the existence of a nuclear safety hazard, and advising the Director and Plant Manager - Humboldt Bay Nuclear, on an appropriate course of action.
- (6) Reviewing the Security Plan and implementing procedures and submitting results and recommended changes to the Director and Plant Manager - Humboldt Bay Nuclear.
- (7) Reviewing the Emergency Plan and implementing procedures and submitting results and recommended changes to the Director and Plant Manager - Humboldt Bay Nuclear.
- (8) Reviewing any accidental, unplanned, or uncontrolled radioactive release including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence to the Senior Vice President - Generation and Chief Nuclear Officer.
- (9) Recommending in writing to the appropriate approval authority, approval or disapproval of the items considered under paragraphs (1) and (2), above.

- (10) Rendering determinations in writing with regard to whether each item considered under paragraphs (1) through (4), above, require prior NRC approval.
- (11) For HB ISFSI, in the event of a disagreement between PSRC members on a matter affecting nuclear or radiological safety, a conservative course shall be followed as determined by the Director and Plant Manager - Humboldt Bay Nuclear. Records of such disagreements shall be included in the meeting minutes.
- (12) Reviewing, prior to approval, new procedures used to handle heavy loads in exclusion areas and changes directly related to methods and routes used to handle heavy loads in exclusion areas.

Records - The PSRC shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of this QA Program section. Copies shall be provided to the Senior Vice President - Generation and Chief Nuclear Officer, and to the quality verification director.

17.3 DESIGN CONTROL

Design activities shall be performed in an orderly, planned, and controlled manner directed to achieving the independent spent fuel storage installation (ISFSI) design that best serves the needs of PG&E and its customers without posing an undue risk to the health and safety of the public.

Design activities shall be controlled to assure that design, technical, and quality requirements are correctly translated into design documents and that changes to design and design documents are properly controlled. Design control procedures shall address responsibilities for all phases of design including:

- (1) Responsibilities
- (2) Interface control
- (3) Design input
- (4) Design performance
- (5) Design verification
- (6) Design change

Systematic methods shall be established and documented for communicating needed design information across the external and internal design interfaces, including changes to the design information, as work progresses. The interfaces between the HB ISFSI engineering organization and other organizations, either internal or external to PG&E, performing work affecting quality of design shall be identified and documented. This identification shall include those organizations providing criteria, designs, specifications, technical direction, and technical information and shall be in sufficient detail to cover each structure, system, or component (SSC) and the corresponding design activity.

Provisions for design input shall define the technical objectives for SSCs being designed or analyzed. For the SSC being designed, or for the design services being provided (for example, design verification), design input requirements shall be determined, documented, reviewed, approved, and controlled.

Required design analyses (such as physics, stress, thermal, hydraulic, and accident analysis; material compatibility; accessibility for inservice inspection, maintenance, and repair; and ALARA considerations) shall be performed in a planned, controlled, and correct manner. PG&E procedures shall identify the review and approval responsibilities for design analyses.

The preparation and control of design documents (such as specifications, drawings, reports, and installation procedures) shall be performed in a manner to assure design inputs are correctly translated into design documents (for example, a documented check to verify the dimensional accuracy and completeness of design drawings and specifications).

PG&E shall provide for reviewing, confirming, or substantiating the design to assure that the design meets the specified design inputs. Design verification shall be performed by competent individuals or groups other than those who performed the original design, but who may be from the same department. Individuals performing the verification shall not:

- (1) Have immediate supervisory responsibility for the individual performing the design. In exceptional circumstances, the designer's immediate supervisor can perform the verification provided:
 - (a) The supervisor is the only technically qualified individual
 - (b) The need is individually documented and approved in advance by the supervisor's management
 - (c) Quality assurance audits cover frequency and effectiveness of use of supervisors as design verifiers to guard against abuse
- (2) Have specified a singular design approach
- (3) Have ruled out certain design considerations
- (4) Have established the design inputs for the particular design aspect being verified

The results of the design verification efforts shall be documented with the identification of the verifier clearly provided. Design verification methods may include, but not be limited to, the following: design reviews, use of alternate calculations, and qualification testing. The design verification shall be identified and documented. The design verification shall be completed prior to relying upon the component system or structure to perform its function. Procedures shall assure that verified computer codes are certified for use and that their applicability is specified.

Proposed changes or modifications to ISFSI systems or equipment that affect nuclear safety shall be designed by a qualified individual or organization, and reviewed by a qualified individual/group other than the individual/group who prepared the change or modification, but who may be from the same organization. These reviews shall include a determination as to whether additional cross-discipline reviews are necessary. If deemed necessary, they shall be performed by review personnel of the appropriate discipline(s). These reviews shall also determine whether an evaluation per 10 CFR 50.59 or 10 CFR 72.48 is necessary. If necessary, one shall be prepared

and presented to the PSRC for review prior to approval.

Each Humboldt Bay ISFSI change or modification shall be approved by the Director and Plant Manager - Humboldt Bay Nuclear, or designee, as specified in administrative procedures, prior to implementation.

Procedures for implementing design changes, including field changes, shall assure that the impact of the change is carefully considered, required actions documented, and information concerning the change transmitted to all affected persons and organizations. These changes shall be subjected to design control measures commensurate with those applied to the original design. Design changes shall be reviewed and approved by the same organization or group that was responsible for the original design.

Document control measures shall be established for design documents that reflect the commitments of the HB ISFSI FSAR Update. These design documents shall include, but are not limited to, specifications, calculations, computer programs, system descriptions, the HB ISFSI FSAR Update when used as a design document, and drawings including flow diagrams, piping and instrument diagrams, control logic diagrams, electrical single line diagrams, structural drawings for major facilities, site arrangements, and equipment locations.

Nonconforming activities such as procedure violations, deviations, or errors and deficiencies in approved design documents, including design methods (such as computer codes), shall be controlled as described in Sections 17.15 and 17.16.

17.4 PROCUREMENT DOCUMENT CONTROL

The procurement documents shall include those requirements necessary to assure that the items and services to be provided will be of the desired quality.

The procurement documents shall also include provisions for the following, as appropriate:

- (1) Basic Technical Requirements - These include drawings, specifications, codes, and industrial standards with applicable revision data; test and inspection requirements; and special instructions and requirements, such as for designing, fabricating, cleaning, erecting, packaging, handling, shipping, and, if applicable, extended storage in the field.
- (2) Quality Assurance Requirements - These include the requirements for the supplier to have an acceptable QA Program; provisions for access to the supplier's facilities and records for source inspection and audit when the need for such inspection and audit has been determined; and provisions for extending applicable QA Program and other requirements of procurement documents to subcontractors and suppliers, including PG&E's access to facilities and records.
- (3) Documentation Requirements - These shall include records to be prepared, maintained, submitted or made available for review and instructions on record retention and disposition.

The procedures that implement procurement document control shall describe the organizational responsibilities for procurement planning; preparation, review, approval and control of procurement documents; supplier selection; bid evaluations; and review and evaluation of supplier QA Programs prior to initiation of activities affected by the program.

Procedures shall be established to review the adequacy of technical and quality assurance requirements stated in procurement documents; determine that requirements are correctly stated, inspectable, and controllable; assure adequate acceptance and rejection criteria; and provide for the preparation, review, and approval of procurement documents in accordance with QA Program requirements. The review and documented concurrence of the adequacy of quality assurance requirements stated in procurement documents shall be performed by independent personnel trained and qualified in applicable QA practices and concepts.

Changes to procurement documents shall be subject to the same control as the original document.

17.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting quality shall be prescribed by and accomplished in accordance with documented procedures, instructions, and drawings.

The vice president in charge of each PG&E organizational unit that performs activities affecting quality is responsible for the establishment and implementation of instructions, procedures, or drawings prescribing such activities. Standard guidelines for the format, content, and review and approval processes shall be established and set forth in a procedure or instruction issued by that organizational unit.

The method of performing activities affecting quality shall be prescribed in documented instructions, procedures, or drawings of a type appropriate to the circumstances. This may include shop drawings, process specifications, job descriptions, planning sheets, travelers, QA manuals, checklists, or any other written or pictorial form provided that the activity is described in sufficient detail such that competent personnel could be expected to satisfactorily perform the work functions without direct supervision.

Within the constraints, limitations, or other conditions as may be imposed by the independent spent fuel storage installation (ISFSI) Technical Specifications and other license requirements or commitments, procedures prescribing a preplanned method of conducting the activities and programs specified shall be established in accordance with the applicable regulations, codes, standards, and specifications.

In addition to the above, ISFSI procedures and programs shall be established and controlled as described below.

- (1) Written procedures shall be established, implemented, and maintained covering the activities referenced in the HB ISFSI Technical Specifications.
- (2) Each procedure of paragraph (1) above, and changes thereto, and all proposed tests or experiments that affect nuclear safety shall be reviewed and approved prior to implementation in accordance with the review and approval requirements below. Each procedure of paragraph (1) above, as modified by Table 17.1-1, shall also be reviewed periodically as set forth in administrative procedures.

These procedure review and approval requirements apply when approving ISFSI programs and procedures, or changes to ISFSI programs and procedures. They also apply when approving or changing corporate procedures and procedures used by support organizations if they could have an immediate effect on ISFSI operations or the operational status of ISFSI SSCs that are important to safety. They do not apply to editorial or typographical changes.

- (3) Each procedure or program required by paragraph (1) above, and other procedures, tests, and experiments that affect nuclear safety or the treatment of radwaste, and changes thereto, shall be prepared by a qualified individual/group. Each procedure, program, test, or experiment, and changes thereto, shall be reviewed by an individual/group other than the individual/group who prepared the proposed document or change, but who may be from the same organization as the individual/group who prepared it. The Director and Plant Manager - Humboldt Bay Nuclear, or his designee, shall approve Humboldt Bay ISFSI procedures prior to implementation, as identified in administrative procedures.
- (4) A responsible organization shall be assigned for each program or procedure required by paragraph (1) above. The responsible organization shall assign reviews of proposed procedures, programs, and changes to qualified personnel of the appropriate discipline(s).
- (5) Individuals responsible for the above reviews shall be knowledgeable in the document's subject area, shall meet or exceed the qualification requirements of Section 4.7.2 of ANSI/ANS 3.1-1978, and shall be designated as qualified reviewers by the Director and Plant Manager - Humboldt Bay Nuclear for Humboldt Bay ISFSI procedures.
- (6) The reviews specified in paragraph (2) above shall include a determination as to whether additional cross-discipline reviews are necessary. If deemed necessary, they shall be performed by review personnel of the appropriate discipline(s).
- (7) The reviews specified in paragraph (2) above shall also determine whether an evaluation per 10 CFR 50.59 or 10 CFR 72.48 is necessary. If necessary, one shall be prepared and presented to the PSRC for review prior to approval.
- (8) Temporary changes to procedures of paragraph (1) above may be made provided:
 - (a) The intent of the original procedure is not altered
 - (b) Administrative controls for approval and timely notification or training of personnel affected by the temporary change shall be implemented.
 - (c) The change is documented, reviewed as described above, and approved by the appropriate approval authority within 14 days of implementation.

17.6 DOCUMENT CONTROL

Documents and changes to documents that prescribe or verify activities affecting quality shall be controlled in a manner that precludes the use of inappropriate or outdated documents. As a minimum, controlled documents include: design documents, including documents related to computer codes; procurement documents; instructions and procedures for such activities as fabrication, construction, modification, installation, test, operation, maintenance, and inspection; as-built documents; quality assurance and quality control manuals and quality-affecting procedures; Humboldt Bay Independent Spent Fuel Storage Installation FSAR Updates; and nonconformance reports.

The organization responsible for establishing instructions, procedures, drawings, or other documents prescribing activities affecting quality is also responsible to develop and implement systematic methods for the control of such documents in accordance with the requirements herein. In those instances where such documents directly involve organizational interfaces, that organization with ultimate responsibility for the issuance of the documents is responsible for establishing the methods for their control.

Procedures and instructions shall assure that documents, including changes, are prepared; reviewed by a qualified individual other than the person who generated the document; approved for release by authorized personnel; distributed to the location where the activity is performed prior to commencing work; and used in performing the activity. Procedures and instructions shall require the development of as-built drawings and the removal or appropriate identification of obsolete or superseded documents.

Procedures and instructions that define methods for implementing the QA Program requirements shall be reviewed and concurred with by quality verification (QV), for compliance and alignment with the Program. Revisions to these documents shall also be reviewed and concurred with by QV if they propose a change to the QA Program as it is described in a commitment to a regulatory agency.

The controls shall identify those responsible for preparing, reviewing, approving, and issuing documents to be used. They shall also define the coordination and control of interfacing documents and shall require the establishment of current and updated distribution lists.

A document control system shall be established to identify the current revision of instructions, procedures, specifications, drawings, and procurement documents. Master lists, when utilized as an element of the document control system, shall be updated and distributed to predetermined responsible personnel.

17.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Supplier activities in providing purchased material, equipment, and services shall be monitored as planned and necessary to assure such items and services meet procurement document requirements.

Procedures shall describe each organization's responsibilities for the control of purchased material, equipment, and services, including the interfaces between all affected organizations.

All materials, equipment, and services shall meet the specified technical and quality requirements. Verification that a supplier can meet the specified technical and quality requirements shall be by one or a combination of the following:

- (1) Evaluation of the supplier's history
- (2) Evaluation of current supplier quality records
- (3) Evaluation of the supplier's facilities, personnel, and implementation of a QA Program

Such evaluations shall be documented. Suppliers whose QA Programs have been found by quality verification (QV), to satisfy specified quality requirements shall be listed on the PG&E Qualified Suppliers List, which is controlled by QV.

Suppliers of commercial grade calibration services may be qualified based on their accreditation by a nationally-recognized accrediting body, as an alternative to qualification by supplier audit, commercial grade survey, or in-process surveillance.

A documented review of the suppliers' accreditation by the purchaser may be used as the qualification method, as described in PG&E commitments to NRC Regulatory Guides 1.123 and 1.144, which are documented in Table 17.1-1. This review shall include, at a minimum, all of the following:

- (1) The accreditation is to ANSI/ISO/IEC 17025
- (2) The accrediting body is either the National Voluntary Laboratory Accreditation Program (NVLAP) or an accrediting body recognized by NVLAP through a Mutual Recognition Agreement (MRA).
- (3) The published scope of accreditation for the calibration laboratory covers the needed measurement parameters, ranges, and uncertainties.

A quality verification plan shall be established and documented that applies to each

procurement and identifies the manner by which PG&E intends (with appropriate QV organization involvement) to assure the quality of the material, equipment, or service as defined in the procurement documents and to accept those items or services from the supplier.

The quality verification plan shall identify inspection, audit, and/or surveillance activities to be performed including the characteristics or processes to be witnessed, inspected, or verified; the method of surveillance; and the extent of documentation required. The timing and sequence of the activities shall be planned to identify any system or product deficiencies before subsequent activities may preclude their disclosure.

The plan shall also be based on consideration of:

- (1) Importance to independent spent fuel storage installation safety
- (2) Complexity of inspectable characteristics
- (3) Uniqueness of the item or service

Supplier performance and compliance with procurement documents may be monitored by either source verification, receiving inspection, or a combination of the two. Source verification activities may consist of inspections, audits, surveillance, or a combination thereof and are conducted at the supplier's facility. When source verification activities are specified in the quality verification plan, the timing and sequence of these activities are to be delineated.

Receiving inspection activities, as required by the quality verification plan, shall be coordinated with source verification activities performed prior to shipments. If sampling is performed, it shall be in accordance with procedures and/or recognized standards. Receipt inspection shall include a review which verifies that supplier quality records required by procurement documents are acceptable and that items are properly identified and traceable to appropriate documentation.

Records of quality verification activities shall be traceable to the materials, equipment, or services to which they apply. Documentation of acceptance in accordance with the procurement quality verification plan shall be available at the site prior to installation or acceptance for use. Documentary evidence that procurement document requirements have been met shall clearly reflect each requirement. Supplier's Certificates of Conformance are periodically evaluated by audits and independent inspections or tests to assure they are valid and the results documented.

When spare or replacement parts are procured, supplier selection and quality verification activities shall be planned and implemented to verify compliance with requirements meeting or exceeding those of the original.

17.8 IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Materials, parts, and components shall be identified and controlled in a manner to preclude the use of incorrect or defective items.

All materials, parts, and components, including partially fabricated subassemblies, batches, lots, and consumables, shall be identified in a manner that each can be related to its applicable drawing, specification, or other technical documentation at any stage from initial receipt through fabrication, installation, repair, or modification. Controls and implementing procedures shall ensure that only correct and accepted items are used during all stages and describe the responsibilities of the involved organizations.

Physical identification of items shall be used whenever possible and practical. Controls may, however, be through physical separation, procedure, or other appropriate means. Identification may be either on the item or on records traceable to the item.

Identification marking, where employed, shall be clear, unambiguous, and indelible and its application shall not impair the function of the identified item or any other item. When an item is subdivided, the identifying marking shall be transferred to each resulting part. Markings shall not be rendered illegible by treatment, process, assembly, installation, or coating unless other means of identification and determining acceptability are provided.

Verification activities, such as inspection, shall be performed to ensure that the provisions of this policy and related implementing procedures are followed for items prior to release for fabrication, assembly, shipping, installation, and use.

When required by code, standard, or specification, traceability of materials, parts, or components to specific inspection or test records shall be provided for and verified.

17.9 SPECIAL PROCESSES

Special processes shall be controlled and performed by qualified personnel using qualified procedures or instructions in accordance with applicable codes, standards, specifications, criteria, or other special requirements.

A special process is an activity in which the quality of the result is highly dependent upon either process variables or the skill and performance of the person doing the work, and the specified quality is difficult to verify by inspection and test after the process is completed.

Special processes include, but are not limited to:

- (1)Welding
- (2)Heat treating
- (3)Nondestructive examination
- (4)Chemical cleaning
- (5)Others as specified in design and procurement documents (examples are certain protective coating applications and concrete batch plant operations, which are controlled by specifications on a case-by-case basis)

The implementing instructions shall contain the criteria for assuring proper process control and shall be qualified and controlled to assure compliance with applicable codes, standards, QA procedures, and design specifications. Substantiating records of qualifications and controls shall be maintained.

17.10 INSPECTION

A comprehensive program of inspection of items and activities affecting quality shall be conducted to verify conformance with established requirements. Procedures shall describe the organizational responsibilities necessary to carry out the inspection program.

The objective of the inspection program shall be to verify the quality of the items and activities and conformance to the applicable documented instructions, procedures, and drawings for accomplishing activities affecting quality. The inspection program, including information relative to individual inspections to be performed, shall be developed based on a review of the design drawings, specifications, and other controlled documents which prescribe items and activities affecting quality. Inspections shall be performed utilizing appropriate inspection procedures and instructions together with the necessary drawings, specifications, and other controlled documents. The inspections shall be documented and evaluated.

Inspection procedures, instructions, or checklists shall provide for the following: identification of characteristics and activities to be inspected; a description of the method of inspection; identification of the individuals or groups responsible for performing the inspection operation; acceptance and rejection criteria; identification of required procedures, drawings, and specifications and revisions; recording the name of the inspector or data recorder and the results of the inspection operation; and specifying necessary measuring and test equipment including accuracy requirements. The inspection program shall include, but not be limited to, those inspections required by applicable codes, standards, specifications, and Independent Spent Fuel Storage Installation (ISFSI) Technical Specifications.

The inspection program shall require inspection of ISFSI modifications, repairs, and replacements to be in accordance with existing design requirements.

The inspection program shall require inspection and/or test of items for each work operation where such is necessary to assure quality. If inspection of processed items is impossible or disadvantageous, indirect control by monitoring of process shall be required. Both inspection and process monitoring shall be required when control is inadequate without both. Both inspection and process control shall be performed when required by applicable code, standard, or specification.

Mandatory quality control inspection hold points shall be identified in the inspection program. When required, the specific hold points shall be indicated in the drawings, procedures, or instructions that prescribe the work activity. Work shall not proceed beyond such hold points without the documented consent of Quality Verification.

When the inspection program permits or requires a sample of a large group of items that are amenable to statistical analysis, the sampling procedures to be used shall be based on recognized standard practices.

Inspections to verify the quality of work shall be performed by qualified individuals other than those who performed or directly supervised the activity being inspected. During the inspection, such persons shall not report directly to the immediate supervisors who are responsible for the work being inspected.

Personnel performing inspections shall be qualified in accordance with applicable regulations, codes, standards, and specifications.

Inspection records shall contain the following where applicable: a description of the type of observation, the date and results of the inspection, information related to conditions adverse to quality, inspector or data recorder identification, evidence as to the acceptability of the results, and action taken to resolve any discrepancies noted.

17.11 TEST CONTROL

A program of testing shall be conducted as necessary to demonstrate that structures, systems, and components will perform satisfactorily in service. This program shall ensure that the necessary testing is identified and performed at the appropriate time in accordance with written test procedures that incorporate or reference the requirements and acceptance limits contained in the applicable design documents.

The program shall cover all required tests, including tests prior to installation, preoperational tests, and operational tests.

The procedures that implement testing shall provide for meeting appropriate prerequisites for the test (for example, environmental conditions, specification of instrumentation, and completeness of tested item), sufficient instruction for the performance of the test, specification of any witness or hold points, acceptance and rejection criteria and limits, and the documentation of the test. The procedures shall provide for evaluation and documentation of the test results and data and their acceptability as determined by a qualified person or group.

Test records shall contain the following where applicable: a description of the type of observation, the date and results of the test, information related to conditions adverse to quality, inspector or data recorder identification, evidence as to the acceptability of the results, and action taken to resolve any discrepancies noted.

17.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Organizational responsibilities shall be delineated for establishing, implementing, and assuring the effectiveness of the calibration program for measuring and test equipment (M&TE). This program shall include the generation, review, and documented concurrence of calibration procedures; the calibration of measuring and test equipment; and the maintenance and use of calibration standards.

M&TE, including reference standards, used to determine the acceptability of items or activities shall be strictly maintained within prescribed accuracy limits.

M&TE, including reference standards, shall be of suitable range, type, and accuracy to verify conformance with requirements.

Procedures for control of M&TE shall provide for the identification (labeling, codes, or alternate documented control system), recall, and calibration (including documented precalibration checks) of the M&TE. The calibration procedures shall delineate any necessary environmental controls, limits, or compensations in excess of those which may be inherent to the general program.

The calibrations shall utilize documented valid relationships to nationally recognized standards or accepted values of natural physical constants. Where national standards do not exist, the basis for the calibration shall be documented. Calibration of M&TE shall be against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated or, when this is not practical, have an accuracy that assures the equipment being calibrated will be within required tolerance and that the basis of acceptance is documented and authorized by responsible management of the PG&E organization performing that activity.

Calibrating standards have greater accuracy than standards being calibrated. Calibrating standards with the same accuracy may be used if it can be shown to be adequate for the requirements and the basis of acceptance is documented and authorized by responsible management.

The calibration intervals, whether calendar- or usage-based, shall be predetermined and documented. Indication of expiration, if feasible, will be displayed on or with the M&TE. Significant environmental or usage restrictions will be indicated on or with the equipment or be factored into the documented system used to control the issuance of the M&TE. Special calibration shall be required whenever the accuracy of the equipment is suspect.

Records shall be maintained to show that established schedules and procedures for the calibration of the M&TE have been followed. M&TE shall be identified and traceable to the calibration test data. Records of the usage of the M&TE shall be maintained to facilitate corrective action in the event of the discovery of a deficiency concerning the calibration or use of M&TE, so that measures may be taken and documented to determine the validity of previous inspections performed and of the acceptability of items inspected or tested since the previous calibration of the deficient M&TE.

17.13 HANDLING, STORAGE, AND SHIPPING

Material and equipment shall be handled, stored, and shipped in accordance with design and procurement requirements in a manner that will prevent damage, deterioration, or loss.

Special coverings, equipment, and protective environments shall be specified and provided where necessary for the protection of particular items from damage or deterioration. When such special protective features are required, their existence shall be verified and monitored as necessary to assure they continue to serve their intended function.

Special handling tools and equipment shall be provided where necessary to ensure items can be handled safely and without damage. Special handling tools and equipment shall be controlled and maintained in a manner such that they will be ready and fit to serve their intended function when needed. Such control shall include periodic inspection and testing to verify that special handling tools and equipment have been properly maintained.

Special attention shall be given to marking and labeling items during packaging, shipment, and storage. Such additional marking or labeling shall be provided as is necessary to ensure that items can be properly maintained and preserved. This shall include indication of the presence of special environments or the need for special control. Provisions shall be described for the storage of chemicals, reagents (including control of shelf life), lubricants, and other consumable materials.

Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by suitably trained individuals in accordance with predetermined work and inspection instructions.

17.14 INSPECTION, TEST, AND OPERATING STATUS

The inspection, test, and/or operating status of material, equipment, and operating systems shall be readily apparent and verifiable.

The procedures used to indicate status shall provide means for assuring that required inspections and tests are performed in the prescribed sequence; acceptability is indicated; and nonconforming items are clearly identified throughout fabrication, installation, test, maintenance, repairs, and modification to prevent inadvertent use or operation. Items accepted and released are identified to indicate their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work. Deviations from the prescribed sequence shall be subject to the same level of control as the generation of the original sequence to prevent the bypassing or omission of a required test or inspection.

Identification of status may be by such means as, but not limited to, tags, stamps, markings, labels, or travelers. In some instances, records traceable to the item may be used. The procedures implementing control of inspection, test, and operating status shall clearly delineate authority for the application, change, or removal of a status identifier.

17.15 CONTROL OF NONCONFORMING CONDITIONS

Items and activities that do not conform to requirements shall be controlled in a manner that will prevent their inadvertent use or installation. Technical decisions as to the disposition of each nonconforming condition shall be made by personnel with assigned authority in the relevant disciplines. The control, review, and disposition of nonconforming conditions shall be accomplished and documented in accordance with approved written procedures and instructions.

Nonconforming conditions shall be documented and affected organizations notified of such conditions. Further processing of the nonconforming conditions and other items affected by them shall be controlled in a manner to prevent their inadvertent use or installation pending a decision on their disposition.

The responsibility and authority for the disposition of nonconforming conditions shall be established and set forth in the applicable procedures and instructions for their control. The rework or repair of nonconforming items and the disposition of operational nonconforming conditions shall be accomplished in accordance with written procedures and instructions. Dispositions involving design changes shall be approved by the organization with the authority for design.

The acceptability of rework or repair of materials, parts, components, systems, or structures shall be verified by reinspecting and retesting the item as originally inspected and tested, or by a method that is at least equal to the original inspection or testing method. Reworked and repaired items shall be reinspected in accordance with applicable procedures and instructions. The acceptability of nonconforming items that have been dispositioned "repair" or "accept-as-is" shall be documented. Such documentation shall include a description of the change, waiver, or deviation that has been accepted in order to record the change and, if applicable, denote the as-built condition.

Corrective action for conditions adverse to quality shall be processed in accordance with Section 17.16.

In cases where required documentary evidence that items have passed required inspections and tests is not available, the associated materials or equipment shall be considered nonconforming. Until suitable documentary evidence is available to show that the material or equipment is in conformance, affected systems shall be considered to be inoperable and reliance shall not be placed on such systems to fulfill their intended safety functions.

Nonconforming conditions that require reporting to the NRC shall be reviewed by the Quality Verification organization. Such review shall include the results of any investigations made and the recommendations resulting from such investigations to preclude or reduce the probability of recurrence of the event or circumstance.

17.16 CORRECTIVE ACTION

Each individual condition adverse to quality shall be identified, controlled, and evaluated, and a disposition shall be determined for the remedial action and corrective action as soon as practicable. These activities shall be performed consistent with Section 17.15, Control of Nonconforming Conditions.

Systematic review and evaluation of all conditions adverse to quality shall be conducted and documented. Conditions adverse to quality shall include, but not be limited to: engineering, design, and drafting errors; equipment failures and malfunctions; abnormal occurrences; deficiencies; deviations; and defective material, equipment, and services.

The review and evaluation shall include identification of quality trends, repetitive occurrences, and significant conditions adverse to quality. The quality trends and other significant review findings shall be analyzed and appropriate corrective action determined. Findings and actual or recommended corrective action shall be reported to management by the responsible organization for review and assessment.

Significant conditions adverse to quality shall be investigated to the extent necessary to assess the root causes and to determine the corrective action required to prevent recurrence of the same or similar conditions. The corrective action required for significant conditions adverse to quality shall be accomplished in a timely manner.

Significant conditions adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to management.

Significant conditions adverse to quality that are related to Independent Spent Fuel Storage Installation (ISFSI) operations or maintenance shall be reported to the Quality Verification organization. Completion of corrective actions for significant conditions adverse to quality shall be reviewed and verified by personnel having no direct responsibility for either the disposition or the corrective action taken.

Follow-up reviews shall be conducted to verify that the corrective action was properly implemented, performed in a timely manner, and that it was effective in correcting the identified condition.

Significant conditions adverse to quality shall be evaluated for reportability to the NRC in accordance with 10 CFR 21, 10 CFR 72.74, and 10 CFR 72.75, the ISFSI Technical Specifications, and other applicable regulations and shall be reported as required.

17.17 QUALITY ASSURANCE RECORDS

Sufficient records shall be maintained to furnish evidence of both the quality of items and activities affecting quality and to meet applicable code, standard, and regulatory requirements. The records include all documents referred to or described in the QA Program or required by implementing procedures such as operating logs, maintenance and modification procedures, related inspection results, and reportable occurrences; and other records required by the independent spent fuel storage installation (ISFSI) Technical Specifications and Code of Federal Regulations. In addition to the records of the results of reviews, designs, fabrication, installation, inspections, calibrations, tests, maintenance, surveillances, audits, personnel qualification, special process qualification, and material analyses for PG&E quality-related activities and ISFSI structures, systems, and components that are important to safety, those of vendors, suppliers, subcontractors, and contractors shall also be maintained.

A management control system for the collection, storage, and maintenance of completed quality assurance (QA) records shall be maintained. This records management program shall be designed and implemented to assure that the QA records are complete, readily retrievable when needed, and protected from damage or destruction during storage by fire, flooding, theft, environmental conditions, or other causes.

QA records stored electronically will follow the guidance for electronic records management given in the Nuclear Information and Records Management Association (NIRMA) technical guidelines, TG 11-1998, "Authentication of Records;" TG 15-1998, "Management of Electronic Records;" TG 16-1998, "Software Configuration Management and Quality Assurance;" and TG 21-1998, "Electronic Records Protection and Restoration." QA records will be stored on electronic media (that is, optical disk, magnetic tape, network array, etc.) meeting the requirements of the NIRMA guidelines. Alternately, records stored on optical disks may meet the requirements of Generic Letter 88-18, "Plant Record Storage on Optical Disk," dated October 20, 1988. Information Systems will determine the appropriate electronic media. Regardless of the electronic media selected, the process must be capable of producing legible, accurate, and complete records during the required retention period.

Backup copies of in-process electronic media records will be maintained in multiple, physically-independent electronic locations. Backup copies of QA records in electronic media will be maintained in multiple, physically-independent electronic locations until such time as images of these records are created, copied, and verified on two copies of an appropriate electronic storage media. The two copies will then be stored in separate physical locations. File legibility verification will be completed on all QA records stored on electronic media by either visually verifying the file legibility or by electronically verifying exact binary file transfer.

Periodic media inspections to monitor image degradation will be conducted in accordance with the NIRMA guidelines or media manufacturers' recommendations. These periodic inspections shall be documented.

QA records stored on electronic media will be refreshed or copied on to new media and subsequently verified if the projected lifetime of that media does not exceed the retention period of the records stored on that media. These requirements meet the intent of Generic Letter 88-18.

Detailed records for items or activities shall be specified by instructions, procedures, drawings, or specification or other documents that prescribe the item or activity and shall be generated by the organization responsible for the item or activity including PG&E and non-PG&E organizations. Each department generating QA records is responsible for transmitting those records to the records processing organization for archival purposes.

All records shall be assigned a retention period in conformance with Title 10, Code of Federal Regulations, other applicable codes, standards, and specifications.

17.17.1 HUMBOLDT BAY ISFSI RECORDS

Important-to-safety records shall be classified as lifetime or nonpermanent. The following records shall be maintained as required for the Humboldt Bay ISFSI:

- (1) Radiation protection program and survey records
- (2) Records associated with reporting defects and noncompliance)
- (3) Records important to decommissioning
- (4) Records of changes to the physical security plan made without prior NRC approval
- (5) Records of changes, tests and experiments, and of changes to procedures described in the ISFSI FSAR Update pursuant to 10 CFR 72.48
- (6) Records showing receipt, inventory, location, disposal, acquisition, and transfer of spent fuel
- (7) A copy of the current inventory of spent fuel in storage at the ISFSI
- (8) A copy of the current material control and accounting procedures
- (9) Other records required by license conditions or by NRC rules, regulations or orders

- (10) Records of the occurrence and severity of important natural phenomena that affect ISFSI design
- (11) QA records (including records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety; and results of reviews, inspections, tests, audits, monitoring of work performance, and material analyses)
- (12) A copy of the current physical security plan, plus any superseded portions of the plan
- (13) A copy of the current safeguards contingency plan procedures, plus any superseded portions of the procedures
- (14) Operating records, including maintenance, alterations or additions made
- (15) Records of off-normal occurrences and events
- (16) Environmental survey records
- (17) Records of employee qualifications and certifications
- (18) Record copies of:
 - ISFSI FSAR Updates
 - Reports of accidental criticality or loss of special nuclear material
 - Material status reports
 - Nuclear material transfer reports
 - Reports of pre-operational test acceptance criteria and results
 - Procedures
 - Environmental Report
 - Emergency Plan
- (19) Construction Records; and
- (20) Records of events associated with radioactive releases.

Facilities for the temporary or permanent storage of completed QA records shall be established in predetermined locations as necessary to meet the requirements of codes, standards, and regulatory agencies. Such facilities shall be constructed and maintained so as to protect the contents from possible damage or destruction.

17.18 AUDITS

The adequacy and effectiveness of the Quality Assurance (QA) Program shall be continually monitored through a comprehensive system of internal and supplier audits. The audit system implemented by the Quality Verification (QV) organization includes all aspects of the QA Program. The audit system shall:

- (1) Verify, through examination and evaluation of objective evidence, that this QA Program has been implemented as required
- (2) Identify any deficiencies or nonconformances in this QA Program
- (3) Verify the correction of any identified deficiencies or nonconformances
- (4) Assess the adequacy and effectiveness of this QA Program

A comprehensive plan for the audit system shall be established and documented. Audit frequencies are determined by a performance-based evaluation plan. This plan uses assessment indicators to identify and schedule audits based on performance results and importance of the activity relative to safety. The plan shall identify the scope of individual audits that are to be performed, the aspects of this QA Program covered by each audit, and the schedule for performing audits. The audit system plan shall be reviewed at least semiannually, and revised as necessary, to assure that coverage and schedule reflect current activities and that audits of independent spent fuel storage installation (ISFSI) activities are being accomplished in accordance with applicable requirements. Other associated activities included as part of the audit program are: indoctrination and training programs; the qualification and verification of implementation of QA programs of contractors and suppliers; interface control among the applicant and the principal contractors; audits by contractors and suppliers; corrective action, calibration, and nonconformance control systems; ISFSI FSAR Update commitments; and activities associated with computer codes.

Auditors shall be independent of direct responsibility for the performance of the activities that they audit, have experience or training commensurate with the scope and complexity of their audit responsibility, and be qualified in accordance with applicable standards.

Auditing shall be initiated as early in the life of an activity as is practicable and consistent with the schedule for accomplishing the activity. In any case, auditing shall be initiated early enough to assure that this QA Program is effectively implemented throughout each activity. Individual audits shall be regularly scheduled on the basis of the status and importance of the activities, which they address.

For audits, other than those whose scheduled frequency is mandated by regulation (such as the Safeguards Contingency Plans or the Security Program), a grace period of up to 90 days may be utilized when the urgency of other priorities makes meeting the specified schedule dates impractical. For audit activities deferred by using a grace period, the next scheduled due date shall be based on the original schedule due date but may not exceed the original due date plus 90 days.

Audit reports shall be prepared, signed by the Audit Team Leader, and issued to responsible management of both the audited and auditing organizations.

Audits are regularly scheduled on a formal audit schedule prepared by QV. The audit schedule is reviewed regularly by the Director - QV, and the schedule is revised as necessary to assure adequate coverage as commensurate with activities and past performance. Audits are performed in accordance with approved audit plans. Such audits may be augmented by other QV assessments and independent inspections. Additional audits may be performed as requested by NSOC, the Senior Vice President - Generation and Chief Nuclear Officer, or the Director - QV.

The following areas shall be audited at least once per 24 months, or more frequently as performance dictates:

- (1) The conformance of ISFSI operation to provisions contained within the applicable Technical Specifications and applicable licenses
- (2) The performance, training, and qualifications of the entire ISFSI staff
- (3) The results of actions taken to correct deficiencies occurring in ISFSI equipment, structures, systems, or method of operation that affect nuclear safety
- (4) The performance of activities required by the QA Program to meet the criteria of Appendix B, 10 CFR 50
- (5) A representative sample of routine ISFSI procedures that are used more frequently than every two years. This audit is to ensure the acceptability of the procedures and to verify that the procedures review and revision program is being implemented effectively.
- (6) The performance of activities required to be audited by ANS-3.2/ANSI N18.7-1976, Section 4.5.
- (7) Review of design documents and process to ensure compliance with the Section 17.3 (i.e., use of supervisors as design verifiers). In addition, QV shall sample and review specifications and design drawings to assure that the documents are prepared, reviewed, and approved in accordance with PG&E procedures and that

the documents contain the necessary QA requirements, acceptance requirements, and quality documentation requirements.

- (8) QV shall audit the departments that qualify personnel and procedures to assure that the process qualification activity, records, and personnel meet the applicable requirements. They shall also audit the organizations implementing special processes to provide assurance that the processes are carried out in accordance with approved procedures by qualified personnel using qualified equipment and that required records are properly maintained.
- (9) The Radiation Protection Program, in accordance with 10 CFR 20.

The following activities shall be audited at least once per 12 months unless specified otherwise. However, if the audit frequencies required by the governing regulations are changed, audit frequencies shall at least meet the revised minimum requirements.

- (1) The Security Program in accordance with 10 CFR 73.55(g)(4) and 10 CFR 73.56(g)
- (2) The Humboldt Bay ISFSI Access Authorization Program.

Management of the audited organization shall review the audit report and respond to any quality problem reports, investigate any significant findings to identify their cause and determine the extent of corrective action required, including action to prevent recurrence. They shall schedule such corrective action and also take appropriate action to assure it is accomplished as scheduled. They shall respond to QV regarding each significant finding stating the root cause, immediate action taken, and the corrective action taken or planned to prevent recurrence. Such responses may be documented directly within electronic databases used for the corrective action program.

QV shall review the written responses to all audit findings, evaluate the adequacy of each response, assure that corrective action to prevent recurrence is identified and taken for each significant finding, and confirm that corrective action is accomplished as scheduled.

Audit records shall be generated and retained by QV for all audits.

**CURRENT REGULATORY REQUIREMENTS AND PG&E COMMITMENTS
PERTAINING TO THE QUALITY ASSURANCE PROGRAM**

Table 17.1-1

Sheet 1 of 12

The Quality Assurance Program for the Humboldt Bay ISFSI, described in the HB QA Plan, program directives, and administrative procedures complies with the requirements set forth in the Code of Federal Regulations. In addition, it complies with the regulatory documents and industry standards listed below.

Changes to this list are not made without the review and concurrence of the Director - Quality Verification.

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
(S.G.) 28	6/72	ANSI N45.2	1971	Quality Assurance Program Requirements for Nuclear Power Plants	
1.38	5/77	ANSI N45.2.2	1972	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	Warehouse personnel will normally visually scrutinize incoming shipments for damage of the types listed in Section 5.2.1, this activity is not necessarily performed prior to unloading. Separate documentation of the shipping damage is not necessary. Release of the transport agent after unloading and the signing for receipt of the shipment provides adequate documentation of completion of the shipping damage inspection. Any damage noted will be documented and dispositioned. Persons performing this visual scrutiny are not considered to be performing an inspection function as defined under Reg. Guide 1.74; therefore they do not require certification as an inspector under Reg. Guide 1.58.

Table 17.1-1

Sheet 2 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.39	9/77	ANSI N45.2.3	1973	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	Housekeeping zones established at the power plants differ from those described in the standard; however, PG&E is in compliance with the intent of the standard.
1.30	8/72	ANSI N45.2.4	1972	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	<p>The evaluation of (data sheet) acceptability is indicated on the results and data sheets by the approval signature (paragraph 2.4).</p> <p>No visual examination for contact corrosion is made on breaker and starter contacts unless there is evidence of water damage or condensation. Contact resistance tests are made on breakers rated at 4 kV and above. No contact resistance test is made on lower voltage breakers or starters (paragraph 3[4]).</p> <p>No system test incorporates a noise measurement. If the system under test meets the test criteria, then noise is not a problem (paragraph 6.2.2).</p>

Table 17.1-1

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.58	9/80	ANSI N45.2.6	1978	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel	<p>ANSI N45. 2. 6 applies to individuals conducting independent QC inspections, examinations, and tests.</p> <p>ANSI/ ANS 3.1-1978 applies to personnel conducting inspections and tests of items or activities for which they are responsible (e.g., surveillance tests, maintenance tests, etc.).</p> <p>Except that inspector/examiner reevaluation due dates may be extended a maximum of 90 days. The next reevaluation due date shall be based on the original scheduled due date but shall not exceed the original due date plus 90 days.</p> <p>NDE personnel shall be qualified and certified in accordance with CP-189-1995.</p>

Table 17.1-1

Sheet 4 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.58 (cont.)					<p>NDE personnel who perform examinations of the containment structure per the requirements of Section XI, Subsections IWE and IWL, visual examination and ultrasonic thickness measurement only, shall be qualified and certified to ANSI/ASNT CP-189-1991.</p> <p>ISI ultrasonic examiners shall meet the additional requirements of ASME Section XI, Appendix VIII, 2001 Edition with no Addenda.</p>
1.116	5/77	ANSI N45.2.8	1975	<p>Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems.</p>	
1.88	10/76	ANSI N45.2.9	1974	<p>Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records</p>	<p>Except PG&E will comply with the 2-hour rating of Section 5.6 of ANSI N45.2.9 issued July 15, 1979.</p> <p>Except PG&E will also meet the intent of the guidelines for the storage of QA records in electronic media as, endorsed by Generic Letter 88-18, "Plant Record Storage on Optical Disks," issued October 20, 1988, and Regulatory Issues Summary 2000-18, "Guidance on Managing Quality Assurance Records in Electronic Media," issued October 23, 2000.</p>

Table 17.1-1

Sheet 5 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.88 (cont.)					Note: PG&E will maintain records of spent fuel and high-level radioactive waste in storage in accordance with ANSI N 45.2.9-1974 rather than 10 CFR 72.72(d). Refer to ISFSI FSAR Update, Section 9.4.2.
1.74	2/74	ANSI N45.2.10	1973	Quality Assurance Terms and Definitions	
1.64	6/76	ANSI N45.2.11	1974	Quality Assurance Requirements for the Design of Nuclear Power Plants	Except PG&E will allow the designer's immediate supervisor to perform design verification in exceptional circumstances and with the controls as described in NUREG-0800, Revision 2, July 1981.
1.144	1/79	ANSI N45.2.12	1977	Auditing of Quality Assurance Programs for Nuclear Power Plants	<p>Except the scheduled date for triennial vendor audits and annual supplier evaluations may be extended a maximum of 90 days. The next scheduled due date shall be based on the original scheduled due date but shall not exceed the original due date plus 90 days.</p> <p>Except that the corrective action program stipulated in the QA Program may be used instead of the requirements of Section 4.5.1 as long as the appropriate time limits are applied to significant conditions adverse to quality. Also, no additional documentation is necessary if needed corrective actions are taken and verified prior to audit report issuance.</p> <p>See Note for Reg Guide 1.144 [S-17.1 (3); S-17.1 (7)]</p>

Table 17.1-1

Sheet 6 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.123	7/77	ANSI N45.2.13	1976	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	<p>In addition to ANSI N45.2.13, Section 10.3.3, PG&E will accept items and services which are complex or involve special processes, environmental qualification, or critical characteristics which are difficult to verify upon receipt by suppliers' Certificate of Conformance if and only if the supplier has been evaluated and qualified utilizing Performance Based Supplier Audit techniques.</p> <p>See Note for Reg Guide 1.123 <i>[S-17.1 (3)]</i></p>
1.146	8/80	ANSI N45.2.23	1978	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	<p>Except that auditor recertification due dates may be extended a maximum of 90 days. The next recertification due date shall be based on the original scheduled due date but shall not exceed the original due date plus 90 days.</p> <p>Except that in lieu of the requirements of 2.3.4 of ANSI N45.2-1978, the prospective lead auditor shall have participated in at least one nuclear quality assurance audit within the year preceding the individual's effective date of qualification.</p>

Table 17.1-1

Sheet 7 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.33	2/78	ANSI N18.7	1976	Quality Assurance Program Requirements (Operation)	<p>Except that PG&E will not perform biennial review of all ISFSI procedures, except under the conditions described in note below (See note at end of table).</p> <p>Except for temporary changes to procedures, PG&E will require a review by an individual who holds a Senior Reactor Operators license only if the procedure is one of the types listed in Section 17.5 (8) of this QA Program. Furthermore, this individual need not be the supervisor in charge of the shift.</p> <p>Except that audit frequencies specified in Regulatory Guide 1.33, Revision 2, need not be met. Audits shall be performed at the frequencies specified in Section 17.18 of this QA Program.</p> <p>Except that audits and reviews of the Emergency Preparedness Program shall be performed in accordance with 10 CFR 50.54(t).</p> <p>Except that a grace period of up to 90 days will be allowed for audit scheduling, except where the schedule is mandated by regulation. The next schedule due date shall be based on the original scheduled date but shall not exceed the original due date plus 90 days.</p>

Table 17.1-1

Sheet 8 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.33 (cont.)					<p>Except that when purchasing commercial-grade calibration services from certain accredited calibration laboratories, the procurement documents are not required to impose a quality assurance program consistent with ANSI N45.2-1971. Alternative requirements described in this QA Program for Regulatory Guide 1.123 may be implemented in lieu of imposing a quality assurance program consistent with ANSI N45.2-1971.</p> <p><i>[S-17.1 (3)]</i></p>
1.8	2/79	ANSI/ANS 3.1	1978	Personnel Selection and Training	<p>Except that for the Quality Verification Director, the one year of qualifying nuclear power plant experience in the overall implementation of the Quality Assurance program can be obtained outside the Quality Assurance organizations.</p> <p>Except that the Radiation Protection Manager's qualifications shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987, for the Radiation Protection Manager.</p>

Table 17.1-1

Sheet 9 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
1.8 (cont.)					<p>Except that the person serving as the manager responsible for the independent review and audit program shall have a minimum of 6 years of professional level managerial experience in the power field. This exception is based on NRC letter to PG&E dated February 6, 1992, issuing Licensing Amendment No. 68/67.</p> <p>Except that the licensed reactor operators and senior reactor operators shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1993 as endorsed by Regulatory Guide 1.8, Revision 3, May 2000 with the exceptions clarified in the current revision to the Operator Licensing Examination Standards for Power Reactors, NUREG-1021, Section ES-202. This exception is based on NRC letter to PG&E dated May 26, 2006, issuing License Amendment Nos. 187/189.</p> <p>HB ISFSI personnel shall meet the requirements of the HB ISFSI Training Program.</p>
---	--	NCIG-01	2	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants	

Table 17.1-1

Sheet 10 of 12

Reg. Guides	Date	Standard No.	Rev.	Title/Subject	Exceptions
---	--	NCIG-02	2	Sampling Plan for Visual Reinspection of Welds	
---	--	NCIG-03	1	Training Manual for Inspection of Structural Weld at Nuclear Power Plants Using the Acceptance Criteria of NCIG-01	

Table 17.1-1

Sheet 11 of 12

Note for Reg. Guide 1.33:

These controls replace the biennial procedure review requirement found in Section 5.2.15 of ANSI N18.7-1976.

1. All applicable ISFSI procedures (shall)* be reviewed following an unusual incident, such as an accident, unexpected transient, significant operator error, or equipment malfunction, and following any modification to a system, as specified by Section 5.2 of ANSI N18.7/ANS 3.2, which is endorsed by Regulatory Guide 1.33.
2. Non-routine procedures (e.g. emergency operating procedures, procedures which implement the emergency plan, and other procedures whose usage may be dictated by an event) (shall)* be reviewed at least every two years and revised as appropriate.
3. Routine ISFSI procedures that have not been used for two years (shall)* be reviewed before use to determine if changes are necessary or desirable.

* The word should has been changed to shall denoting a regulatory commitment.

Note for Reg. Guide 1.144:

The following interpretation is added with respect to Regulatory Guide 1.144, Section C.3.b(2):

When purchasing commercial-grade calibration services from calibration laboratories accredited by a nationally-recognized body, the accreditation process and accrediting body may be credited with carrying out a portion of the Purchaser's duties of verifying acceptability and effective implementation of the calibration service supplier's quality assurance program.

Nationally-recognized accrediting bodies include the National Voluntary Laboratory Accreditation Program (NVLAP) administered by the National Institute of Standards and Technology (NIST) and other accrediting bodies recognized by NVLAP via a Mutual Recognition Agreement (MRA)

In lieu of performing an audit, accepting an audit by another licensee, or performing a commercial-grade supplier survey, a documented review of the suppliers' accreditation shall be performed by the Purchaser. This review shall include, at a minimum, verification of all the following:

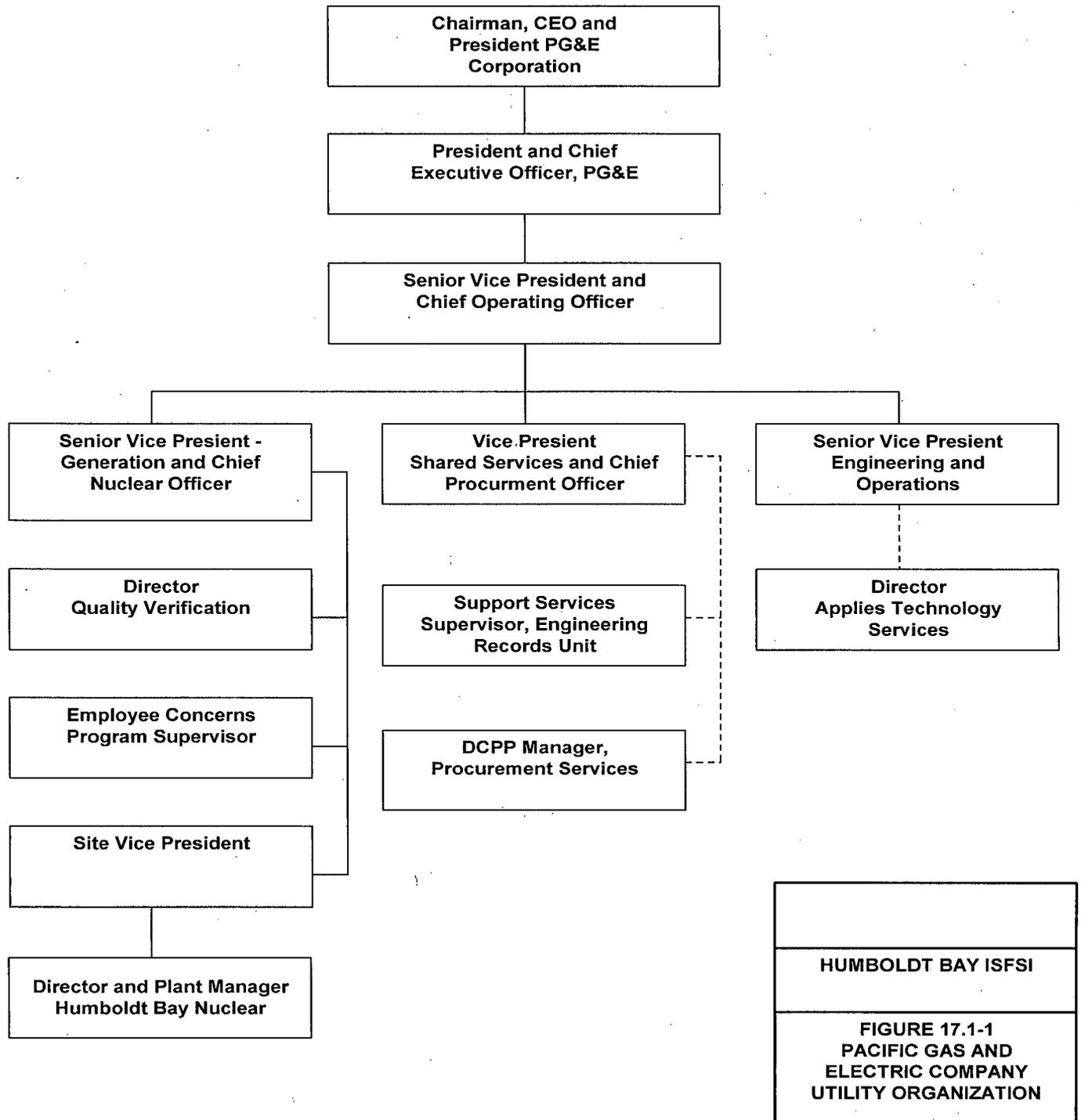
- (1) The accreditation is to ANSI/ISO/IEC 17025
- (2) The accrediting body is either NVLAP or A2LA, which is an accrediting body recognized by NVLAP through an MRA.
- (3) The published scope of accreditation for the calibration laboratory covers the needed measurement parameters, ranges, and uncertainties.

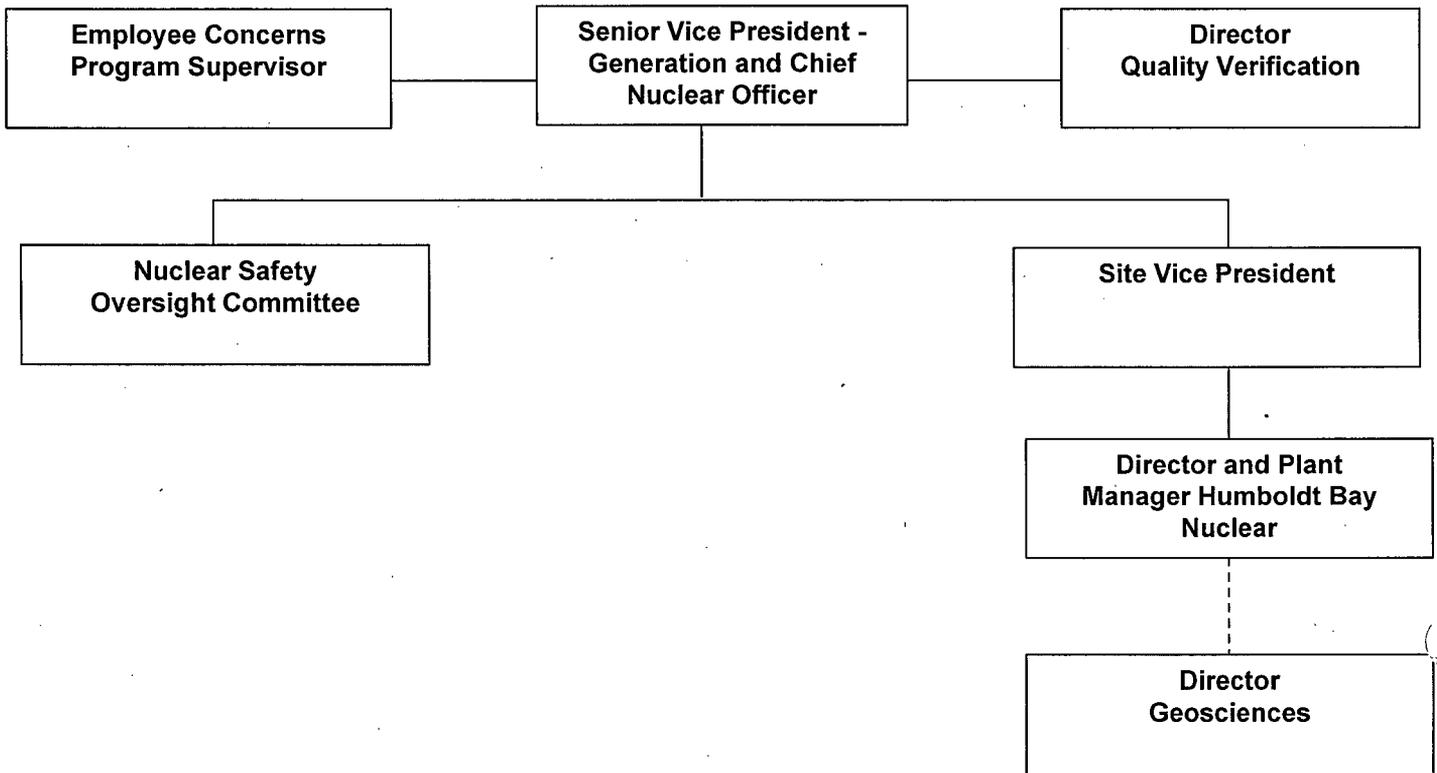
Note for Reg. Guide 1.123:

The requirements of ANSI N45.2.13, Section 3.2, "Content of the Procurement Documents," Subsection 3.2.3, "Quality Assurance Program Requirements" are accepted with the following exception:

When purchasing commercial-grade services from calibration laboratories accredited by a nationally-recognized accrediting body, the procurement documents are not required to impose a quality assurance program consistent with ANSI N45.2-1971. Nationally-recognized accrediting bodies include the NVLAP administered by the NIST and other accrediting bodies recognized by NVLAP via a MRA. In such cases, accreditation may be accepted in lieu of the Purchaser imposing a QA Program consistent with ANSI N45.2-1971, provided all the following are met:

- (1) The accreditation is to ANSI/ISO/IEC 17025
- (2) The accrediting body is either NVLAP or A2LA, which is an accrediting body recognized by NVLAP through an MRA.
- (3) The published scope of accreditation for the calibration laboratory covers the needed measurement parameters, ranges, and uncertainties.
- (4) The purchase documents impose additional technical and administrative requirements, as necessary, to satisfy DCPD QA Program and technical requirements, including the requirement that the calibration/certificate report include identification of the laboratory equipment/standard used
- (5) The purchase documents require reporting as-found calibration data when calibrated items are found to be out-of-tolerance.





HUMBOLDT BAY ISFSI
FIGURE 17.1-2 NUCLEAR QUALITY IN THE UTILITY ORGANIZATION

Enclosure 3
PG&E Letter HBL-10-005

**HBPP UNIT 3 TECHNICAL
SPECIFICATIONS BASES, REVISION 3**



Nuclear Power Generation
Humboldt Bay
Power Plant

NUMBER L-7
VOLUME 4
REVISION 3
EFFEC DATE 1-12-09
PAGE 1 of 1

TITLE
TECHNICAL SPECIFICATIONS
BASES

APPROVED BY

ORIGINAL SIGNED 12-16-08
DIRECTOR/PLANT MANAGER / DATE
HB NUCLEAR

(Procedure Classification – Quality Related)

1.0 DESCRIPTION

The technical specifications (TS) bases were initially implemented on January 17, 2003. As required by TS 5.6.2.d, the TS bases are required to be submitted to the NRC on a 24-month frequency, consistent with the DSAR revision frequency as identified in 10 CFR 50.71 (e).

2.0 ATTACHMENT

2.1 Technical Specifications Bases for Humboldt Bay Power Plant Unit 3.

3.0 PROCEDURE OWNER

Supervisor of Licensing

Technical Specifications

Bases

Humboldt Bay Power Plant

Unit 3

Eureka, California

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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY B 3.0-1

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY B 3.0-2

B 3.1 DEFUELED SYSTEMS B 3.1-1

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 and 3.0.2 establish general requirements applicable to all Specifications and apply at all times, unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each specification).
LCO 3.0.2	<p>LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:</p> <ol style="list-style-type: none">Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; andCompletion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified. <p>Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.</p> <p>The Completion Times of the Required Actions are also applicable when a specified condition in the Applicability is entered intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances; preventive maintenance, corrective maintenance, or investigation of problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise the safe storage of irradiated fuel. Intentional entry into ACTIONS should not be made for convenience.</p>

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs	SRs 3.0.1 through 3.0.3 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise specified.
SR 3.0.1	<p>SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that surveillances are performed to verify that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.</p> <p>Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.</p>
SR 3.0.2	<p>SR 3.0.2 permits a 25 percent extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., other ongoing Surveillance or maintenance activities).</p> <p>The 25 percent extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. Any exceptions to SR 3.0.2 are stated in the individual Specifications.</p> <p>The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.</p>

(continued)

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring an affected variable outside the specified limits when Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours applies from the point of time that it is discovered that the Surveillance has not been performed in accordance to SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3. is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals.

If Surveillance is not completed within the allowable delay period, then the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If Surveillance is failed within the delay period, then the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

B 3.1 DEFUELED SYSTEMS

B 3.1.1 Fuel Storage Pool Liner Water Level

BASES

BACKGROUND A stainless steel liner covering the inside surface of the fuel storage pool was installed in 1963. It formed a nominal ¼-inch gap between the walls and floor of the pool and the liner. The water level in the gap is maintained at a level below the fuel storage pool water level and below the exterior groundwater level. This is done to capture leakage from the fuel storage pool and to preclude leakage from the fuel storage pool or the liner gap to surrounding groundwater.

APPLICABLE SAFETY ANALYSES Based on the fuel storage pool water radionuclide concentrations being maintained at low levels, the gap water level being maintained at a level that is less than both the fuel storage pool level and the surrounding groundwater level, and the limited amount of leakage to the gap per day, there is reasonable assurance that the leakage will have no environmental significance and will not affect the health and safety of the public.

LCO The fuel storage pool liner water level is required to be at an elevation of less than + 9 inches. Monitoring and maintaining this liner water level minimizes the potential for leakage from the fuel storage pool to surrounding groundwater.

APPLICABILITY This LCO applies when the water in the pool or liner is contaminated with radioactive materials.

ACTIONS A.1
This action is intended to restore the fuel storage pool liner level as soon as possible to minimize the potential for pool leakage to the surrounding groundwater.

SURVEILLANCE REQUIREMENTS SR 3.1.3.1
This SR ensures that the liner water level is within the established limit. The water level in the fuel storage pool liner gap must be checked periodically. The 24 hour Frequency is based on engineering judgment and is considered adequate because of the normally low rate of leakage into the gap and the available indication of pool level changes.
